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STANDARD REVIEW PLAN FOR THE REVIEW OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

LWR EDITION

SEPTEMBER 1975

OFFICE OF NUCLEAR REACTOR REGULATION
U. S. NUCLEAR REGULATORY COMMISSION

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INTRODUCTION

Standard Review Plans (SRP) are prepared for the guidance of staff reviewer of the Office of Nuclear Reactor Regulation who perform the detailed safety review of applications to construct or operate nuclear power plants. A primary prupose of the SRP is to improve the quality and uniformity of staff reviews, and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. A second purpose of the SRP is to implement Nuclear Reactor Regulation policy on making information about regulatory matters widely available and to improve communication and understanding of the staff review process by interested numbers of the public and the nuclear power industry. The application and use of the Standard Review Plans by the staff should have a stabilizing effect on the licensing process that will benefit the interests of both the public and the nuclear power industry.

Section 50.34 of 10 CFR Part 50 of the commission's regulations requires that each application for a construction permit for a nuclear facility shall include a Preliminary Safety Analysis Report (PSAR), and that each application for a license to operate such a facility shall include a Final Safety Analysis Report (FSAR). Section 50.24 specifies, in general terms, the information to be supplied in a safety analysis report (SAR).

In information provided in the SAR must be sufficiently detailed to permit the staff to determine whether the plant can be built and operated without undue risk to the health and safety of the public. Prior to submission of an SAR, an applicant should have designed and analyzed the plant in sufficient detail to conclude that it can be built and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

The Standard Review Plans are the result of many years of experience by the staff in establishing and promulgating standards to enhance the safety of the nuclear facilities, and in assessing Safety Analysis Reports. The information requirements for current SAR's have increased substantially in scope and in depth of detail as compared to the requirements for typical SAR's submitted in the mid 1960's. A great deal of progress has also been made in the methods of review and in the development of regulations, guides and standards since the early years of review. These Standard Review Plans may be considered a part of a continuing regulatory standards development acitivity that not only documents current methods of review, but provides the base of orderly modifications of the review process in the future.

The Standard Review Plans are written so as to cover a variety of site conditions and plant designs. For any given application, the staff reviewers will select and emphasize particular aspects of each SRP as is appropriate for that application. In some cases, the major portion rather than in the context of reviews of particular applications from utilities. In other cases a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed. For these and other similar reasons, the staff does not expect to carry out in detail all of the review steps listed in each plan in the review of every application.

Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants - LWR Edition" identifies the information that is required by the staff in its evaluation of an application. The Standard Review Plans describe the review process conducted in the Office of Nculear Reactor Regulation following receipt of the application. The safety review is performed by seventeen functional branches. The areas of responsibility for each branch are keyed to the sections of the Standard Format, Revision 2. In addition to the areas of primary responsibility, branches may have secondary responsibilities in areas where other branches need support. One of the objectives of the SRP is to enunciate clearly the review responsibilies of the various branches and to define the sometimes complex interfaces between them.

The first major subsection of each review plan, "Areas of Review", describes the scope of review, i.e., what is being reviewed by the branch having primary review responsibility. This section contains a description of the systems, components, analyses, data, or other information that is reviewed as part of the particular Safety Analysis Report section in question. This section also contains a discussion of the information needed or the review expected from other branches in order for the primary review branch to complete its review.

The second subsection of each review plan, "Acceptance Criteria", contains a statement of the purpose and technical bases for the review. The "bases" consist of specific criteria such as NRC Regulatory Guides, General Design Criteria, ASME Code requirements, Branch Technical Positions, or other criteria.

The third subsection of each review plan is the "Review Procedures", which discusses how the review is accomplished. The procedures in use for reviewing and approving systems, components, data, etc., that are described in the first subsection of the plan using the criteria given in the second subsection are described. The section is generally a step-by-step procedures that the reviewer goes through to provide reasonable verification that the applicable safety criteria have been met.

The fourth review plan subsection, "Evaluation Findings", presents the type of conclusion that are sought in the particular review area. The final review plan subsection lists the references utilized in the review process.

Some plans have Branch Tachnical Positions, or Appendices, attached. These documents typically set forth the solutions and approaches determined to be acceptable in the pst by the staff in dealing with a specific safety problem or safety-related design area. These solutions and approaches are codified in this form so that staff reviewers can take uniform and well-understood positions as the same safety problems arise in future cases. Some Branch Technical positions and Appendices may be converted into Regulatory Guides if it appears that this step would aid the review process. Like Regulatory Guides, the Branch Technical Positions and Appendices represent solutions and approaches that are acceptable to the staff, but they are not required as the only possible solutions and approaches. However, applicants should recognize that, as in the case of Regulatory Guides, substantial time and effort on the part of the staff has gone into the development of the Branch Technical Positions and Appendices and that a corresponding amount of time and effort will probably be required to review and accept new or dfferent solutions and approaches. Thus, applicants proposing other solutions and approaches to safety problems or safety-related design areas than those described in the Branch Technical Positions and Appendices must expect longer review times and more extensive questioning in these areas. The staff is willing to consider proposals for other solutions and approaches on a generic basis, apart from a specific license application, to avoid the impact of the additional review time on individual cases.

In February 1972 the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants was issued for information and comment. This document identified the information needed in safety analysis reports. After reviewing the comments received, the Standard Format was revised and reissued in October 1972 (Rev. 1). A further revision in the Standard Format has been made in connection with the preparation of the SRP. This new version is planned for issue in September 1975.

The SRP are keyed to Revision 2 of the Standard Format, and the SRP are numbered according to the section numbers in Revision 2. Review plans have not been prepared for SAR sections that consist of background or design data which are included for information or for use in the review of other SAR sections. The individual SRP address, in detail, the matters that are reviewed, the basis for review, how the review is accomplished, and the conclusions that are sought. The Table of Contents that follows identifies the 224 SRP.

For some time after the SRP are published, applications being reviewed will conform to Revision 1 of the Standard Format, rather than Revision 2. Staff reviewers will adapt the SRP to the particular needs of applications based on Revision 1. Staff reviewers also will make appropriate allowance for the difference in information requirements between Revision 1 and Revision 2 when determining the acceptability of applications for review.

Like the Standard Format, Revision 2, the SRP are directed toward water-cooled reactor power plants. Staff reviewers will adapt the SRP for use in the reviews of other reactor types where applicable.

The SRP will be revised and updated periodically as the need arises to clarify the content or correct errors and to incorporate any modification approval by the Director of the Office of Nuclear Reactor Regulation. Comments and suggestions for improvement will be considered and should be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Notices of errors or omissions will be appreciated; they should be sent to the same address.

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SECTION 2.1.1 SITE LOCATION AND DESCRIPTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

Reactor location is reviewed (1) as identified by latitude and longitude and by the UTM* coordinate system; (2) with respect to political subdivisions; and (3) with respect to prominent natural and man-made features of the area to ascertain the accuracy of the applicant's safety analysis report (SAR) description and for use in population analyses (Standard Review Plan 2.1.3).

The site area which contains the reactors and associated principal plant structures is reviewed to determine the distance from the reactor to boundary lines of the exclusion area, including the direction and distance from the reactor to the nearest exclusion area boundary line. The location and orientation of plant structures within the exclusion area are reviewed to identify potential release points and their distances to plant boundary lines. The location, distance, and orientation of plant structures with respect to highways, railways, and waterways which traverse or lie adjacent to the exclusion area are reviewed to assure that they are adequately described to permit analyses (Standard Review Plan 2.2) of the possible effects on the plant of accidents on these transportation routes.

II. ACCEPTANCE CRITERIA

The size of the plant exclusion area and the location of the plant within the area should be such as to provide reasonable assurance that the guidelines of 10 CFR Part 100 will be met.

Highways, railways, and waterways which traverse the exclusion area should be sufficiently distant from plant structures so that routine use of these routes is not likely to interfere with normal plant operation (Ref. 2).

*Universal Transverse Mercator coordinate system as found on USGS topographical maps.

Information included in this section should allow two types of safety analyses to be conducted. The first addresses the consequences in the unlikely event that a serious release of radioactive material should occur. The second addresses the effect that accidents on, or routine use of, routes on or near the site will have on the operation of the plant. Adequacy of the data for these purposes should be decided jointly with the reviewers having primary responsibilities for the particular analyses involved.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

The information in this section of the SAR forms the basis for evaluations performed in various other sections. The purpose of this review is to establish the validity of the basic data. Check the UTM coordinates to assure that they include the zone number and that the Northing and Easting are presented to within 100 meters. The latitude and longitude should be checked to assure that they are expressed to the nearest second.

Cross-check the minimum exclusion area distance with the minimum distance used in the Accident Analyses, SAR Section 15. In general, a minimum exclusion boundary distance of 0.4 miles provides assurance that engineered safety features can be added (if necessary) that will bring doses within Part 100 guidelines. At the operating license stage, the acceptability of the exclusion area and low population zone with respect to Part 100 dose criteria will be reaffirmed using the latest available engineered safety features design data and X/Q values. The final determination of acceptability must be made in conjunction with the analyses of the accidents postulated and evaluated in Section 15. Scale the map provided to check distances specified in the SAR and to determine the distance-direction relationships to area boundaries, roads, railways, waterways, and other significant features of the area. At the operating license stage, the location and orientation of plant structures and effluent release points with respect to the exclusion area and plant property boundaries, transportation routes and political subdivisions will be reviewed to identify any changes since the construction permit (CP) review. Where changes have occurred, new analyses may be required to ensure that the findings reached during the CP review are not affected by these changes.

If, in the reviewer's judgment, maps of larger scale are desirable, they may be requested from the U.S. Geological Survey (USGS). The USGS map index should be consulted for the specific names of the 7-1/2 minute quadrangles that bracket the site area. If available, these maps provide topographic information in addition to details of prominent natural and man-made features in the site area. This information should be supplemented by updated information as available, e.g., aerial photographs or information obtained on the site visit. (Ref. 4). Check the plant layout to determine that the orientation of plant structures with respect to nearby roads, railways, and waterways is clearly shown. Check to see that there are no obvious ways in which transportation routes which traverse the exclusion area can interfere with normal plant operations.

IV. EVALUATION FINDINGS

Summary descriptions of the site location, the site itself, and transportation routes on or near the site will be prepared for the staff safety evaluation report. Any deficiencies of site parameters with respect to the proposed plant will be noted.

V. REFERENCES

1. U.S. Geological Survey Topographical Map Indices (one for each state).

2. 10 CFR Part 100, "Reactor Site Criteria," Section 100.3(a).

3. AEC Manual Appendix 0621, "Damage Assessment Handbook," Part III, "Universal Transverse Mercator Coordinate System."

4. Appendix A, Standard Review Plan 2.1.1, "Site Visits - Suggested Procedure for Site Analysts," attached.

5. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

APPENDIX A

STANDARD REVIEW PLAN 2.1.1

SITE VISITS - SUGGESTED PROCEDURE FOR

SITE ANALYSTS

A. GENERAL

Site visits are designed to gain information which supplements that contained in the preliminary safety analysis report (PSAR). This information, since it is derived independently, makes it possible to verify and substantiate the findings reached by the applicant in the PSAR. In addition, new information obtained during the course of the visit may lead to the identification of safety issues which have not been adequately addressed in the PSAR.

This procedure should be used in conjunction with the review procedures for SAR Sections 2.1 and 2.2, which discuss the specific site characteristics that may be important to safety. The "site" referred to here is the property owned by the applicant and the surroundings to a distance of several miles. Not all items listed can be done on each review. The judgement of the Site Analyst must be exercised to make sure that the limited time available is spent on issues that are important for the particular case.

The five suggested phases of a site visit are:

1. Site orientation and identification of prominent site features.
2. Review and discussion of draft questions.
3. Visit to plant site.
4. Supplementary visits.
5. Administrative followup.

The goals and procedures for each phase are described in the following sections. A Site Analyst may find it convenient to modify the phases and the procedures to suit himself or the particular plant. The procedures are written with the construction permit phase of the review in mind. A site visit is also made at the operating license stage but is primarily confirmatory.

B. SITE ORIENTATION AND IDENTIFICATION OF PROMINENT SITE FEATURES

In this phase, the Site Analyst familiarizes himself with the site and its surroundings on the basis of information contained in the SAR and other sources available in the office. He notes those things about the site which may be significant to safety so that they can be seen during the visit to the plant site.

The Site Analyst should orient the plant site with respect to prominent landmarks (roads, rivers, railroads, towns, etc.). Based on information contained in the SAR, he locates the safety-related features of the site which the applicant has analyzed, reviews the findings, and identifies problem areas which need additional attention.

The Site Analyst can obtain his own set of maps for the plant site by checking the U.S. Geological Survey (USGS) index in the Site Analysis Branch office and ordering the appropriate maps. The "7-1/2 second" coverage maps, representing a 1:24,000 scale, work well. These can be ordered from the USGS office in Washington.

Compare the SAR maps and USGS maps and identify any significant discrepancies which may exist between them. Features related to plant safety should be located on the USGS maps. Study the USGS maps to see if any other features are shown which may also relate to plant safety.

At this point, the Site Analyst can list those site-related features that may affect plant safety. Some may be considered less important than others but they should still be noted. This list is used in generating draft first round question (Q-1) input, selecting items to be seen on the visit to the plant site and acquiring supplemental information.

C. <u>REVIEW AND DISCUSSION OF DRAFT Q-1'S</u>

This phase is used to make sure the applicant understands the reasons for the questions, knows the information which is needed in the reply and understands how the information will be used. In so doing, the chances of getting the desired information in the Q-1 reply will be improved.

Every effort is made to make sure that the applicant understands each of the Accident Analysis Branch draft questions. Explain why the question is being asked. It may stem from errors, lack of completeness, or omission. It may be based on discrepancies between the PSAR and USGS maps. Tell the applicant exactly what information should be included in his reply and, if necessary, how it should be presented. Let the applicant know how the information which he supplies will be used.

This meeting may be used as an opportunity to ask the applicant any other questions which are not part of the draft question list. For the most part, these may be general information questions which yield useful background information. They may also include questions on the terminology used in the PSAR. The Site Analyst can also point out any typographical or editing errors which he has noted in the PSAR.

These discussions may occur either before or after the visit to the plant site but should generally be held in conjunction with the site visit.

D. <u>VISIT TO THE PLANT SITE</u>

The plant site is visited in order to inspect the area and observe the prominent features of the site. These features can be identified and located in the first phase but a site visit is necessary to aid the Site Analyst in obtaining a perspective on the overall effect that they may have on plant safety. The plant site and its surrounding area should be viewed with an open attitude so that unexpected or new features can be recognized. Upon its completion, the Site Analyst should have a thorough understanding of the relationship between site-related features and plant safety. The Site Analyst may want to be prepared to take photographs on and around the plant site.

The applicant should take the Site Analyst to the plant site and to the proposed location of the reactors, cooling towers, intake/discharge structures, settling basins, etc. Check for any features in the immediate areas of these locations which may adversely affect their safe operation (sources of missiles, tall structures, excavations, etc.). Try to visualize the area as it will exist when construction is completed.

Try to see as much of the remainder of the plant site as possible. If any part is not accessible by automobile or rough terrain vehicle, the use of a helicopter might be recommended. If a helicopter or plane is available, it may be used for aerial observation of the plant site. Look for evidence of any activities which need to be evaluated such as hunting, grazing, mining, drilling, flooding, etc.

Adjacent and nearby properties should be looked at to develop a feel for the density of homes. Public and commercial facilities around the plant site should be viewed. Nearby towns, industries, military facilities, airports, recreation sites, etc., should be visited and activity in and around these places should be observed. Note their locations so that supplementary visits (see next section) can be made, if desired. Evidence of major construction or land development projects should be noted and checked out.

Transportation routes (including pipelines) which pass through or near the plant site should be inspected and some time should be spent at each of them to obtain a sampling of the density and type of traffic using the route. It may be possible to determine that they do not represent significant hazards or that a hazard may exist, but additional information is required to assess it. Note the frequency of aircraft flying near the plant site. Compare all of these observations to those which are stated or implied in the PSAR.

E. SUPPLEMENTARY VISITS

The purpose of this phase is to gain information independent of the applicant. The Site Analyst can then use his own sources of information to verify, supplement, or oppose the findings stated in the PSAR. The Site Analyst should use this opportunity to develop any information which he deems appropriate based on what he has learned from the SAR, his map studies, and visit to the site.

The Site Analyst should make his own arrangements for supplementary visits. If the applicant offers his assistance, it is preferable not to accept it. Remember, the objective is to develop your own sources of information, not to redevelop those of the applicant. It may be desirable to allow an extra day for this activity.

Selection of parties to contact is based on the local telephone listings. First, contact the Federal, state, and local government offices which appear, by their titles, to be potential sources of information. Government offices are contacted first because they are probably accustomed to these types of requests, are familiar with local activities, and are in a position to refer you to contacts in local businesses and industries. Examine the local listings of government offices and pick out those offices whose titles seem to be applicable. As an example, on the River Bend plant site visit, the following contacts were made:

State of Louisiana

Highway Department

Safety Section

Highway Safety Commissioner

Liquified Petroleum Gas Commission

State Police, Explosives & Metals Division

U. S. Government

U. S. Coast Guard

Coast Guard and Marine Shipping

Vessel Documentation

Shipping Commission

Department of Transportation, Federal Highway Administration, Motor Carrier Safety Officer

Commerce Department, Economic Development Commission

Corps of Engineers

Parish of East Baton Rouge

Agriculture Stabilization & Conservation Service

City of Baton Rouge

Port of Baton Rouge

Any office having responsibilities for safety, transportation, hazardous materials controls, planning, economic development, explosives, liquified gas, chlorine, etc., should be contacted.

Check with the local governments (City Hall and County Court House) for any local agencies which might be of assistance. Planning, Safety, Development, etc., Commissions are sometimes organized within them.

Military facilities, local officials, and the larger industrial firms near the plant site should be contacted and arrangements made to talk with their public relations personnel. Discuss the proposed nuclear power plant with them, explain any interaction which you believe their operation will have on the plant and ask for their comments. Ascertain if any future changes are planned in their operations. Obtain information on any hazardous materials which these facilities may store or use. Their operating experiences (accidents, consequences, procedures, etc.) with these materials may be pertinent information to obtain.

Check with the local Agriculture Stabilization and Conservation Service (ASCS) office and obtain the identification of the latest aerial photographs of the plant site. Aerial photographs covering an area about two miles around the plant site are useful to have because they are generally more up-to-date than maps and may reveal features which cannot be identified from maps. These maps can be ordered through the Administrative Services Branch.

Other possible contacts are commercial associations that are listed in the telephone directory such as; American Trucking Association, American Waterways Operators Association, Airplane Owners & Pilots Association, and Liquified Petroleum Gas Association; and civic organizations such as the Chamber of Commerce and Better Business Bureau.

F. ADMINISTRATIVE FOLLOWUP

In this phase, the Site Analyst organizes, evaluates, and records the information he has obtained. He identifies areas where more information is needed. Any contacts that have not been pursued can be done so by telephone. At some point, it may become necessary to revisit the area to obtain additional information which your sources may have developed for you.

Check to see that you have all the information you need. Make sure that the SAR and amendments reflect all important aspects which you have identified. Draft question lists should be modified appropriately before formal transmittal to the applicant. The last task is to organize what information you have for input into the staff safety evaluation report, as outlined in Standard Review Plan 2.1.1.

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SECTION 2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Office of the Executive Legal Director (OELD)
Industrial Security and Emergency Planning Branch (ISEPB)
Division of Operational Safety, Emergency Preparedness Branch (DOS)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The applicant's legal authority to determine all activities within the designated exclusion area, as described in his safety analysis report (SAR), is reviewed to establish that the exclusion area meets the requirements of 10 CFR Part 100. 10 CFR § 100.3(a) requires that a reactor licensee have authority to determine all activities within the designated exclusion area, including the exclusion and removal of personnel and property. Determination of the accetability of the designated exclusion area establishes the minimum distance to the exclusion area boundary that is used in dose computations.

All activities that may be permitted within the designated exclusion area, and that will not be related to routine operation of the plant, are reviewed to assure that they will not be incompatible with normal and emergency plant conditions.

In any case where the applicant does not own all the land within the designated exclusion area, assistance may be required of OELD in determining whether or not the designated exclusion area is acceptable under 10 CFR Part 100. Also, in some cases public roads which lie within the proposed exclusion area may have to be abandoned or relocated to permit plant construction. OELD assistance may be required to assure that no legal impediments to such abandonment or relocation are likely to ensue.

For cases where activities unrelated to plant operation may be permitted within the exclusion area, the reviewer should consult with the ISEPB regarding the adequacy of planning (construction permit stage) or plans (operating license stage) for protective measures for members of the public who may be within the exclusion area.

Where the need arises, it may be necessary to request the SAB to provide relative concentrations (X/Q) for distances less than the minimum distance to the exclusion area boundary.

II. ACCEPTANCE CRITERIA

Absolute ownership of all lands within the exclusion area is considered to carry with it the required authority to determine all activities on this land and is acceptable. If the required authority is contingent upon future procurement of ownership, or by lease, contract, or other means, determination of whether or not the applicant's claimed authority meets or is likely to meet the requirements of Part 100 by the time of issuance of the staff safety evaluation report must be made by OELD. OELD must also determine that there is reasonable assurance that proposed public road abandonment or relocation will be achieved.

In some cases, the designated exclusion area may extend into bodies of water such as a lake, reservoir, or river which is routinely accessible to the public. This is acceptable provided the applicant has made appropriate arrangements with the local, state, federal, or other public agency having authority over the particular body of water. The arrangements made should provide for the exclusion and ready removal in an emergency, by either the applicant or the public agency in authority, of any persons on those portions of the body of water which lie within the designated exclusion area.

Activities within the exclusion area which are not related to plant operation are acceptable provided that no individual engaged in such activities is likely to receive, as a consequence of the design basis accidents postulated and evaluated in SAR Section 15, radiation doses which exceed 10 CFR Part 100 guidelines.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

The reviewer should determine the basis on which the applicant claims authority within the exclusion area. If absolute ownership of all lands within the area is claimed, the acceptance criteria are satisfied. If any other method is claimed as providing the required authority, a memorandum should be prepared for OELD containing all of the appropriate information in the SAR, including copies of applicable SAR papes and figures, and requesting a written response as to whether or not the applicant's claimed authority meets the requirements of 10 CFR § 100.3(a). The memorandum should also include information in the PSAR which describes the applicant's plans, procedures, and schedule for obtaining any abandonment or relocation of public roads which may be required. At the operating license stage, the applicant's authority to determine all activities within the designated exclusion area will be reviewed. This review will emphasize those areas where the applicant did not possess absolute authority at the construction permit review.

If the designated exclusion area is traversed by a highway, railway, waterway, or other transportation route accessible to the public, the reviewer requests from ISEPB a written confirmation (buckslip) that the applicant's emergency plan includes adequate provisions for control of traffic on these routes in the event of an emergency. At the construction permit stage a finding that such provisions are feasible is adequate.

If activities unrelated to plant operation are to be permitted within the exclusion area, it will be necessary to determine that the potential radiation exposures to persons engaged in those activities resulting from the design basis accidents postulated and evaluated in SAR Section 15 do not exceed the guidelines of 10 CFR Part 100. The same method and model is to be used as was used to calculate the 2-hour exclusion area boundary dose.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has described the plant exclusion area, the authority under which all activities within the exclusion area can be controlled, and the methods by which access and occupancy of the exclusion area can be controlled during normal operation and in the event of an emergency situation. The applicant has the required authority to control activities within the designated exclusion area, including the exclusion and removal of persons and property, and has established acceptable methods for control of the designated exclusion area. It is concluded, in view of the results of the dose computations of Section 15, that the exclusion area meets the guidelines of 10 CFR Part 100, and is acceptable."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.1.3 POPULATION DISTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Industrial Security and Emergency Planning Branch (ISEPB)

I. AREAS OF REVIEW

The areas of the applicant's safety analysis report (SAR) relating to the population surrounding a nuclear facility are reviewed to determine:

1. The present population (based on 1970 census data), and a comparison of the applicant's projected population growth with independent projections made by other agencies such as the Census Bureau, Bureau of Economic Analyses, Envrionmental Protection Agency, and local and state agencies and Councils of Government.

2. Whether population density should be a significant consideration at the construction permit (CP) stage in alternate site evaluation. Present and projected transient populations, appropriately weighted by occupancy, should be included. Computation of the site population factor (SPF) may also be included.

3. Acceptability of the specified low population zone (LPZ). Acceptability of the LPZ with respect to the probability that appropriate protective measures can be taken in behalf of the populace contained therein in the event of a serious accident will be determined by the ISEPB. Dose computations to determine compliance with the LPZ dose guidelines of 10 CFR Part 100 are described in the Standard Review Plans for SAR Section 15.

4. The distance to the nearest boundary of the closest population center (as defined in Part 100), and its relationship to the low population zone outer boundary distance. The boundary shall be determined upon consideration of population distribution, and political boundaries shall not be controlling in this determination.

II. ACCEPTANCE CRITERIA

The data on present population in the region of the site should be based on 1970 census data and are acceptable if so based and if the updated (to the year of application) population numbers check reasonably well against other independently-obtained population data, if available; e.g., General Services Administration (GSA) or Oak Ridge National Laboratory (ORNL) population counts. The projected populations at the approximate year of plant startup and over the expected life of the plant are acceptable if they check reasonably well against independently-obtained population projections, if available; e.g., OBERS,[1/] BEA,[1/] or Water Resource Council.

If, at the CP stage, the population density, including weighted transient population, projected at the time of initial plant operation exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1,000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given by the staff to the consideration of alternative sites in the environmental review.

Transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part-time, or engage in recreational activities, and are not permanent residents of the area.

The specified low population zone is acceptable if (a) ISEPB has determined that appropriate protective measures could be taken in behalf of the enclosed populace in the event of a serious accident; (b) dose computations for the outer boundary of the LPZ, as discussed in the review plans for Section 15, are within Part 100 guidelines; and (c) the nearest boundary of the closest population center (as defined in Part 100) is at least one and one third times the distance from the reactor to the outer boundary of the low population zone.

The population center distance is acceptable if there are no likely concentrations of greater than 25,000 people over the plant lifetime closer than the distance designated by the applicant as the population center distance.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

The reviewer compares the SAR population data, both present (based on 1970 census) and projected, against whatever independent population data is available (e.g., GSA or ORNL population counts, OBERS population projections, U.S. Census Bureau data). Specific

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[1/] OBERS is the descriptive title of a projection program conducted by the U.S. Department of Commerce former Office of Business Economics (OBE), now renamed the Bureau of Economic Analysis (BEA), and the Economic Research Service (ERS) of the U.S. Department of Agriculture.

comparisons should be made of population projections for the approximate year of plant startup and for the expected lifetime of the plant. At the operating license stage, any new population data and projections developed since the construction permit review will be evaluated and compared with previous data. Significant discrepancies will be analyzed to determine the effect on the acceptability of the low population zone and emergency evacuation capabilities. The nearest boundary to the closest population center will be compared with the low population zone outer boundary to ensure that Part 100 guidelines are satisfied. One way of comparing with OBERS projections is as follows:

1. Determine the Bureau of Economic Analysis (BEA) economic areas which lie entirely or partially within a 50 mile radius of the proposed plant. If only a small part of any such area is within the circle, neglect it.

2. Add the 1970 population figures for all BEA areas determined in the first step, and add the BEA projected population for these areas for each of the years for which population projections are to be compared.

3. Find the growth factor for each projected year by taking the ratio of the total projected population in the BEA areas considered to the total 1970 population in those areas.

4. Tabulate, for various radii from the plant, the applicant's 1970 populations; the applicant's projected population; the projected population using the OBERS growth factors derived above; and the ratio of the OBERS projection to the applicant's projection.

5. If the applicant's projections of population growth within 50 miles are significantly less than the projections made by the above method, a more detailed examination of the bases used by the applicant should be made.

The Water Resources subregion projections can be calculated by the same method described for OBERS above. These can be used when the OBERS areas are too large to afford a good comparison.

Population data of specific towns and cities within the low population zone can be checked against population data as contained in the Department of Commerce publication, "1970 Census of Population - Characteristics of the Population."

At the CP stage, the cumulative population density is determined out to a distance of 30 miles using projected populations for the expected year of plant startup, and for the projected lifetime of the plant.[2/] An enclosure on population density is prepared for

[2/] Transient population, appropriately weighted for occupancy, should be included in the population data used if the transient population is unusually large or if the resident population approaches or exceeds 500 people per square mile.

the Environmental Report acceptance review memorandum, noting whether or not the density averaged over any radial distance out to 30 miles exceeds 500 people per square mile for the projected year of plant startup, or 1000 people per square mile over the projected lifetime of the plant. Documentation of this review should be provided to Environmental Projects. The SPF calculations should also be performed at this time.

For cases which just exceed or fall below the above guidelines, an examination of the particular population distribution (as reflected by the computed SPF) may be required. (SPF is a population-weighting concept used in conjunction with population density to compare uniform and nonuniform population distributions. (See Ref. 1.))

Site population is tabulated or plotted against an envelope of previously licensed site populations to determine the relative population characteristics of the proposed site. Curves showing current and projected population as a function of distance may be prepared for use in the staff's safety evaluation report (SER).

The reviewer determines that the current and projected population data for the LPZ includes transients (e.g., workers, occupants of schools, hospitals, etc., recreational facilities).

The reviewer obtains from ISEPB written confirmation (buckslip) of acceptability of the LPZ with respect to their determination that there is reasonable assurance that appropriate protective measures could be taken in behalf of the people within the LPZ in the event of a serious accident.

The reviewer determines that the nearest boundary of the closest population center is at least one and one third times the distance to the outer boundary of the low population zone, considering local groupings of communities and their projected growth rates over the plant lifetime.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff SER:

> "The present and projected populations surrounding the site, including transients, have been reviewed and comparison with independently obtained population data confirms the applicant's estimates.
>
> "On the basis of the specified low population zone and population center distance, and the calculated radiological consequences of design basis accidents at the outer boundary of the low population zone (Section 15), it is concluded that the low population zone and population center distance meet the guidelines of 10 CFR Part 100 and are acceptable."

V. REFERENCES

1. J. Kohler, A. Kenneke, and B. Grimes, "A Technique for Consideration of Population in Site Comparison," presented at the ANS Siting Conference, Portland, Oregon, August 1974.

2. "1972 OBERS Projections," Vol. 1-5, U.S. Water Resources Council, Washington, D.C. (1972).

3. "1970 Census of Population, Characteristics of the Population," Vol. 1, Part A, Sections 1 and 2, Bureau of the Census, U.S. Department of Commerce (1972).

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTIONS 2.2.1 and 2.2.2 LOCATIONS AND ROUTES, DESCRIPTIONS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

1. The locations and distances from the nuclear plant of the various industrial, military, and transportation facilities and routes identified by the applicant as being in the vicinity of the plant are reviewed to identify those activities that may require further investigation and detailed evaluation in order to determine design basis events for the plant. Where available, sources of data independent of the applicant's safety analysis report (SAR) will be used.

2. The descriptive information and statistical data submitted to describe the facilities and the products and materials regularly manufactured, stored, used or transported in the vicinity of the nuclear plant are reviewed to identify potentially hazardous facilities and materials and to establish the maximum quantities of hazardous materials that should be considered in subsequent analyses.

3. Available statistical data pertaining to the nearby transportation routes such as made of transportation, frequency of shipment, frequency of accidents, and the maximum quantities of hazardous materials per shipment are reviewed to establish that sufficient information is available to perform a probability analysis, if required, to determine design basis events.

4. The descriptions of certain significant facilities in the vicinity of the plant, such as airports, waterways, pipelines, or installations which, because of their proximity and the presence of hazardous materials, pose a potential threat to the safety-related features of the plant are reviewed to determine which of these facilities and associated activities may be candidates for design basis events. (A design basis event is a postulated occurrence against which the design of plant safety-related features are evaluated to assure that the postulated event will have no adverse effects.)

II. ACCEPTANCE CRITERIA

1. Data in the SAR is acceptable if it adequately describes the locations and distances of industrial, military, and transportation facilities in the vicinity of the plant, and is in agreement with data obtained from other sources, when available.

2. Descriptions of the nature and extent of activities conducted at nearby facilities, including the products and materials likely to be processed, stored, used, or transported, are acceptable if they are adequate to permit evaluations of possible hazards.

3. Where potentially hazardous materials may be processed, stored, used, or transported in the vicinity of the plant, sufficient statistical data on such materials should be provided to establish a basis for evaluating the potential hazard to the plant.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

1. The reviewer should be especially alert, in the construction permit (CP) stage review, for any potentially hazardous activities in close proximity (up to 2 miles) of the plant. All identified facilities and activities within five miles of the plant should be reviewed. Facilities and activities at greater distances should be considered if they are unusually large (e.g., a large liquid natural gas (LNG) facility or airport) or otherwise have the potential for affecting plant safety-related features. At the operating license (OL) stage, most hazards will already have been identified. Emphasis should be placed on any new information. At the operating license stage, any analyses pertaining to potential accidents involving hazardous materials or activities in the vicinity of the plant will be reviewed to ensure that results are appropriate in light of any new data or experience which is then available.

2. Information should be obtained by the reviewer from sources other than the SAR wherever available, and should be used to check the accuracy and completeness of the information submitted in the SAR. This independent information may be obtained from sources such as U. S. Geological Survey (USGS) maps and aerial photos, published documents, contacts with state and federal agencies, and from other nuclear plant applications (especially if they are located in the same general area or on the same waterway). Information may also be obtained during the site visit and subsequent discussions with local officials. (See Appendix A to Standard Review Plan 2.1.1 for further guidance with regard to site visits.)

3. The specific information relating to types of potentially hazardous material, including distance, quantity, and frequency of shipment, is reviewed to eliminate as many of the

the potential accident situations as possible by inspection, based on past review experience. At the operating license stage, nearby industrial, military and transportation facilities and transportation routes will be reviewed for any changes or additions which may affect the safe operation of the plant. If these changes alter the data or assumptions used in previous hazards evaluations or demonstrate the need for new ones, appropriate evaluations will be performed.

The maximum quantities of explosives likely to be processed, stored, used, or transported in the vicinity of the plant are reviewed to determine if an explosion of this material is capable of producing blast overpressures on the order of 1.0 psi or greater at the plant. References on quantity-distance relationships, e.g., U.S. Army Technical Manual TM5-1300 and Regulatory Guide 1.91, should be consulted.

Regulatory Guides 1.78 and 1.95 are consulted to determine if a potentially hazardous situation exists with regard to chemical releases.

The problems of pipeline rupture and other flammable gas releases are reviewed on an individual case basis by evaluating analyses provided by the applicant, and may also involve independently checking the gas cloud size and TNT equivalency derived by the applicant.

The distance from nearby railroad lines is checked to determine if the plant is within the range of a "rocketing" tank car which, from the National Transportation Safety Board report on the Laurel, Mississippi train accident, dated October 6, 1969, is taken to be 1100 feet, with the range for smaller pieces extending to 1600 feet.

4. The potential accidents which cannot be eliminated from consideration as design basis events because the consequences of the accidents, if they should occur, could be serious enough to affect plant safety-related features, are identified. The Branch Chief is consulted to determine if further detailed investigations by the AAB staff are warranted or if the applicant should be requested to provide additional information.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be used in the staff's safety evaluation report:

"The nature and extent of activities involving potentially hazardous materials which are conducted at nearby industrial, military, and transportation facilities have been evaluated to determine if such activities have the potential for adversely affecting plant safety-related structures. Based on evaluation of information contained in the SAR, as well as information independently obtained by the staff, it is concluded that such activities are not likely to have an adverse effect on the plant safety-related structures."

If the activities are identified as being potentially hazardous, the evaluations described in Standard Review Plan 2.2.3 are performed and conclusions are drawn with respect to the inherent capability of the plant or special plant design measures to prevent radiological releases in excess of the 10 CFR Part 100 guidelines.

V. REFERENCES

1. Department of the Army Technical Manual TM5-1300, "Structures to Resist the Effects of Accidental Explosions," June 1969.

2. Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites."

3. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

4. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

5. National Transportation Safety Board Railroad Accident Report, "Southern Railway Company, Train 154, Derailment with Fire and Explosion, Laurel, Mississippi, January 25, 1969," October 6, 1969.

6. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

The applicant's determination of potential accident situations in the vicinity of the plant that are to be considered as design basis events is reviewed. (See Standard Review Plans 2.2.1 and 2.2.2.)

The applicant's probability analyses of statistical data pertaining to potential accidents involving hazardous materials or activities in the vicinity of the plant, if such analyses have been performed, are reviewed to determine that appropriate data and analytical models have been utilized.

The analyses of the consequences of accidents involving nearby industrial, military, and transportation facilities which have been identified as design basis events are reviewed.

II. ACCEPTANCE CRITERIA

The identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant is acceptable if the design basis events include each postulated type of accident for which a realistic estimate of the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines exceeds the NRC staff objective of approximately 10^{-7} per year. The methods of calculating the radiological exposures resulting from these events are acceptable if they are consistent with methods used for calculation of other accident radiological exposures (e.g., SRP 15.6.5.) Because of the difficulty of assigning precise numerical values to the probability of occurrence of the types of potential hazards generally considered in this review plan, judgment must be used as to the acceptability of the overall risk presented by an event.

In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to potential consequences in excess of 10 CFR Part 100 exposure guidelines should be estimated using assumptions that are as realistic as is practicable. In addition, because of the low probability events under consideration, valid statistical data are often not available to permit accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The effects of design basis events have been appropriately considered if analyses of the effects of those accidents on the safety-related features of the plant have been performed and appropriate measures (e.g., hardening, fire protection) to mitigate the consequences of such events have been taken.

III. REVIEW PROCEDURES

The judgment on the areas to be given attention and analysis during the review is based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved. The selection and emphasis of the areas covered by the review is made by the reviewer on each case.

In some cases it may be necessary to consult with or obtain specific data from other branches, such as the Structural Engineering Branch (SEB) or Auxiliary and Power Conversion Systems Branch (APCSB), regarding possible effects of external events on plant structures or components.

The applicant's probability calculations are reviewed, and an independent probability analysis is performed by the staff if the potential hazard is considered significant enough to affect the licensability of the site or is important to the identification of design basis events.

The design parameters (e.g., overpressure) and physical phenomena (e.g., gas concentration) selected by the applicant for each design basis event are reviewed to ascertain that the values are comparable to the values used in previous analyses and found to be acceptable by the staff.

Each design basis event is reviewed to determine that the effects of the event on the safety features of the plant have been evaluated. If an accidental explosion is considered to be a design basis event, an analysis of the missiles generated in the explosion should be analyzed under the procedures given in the standard review plans (SRP) for Section 3.5.

If accidents involving release of smoke, flammable or nonflammable gases, or chemical bearing clouds are considered to be design basis events, an evaluation of the effects of these accidents on control room habitability should be made in SAR Section 6.4 and on the operation of diesels and other safety-related equipment in SAR Chapter 9.

IV. EVALUATION FINDINGS

If the reviewer verifies that sufficient information has been provided and that the evaluation is sufficiently complete and adequate to meet the acceptance criteria in Section II of this SRP, conclusions of the following type may be prepared for the staff's safety evaluation report:

> "The applicant has identified potential accidents in the vicinity of the plant which should be considered as design basis events and has provided analyses of the effects of these accidents on the safety-related features of the plant. The applicant has demonstrated that the plant is adequately protected and can be operated with an acceptable degree of safety with regard to potential accidents which may occur as the result of activities at nearby industrial, military, and transportation facilities."

V. REFERENCES

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.3.1 REGIONAL CLIMATOLOGY

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning averages and extremes of climatic conditions and regional meteorological phenomena which affect the safe design and siting of the plant. The review covers the following specific areas:

1. A description of the general climate of the region with respect to types of air masses, synoptic features (high and low pressure systems and frontal systems), general air-flow patterns (wind direction and speed), temperature and humidity, precipitation (rain, snow, and sleet), and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions.

2. Seasonal and annual frequencies of severe weather phenomena including hurricanes, tornadoes, waterspouts, thunderstorms, lightning, hail (including probable maximum size), freezing rain, dust (sand) storms, and high air pollution potential.

3. Meteorological conditions used as design and operating bases including:

   a. The maximum snow and ice load (water equivalent) that the roofs of safety-related structures are capable of withstanding during plant operation.

   b. Ultimate heat sink meteorological conditions resulting in maximum evaporation and drift loss of water and minimum water cooling.

   c. Tornado parameters including translational speed, rotational speed, and maximum pressure differential with the associated time interval.

   d. 100-year return period "fastest mile of wind" including vertical velocity distribution and gust factor.

   e. Probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and, where applicable, dust (sand) storms.

f. Other meteorological and air quality conditions used for design and operating basis considerations.

II. ACCEPTANCE CRITERIA

The information in this section will be acceptable if the regional meteorological conditions and phenomena which affect the safe design and siting of the plant are presented and substantiated in accordance with acceptable practice and data as promulgated by the National Oceanic and Atmospheric Administration (NOAA), industry standards, and Commission guides, criteria, and regulations. More specifically the following criteria are used to determine acceptability.

The description of the general climate of the region should be based on standard climatic summaries compiled by NOAA. Consideration of the relationships between regional synoptic-scale atmospheric processes and local (site) meteorological conditions should be based on appropriate meteorological data.

Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military or other stations recognized as standard installations which have long periods of record. The applicability of these data to represent site conditions during the expected period of reactor operation must be substantiated using sound meteorological judgment and data.

Design basis tornado parameters should be based on Regulatory Guide 1.76 (Ref. 2) or an adequately substantiated study must have been performed to demonstrate that lower values apply to the specific site. Operating basis wind velocity (fastest mile of wind) should be based on a standard such as that published by the American National Standards Institute (ANSI) with suitable corrections for local conditions. The ultimate heat sink meteorological data, as stated in Regulatory Guide 1.27 (Ref. 1) should be based on long-period regional records which represent site conditions. Freezing rain estimates are to be based on representative NWS station data. All other meteorological and air quality data used for safety-related plant design and operating bases should be documented and substantiated.

High air pollution potential information should be based on U.S. Environmental Protection Agency (EPA) studies.

III. REVIEW PROCEDURES

1. General Climate

The general climatic description of the region in which the site is located is reviewed for completeness and authenticity. Climatic parameters such as air masses, general airflow, pressure patterns, frontal systems, and temperature and humidity conditions reported by the applicant are checked against standard references (Ref. 3 and 4) for appropriateness with respect to location and period of record.

The applicant's description of the role of synoptic-scale atomospheric processes on local (site) meteorological conditions is checked against the descriptions provided in References 4 and 5 and the reviewer's knowledge of the area.

2. Regional Meteorological Averages and Extremes

Since meterological averages and extremes can only be obtained from stations in the region of the site which have long periods of record, and the stations are not usually very close to the site, a determination of the representativeness of the data to site conditions is the primary concern in the review. A determination of the adequacy of the stations and their data is also made.

Recorded meterological averages and extremes are checked against standard publications such as Reference 6. Snow and ice load adequacy is confirmed using ANSI A58.1-1972 (Ref. 7) and regional data available in References 4,5, and 6. References 4 and 5 provide information on other averages and extremes. References 8 and 9 provide information on high air pollution potential for verification. Extreme winds and specific vertical velocity distribution are checked against References 7 and 11. Gust factors are checked against Reference 7. The design basis tornado parameters are checked for agreement with Regulatory Guide 1.76 (Ref. 2) and tornado data are verified using the procedures and data in WASH-1300 (Ref.10).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports concluding statements of the following type to be included in the staff's safety evaluation report:

"The applicant has provided an adequate description of the regional meteorological conditions of importance to the safe design and siting of this plant."

This statement will be followed by a resume of the general climate and the meteorological design parameters used for the plant.

V. REFERENCES

1. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 1.

2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

3. U.S. Department of Commerce,"Climatic Atlas of the United States," Environmental Data Service, NOAA, June 1968.

4. U.S. Department of Commerce, "Local Climatological Data and Comparative Data," Environmental Data Service, NOAA, published annually for all first-order NWS Stations.

5. U.S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, published annually by state.

6. U.S. Department of Commerce, "Storm Data," Environmental Data Service, NOAA, published monthly.

7. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and other Structures," American National Standards Institute (1972).

8. G. C. Holzworth, "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, USEPA, January 1972.

9. J. Korshover, "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1965," Publication No. 99-AP-34, Public Health Service (1967).

10. E.H. Markee Jr., "Technical Basis for Interim Regional Tornado Criteria," WASH-1300, USAEC, May 1974.

11. H.C.S. Thom, "New Distribution of Extreme Winds in the United States," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, pp. 1787-1801, July 1968.

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SECTION 2.3.2 LOCAL METEOROLOGY

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning the local (site) meteorological parameters, an assessment of the potential influence of the plant and its facilities on local meteorological conditions, and a topographical description of the site and its environs. The review covers the following specific areas.

1. A description of the local (site) meteorology in terms of airflow, temperature, atmospheric water vapor, precipitation, fog, and atmospheric stability.

2. An assessment of the influence of the plant and its facilities on the local meteorological parameters listed in (1), including the effects of plant structures, terrain modification, and heat and moisture sources due to plant operation.

3. A topographical description of the site and its environs, as modified by the plant structures, including the site boundary, exclusion zone, and low population zone.

II. ACCEPTANCE CRITERIA

The information in this section will be acceptable if the local meteorological and topographic descriptions of the site area applicable both before plant construction and during plant operation are adequately documented such that meteorological impacts on plant design and operation as well as the impact of the plant on local meteorological conditions can be reliably predicted. Specifically, the following information is needed for acceptance. This information should be fully documented and substantiated as to its representativeness of conditions at and near the site.

1. Local summaries of meteorological data based on onsite data and National Weather Service station summaries or other standard installation summaries from appropriate nearby locations should be presented.

The following summaries are required:

a. Monthly and annual wind roses and tabular data, and wind direction persistence summaries.

b. Monthly and annual air temperature summaries including averages, measured extremes, and diurnal variations.

c. Monthly and annual summaries of atmospheric water vapor (absolute and relative) including averages, measured extremes, and diurnal variations.

d. Monthly and annual summaries of precipitation (rain and snow) including averages, measured extremes, and intensity-duration data.

e. Monthly and annual summaries of fog (and smog) including expected values and extremes of frequency and duration.

f. Monthly and annual summaries of atmospheric stability (ΔT), including frequency and duration (persistence) of inversion conditions if data are available.

2. A discussion and evaluation of the influence of the plant and its facilities on the local meteorological conditions are required. Potential changes in normal and extreme values presented in SAR Section 2.3.2.1 resulting from plant construction and operation should be made.

3. A complete topographical description of the site and environs out to a distance of 50 miles from the plant, as described in Standard Format Section 2.3.2.3 should be provided (Ref. 1).

III. REVIEW PROCEDURES

1. The summaries listed in Section 2.3.2.1 of the Standard Format are reviewed for completeness and adequacy of basic data. The wind and atmospheric stability data should be based on onsite data if possible since airflow and vertical temperature structure can vary substantially from one location to another and are inputs to the assessment of atmospheric diffusion conditions at the site. The other summaries should be based on nearby representative stations with long periods of record since the locally measured extremes in intensity and frequency are compared to design basis values presented in Section 2.3.1 of the safety analysis report or are used by other branches to determine whether these meteorological conditions are limiting con-itions for design and emergency procedures. When offsite data are used, a determination is made of how well the data represent site conditions and whether more representative data are available. National Oceanic and Atmospheric Administration (NOAA) state meteorological summaries (Ref. 2), local climatological data (Ref. 3), and various NOAA Environmental Data Service summaries are used by the reviewer to evaluate the representativeness of stations and periods of record. The reviewer visits all primary meteorological data collection locations.

2. The review procedure for evaluating the contents of Section 2.3.2.2 of the SAR is as follows:

   a. Determine the terrain modifications that will occur as a result of plant construction such as removal of trees, leveling of ground, and installation of lakes and ponds.

   b. Determine the location, size, and materials used for plant structures including buildings, switchyard gear, parking lots, and roads.

   c. Determine and quantify the heat and moisture sources that will result from plant operations.

   d. Relate the input information in items a,b, and c, above, to local meteorological modifications.

   e. Compare the reviewer's assessment with that of the applicant.

3. Section 2.3.2.3 is reviewed for completeness in accordance with the specifications of the Standard Format. The reviewer assures that all topographic maps and topographic cross-sections presented by the applicant are legible and well-labeled so that the information needed during the review can be readily extracted. Reference points and the direction of true north should be checked carefully. Points of interest such as plant structures, site boundary, and exclusion zone should be marked on the maps and diagrams.

   The reviewer compares the applicant's assessment of the effect of topography to standard assessments such as those presented in "Meteorology and Atomic Energy - 1968" (Ref. 4) and decides whether the standard regulatory atmospheric diffusion models (discussed in Standard Review Plans 2.3.4 and 2.3.5) are appropriate for this site.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports concluding statements of the following type, to be included in the staff's safety evaluation report:

> "The applicant has provided adequate information on local meteorological and air quality conditions that are of importance to the safe design and siting of this plant."

This statement will be preceded by a resume of local meteorological and air quality parameters appropriate to the site.

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

2. U.S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, published annually by state.

3. U.S. Department of Commerce, "Local Climatological Data and Comparative Data," Environmental Data Service, NOAA, published annually for all first-order NWS Stations.

4. D.H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

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SECTION 2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAMS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning the onsite meteorological measurement programs including instrumentation, data summaries, and, at the operating license (OL) stage, provisions of the technical specifications. The review covers the following specific areas:

1. The meteorological instrumentation review includes siting of sensors, sensor performance specifications, methods and equipment for recording sensor output, the quality assurance program for sensors and recorders, and data acquisition and reduction procedures.

2. The review of meteorological data summaries includes consideration of the period of record and amenability of the data for use in making atmospheric diffusion estimates.

3. The review of meteorological technical specifications includes consideration of instrument siting, instrument specifications, control room monitoring, and data reporting and storage.

II. ACCEPTANCE CRITERIA

1. Generally the onsite meterological programs must produce data which can be summarized to provide an adequate meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents. Guidance on an adequate program is given in Regulatory Guide 1.23. More specifically:

   a. The siting of meteorological sensors should satisfy the intent and recommendations of Regulatory Guide 1.23 or state-of-the-art procedures.

b. The meteorological sensors should meet the sensitivity recommendations of Regulatory Guide 1.23 and be capable of withstanding the expected range of environmental conditions at the site such that adequate data recovery is anticipated. Any deviation from Regulatory Guide 1.23 must be justified.

c. The meteorological recording systems must be capable of providing accurate, reliable data.

d. The instrument surveillance and calibration procedures must provide reasonable assurance that adequate, accurate data will be obtained.

e. The data acquisition and reduction procedures should provide average data which are within the accuracy guidelines of Regulatory Guide 1.23. Any deviation must be justified.

2. The following criteria are used to judge the acceptability of meteorological data summaries for atmospheric diffusion estimates.

a. For the preliminary safety analysis report (PSAR), if adequate onsite meteorological data are not available at docketing, the best available (onsite and offsite) meteorological data to describe the atmospheric diffusion characteristics of the site in the form of joint frequency distributions of wind direction and wind speed by atmospheric stability class must be presented. Evidence of how well these data represent long-term conditions at the site must be presented. Adequate onsite meteorological data must be provided prior to or with the scheduled response to the first set of requests for additional information in the PSAR review.

For site suitability reviews, at least six months of onsite meteorological data with evidence of how well these data represent long-term conditions at the site must be presented (See Regulatory Guide 4.2.1).

b. For the final safety analysis report (FSAR), at docketing, or for the PSAR if adequate onsite meteorological data have been collected, one year (and, preferably, two or more whole years) of onsite meteorological data must be provided in the form of joint frequency distributions of wind direction and wind speed by atmospheric stability class. Evidence of how well these data represent long-term conditions of the site must also be presented.

Regulatory Guide 1.23 provides guidance on an acceptable format for meteorological data summaries and adequacy of data.

III. REVIEW PROCEDURES

1. Meteorological Instrumentation

The basic meteorological parameters measured by instrumentation at all sites should include wind direction and wind speed at two levels, ambient air temperature difference between two levels, temperature, and atmospheric moisture (at sites where water vapor is emitted, as from cooling towers or spray ponds).

a. Instrument Siting

Instrument types, heights, and locations are compared generally to the recommendations of Regulatory Guide 1.23, Sections C.1 and C.2. Detailed review procedures follow.

(1) Local Exposure of Instruments

The local exposure of the wind and temperature sensors is reviewed to assure that the measurements will represent the general site area. A determination is made whether the tower which supports the sensors will influence the wind or temperature measurements. Professional experience and studies have shown that wind sensors should be mounted on booms such that the sensors are at least one tower width away from an open-latticed tower and at least two stack or tower widths away from a stack or closed tower. For temperature sensors, mounting booms need not be as long as those for wind sensors but must be unaffected by thermal radiation from the tower itself. No temperature sensors may be mounted directly on stacks or closed towers. Mounting booms for all sensors should be oriented normal to the prevailing wind at the site.

A determination is made whether the terrain at or near the base of the tower will unnaturally affect the wind or temperature measurements. Heat reflection characteristics of the surface underlying the meteorological tower (grass, soil, gravel, paving, etc.) are estimated to assure that localized influences on measurements are minimal. The position, size, and materials used in the construction of the recorder shack and nearby trees are also examined for potential localized influence on the measurements.

(2) General Exposure of Instruments

Since the objective of the instrumentation is to provide measurements which represent the overall site meteorology without plant structure interference, the tower position(s) must have been selected with this general objective in mind. Examination of topographical maps, which have been modified to show finished plant grade, and a site visit along with professional judgement on airflow patterns are used to determine and evaluate the representativeness of the location(s).

The plant structure layout including structure heights are examined for potential influence on meteorological measurements. In general, sensors should be located at least five building heights away from the buildings to minimize this influence.

b. Meteorological Sensors

The type and performance specifications of the sensors are evaluated. Manufacturers' specifications and analysis, and operating experience for these sensors are considered in evaluation of adequacy with respect to accuracy and the potential for acceptable data recovery. Standardized evaluations such as Reference 5 and operational experience reports contained in research papers are utilized.

The suitability of the specific type of sensor for use in the environmental conditions at the site is evaluated. To this end, the range of wind conditions and the ability of the sensors to withstand corrosion, blowing sand, salt, air pollutants, birds, and insects are considered.

If the sensors are new and unique, a meteorological instrumentation expert (e.g., NOAA, Idaho Falls) is consulted.

c. Recording of Meteorological Sensor Output

The methods of recording (e.g., digital or analog, instantaneous or average, engineering units or raw voltages) and the recording equipment including performance specifications and location of this equipment are evaluated. Manufacturers' specifications and operating experience for the recorders are considered in evaluation of adequacy with respect to accuracy and the potential for acceptable data recovery.

The controlled environmental conditions in which the recorders are kept (instrument shack or control room) are reviewed for adequacy in accordance with the manufacturers' specifications. The ability to obtain a direct readout from the recorders in situ during routine inspection of systems is checked so that the inspector will be able to relate the recorder output directly to what the sensor should be seeing. Some specific recommendations are contained in Regulatory Guide 1.23, Section C.3.

The reviewer determines that there are provisions for proper monitoring of wind direction, wind speed, and vertical temperature difference in the control room during plant operation.

d, Instrumentation Surveillance

The inspection, maintenance, and calibration procedures and their frequency are evaluated. These surveillance procedures and the frequency of attention that the instrumentation systems receive are compared to operating experience at this site and other sites with similar instrumentation with the objective of determining that acceptable data recovery with acceptable accuracy will be obtained throughout the duration of the meteorological program. Guidelines for acceptable accuracy and acceptable data recovery are specified in Regulatory Guide 1.23, Sections C.4 and 5. Any deviations from Regulatory Guide 1.23 must be justified.

e. Data Acquisition and Reduction

The procedures, including both hardware and software, for data acquisition and reduction are evaluated. Since there are many methods of acquiring data from meteorological measurement systems which are acceptable to the staff, the review procedure varies. The basic components of the program which are reviewed to ascertain the acceptability of data acquisition and reduction are:

(1) Accuracy of measuring in units of direct measurement and their precision.
(2) Accuracy in conversion of direct measurement units to meteorological units.
(3) Accuracies involved in frequency and mode (instantaneous or average) of sampling.
(4) Time over which system outputs are averaged for final disposition and accuracy of these data.

Since the instrument accuracy suggestions in Regulatory Guide 1.23 refer to overall system accuracy for instantaneous recorded values or time averaged values, the overall system accuracy is evaluated in addition to the component (sensor, recorder, and reduction) accuracies. The evaluation consists primarily of using statistical procedures for compound errors based on sensor accuracy, recorder accuracy, conversion of units accuracy, and frequency and mode of sampling (Ref.6).

2. Meteorological Data Summaries

Annual (representing the annual cycle) joint frequency distributions of wind direction and wind speed by atmospheric stability class are evaluated from the viewpoint of sufficiency of detail to permit the staff to make an independent determination of the atmospheric diffusion conditions and relative concentrations for accidental and routine atmospheric releases of radioactive effluents from the reactor and its facilities. The distributions are to be based wholly on onsite data, a combination of onsite and offsite data, or offsite data in accordance with the criteria of sections II.2.a and b of this plan. The joint frequency distributions are compared to the example distribution given in Regulatory Guide 1.23.

"Calm" wind conditions (which should be defined as wind speeds less than the starting speed of the anemometer or vane, whichever is higher) are checked for appropriateness and appearance in the distributions as a separate speed class, without directional assignment, by atmospheric stability class.

Annual joint frequency distributions for each expected mode of release (i.e., ground level and elevated) are checked for appropriateness of heights of measurements of wind direction, wind speed, and atmospheric stability. Winds at the 10-meter level and temperature difference (ΔT) between the vent height and the 10-meter level are used for vent and penetration releases. Winds from near release height and ΔT between release height and the 10-meter level are used for stack releases. A stack is defined as a release point which is greater than twice the height of adjacent structures.

The climatic representativeness of the joint frequency distribution is checked by comparison with nearby stations which have collected reliable meteorological data over a long period of time (10-20 years). The distributions are compared with sites in similar geographical and topographical locations to assure that the data are reasonable.

3. Meteorological Technical Specifications

The applicant's technical specifications are reviewed at the OL stage to determine if the operational meteorological monitoring program meets the recommendations of Regulatory Guide 1.23 with respect to tower siting, instrumentation specifications, and control room monitoring, and if the reporting requirements meet the recommendations of Regulatory Guide 1.21, Rev. 1. Deviations from the Regulatory Guides may be accepted if justified.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan and that his evaluation supports the following type of concluding statement, to be used in the staff's safety evaluation report:

"The onsite meteorological measurements program has been compared with the recommendations and intent of Regulatory Guide 1.23. The staff concludes that the meteorological measurements program (is expected to produce/has produced) data which, in turn (can be summarized/have been summarized) to provide an adequate meteorological description of the site and its vicinity for the purpose of making atmospheric diffusion estimates for accidental and routine airborne releases of effluents from the nuclear facility."

For the CP review, if adequate meteorological data have not been acquired by the applicant and presented to the staff, a statement requiring the applicant to obtain adequate data in a timely manner will be added.

The input to the safety evaluation report will also include a brief summary description of the onsite meteorological measurements program covering the following items:

1. Height and location of meteorological sensors by type.
2. Period of data record.
3. Data recovery.
4. Period of data record and meteorological parameters used for atmospheric diffusion estimates.

V. REFERENCES

1. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

2. Regulatory Guide 1.23, "Onsite Meteorological Programs."

3. Regulatory Guide 4.2.1, "Additional Guidance - Environmental Data."

4. R. C. Hilfiker, "Exposure of Instruments," Chapter in Air Pollution Meteorology Manual, Training Course 411 conducted by USEPA Air Pollution Training Institute, Research Triangle Park, North Carolina, August 1973.

5. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

6. C. E. P. Brooks and N. Caruthers, "Handbook of Statistical Methods in Meteorology," M.O. 538, Her Majesty's Stationary Office, London (1953).

7. D. A. Mazzarella, "An Inventory of Specifications for Wind Measuring Instruments," Bull. Amer. Meteor. Soc. 53, 860 (1972).

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SECTION 2.3.4 SHORT TERM DIFFUSION ESTIMATES

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning atmospheric diffusion estimates for accidental releases of effluents to the atmosphere. The review covers the following specific areas:

1. Atmospheric diffusion models to calculate relative concentrations for accidental radioactive and toxic gas release modes as determined by Accident Analysis Branch.

2. Meteorological data summaries used as input to diffusion models.

3. Derivation of diffusion parameters from meteorological data.

4. Probability distributions of relative concentrations.

5. Relative concentrations used for assessment of consequences of radioactive releases for design basis (10 CFR Part 100) accidents, onsite and offsite toxic gas releases, and accidents that result in limited releases of radioactivity.

II. ACCEPTANCE CRITERIA

This section will be acceptable if the applicant has provided conservative estimates of atmospheric diffusion at appropriate distances from the source for postulated accidental releases of radioactive and toxic materials to the atmosphere considering the plant as both a source and a receptor. Guidelines for acceptability of models and conservatism appropriate to design basis calculations are Regulatory Guides 1.3, 1.4, 1.5, 1.23, 1.24, 1.25, and 1.77; National Oceanic and Atmospheric Administration (NOAA) Technical Memorandum ERL ARL-42; standard references such as "Meteorology and Atomic Energy - 1968" and Accident Analysis Branch and Site Analysis Branch positions. Since the staff makes an independent evaluation of atmospheric diffusion estimates based on data from the onsite meteorological measurements program and other nearby meteorological data, it is not necessary for the applicant to duplicate the staff's final estimates. However, the applicant's diffusion estimates should reasonably reflect staff positions and state-of-the-art atmospheric diffusion knowledge. Specifically the following information is required:

1. The atmospheric diffusion models used by the applicant to calculate relative concentrations resulting from accidental airborne releases of radioactive and toxic gases must be documented in detail and substantiated so that the staff can evaluate their appropriateness to site and plant characteristics.

2. Meteorological data summaries to be used as input to the diffusion models must be presented in joint frequency distribution form. These summaries must have been generated from the best available annual periods of data on record and contain data acceptable to the staff which represent appropriate hourly values of wind direction, wind speed, and atmospheric stability for each mode of accidental release.

3. The atmospheric fiffusion parameters, such as lateral and vertical plume spread (σ_y and σ_z) as a function of distance and windspeed, must be related to measured meteorological parameters and substantiated as to their validity and degree of conservatism for use in estimating the consequences of accidents within the range of distances which are of interest for the plant.

4. Cumulative frequency distributions of relative concentrations (X/Q) based on mode of release over appropriate time periods and on the aforementioned atmospheric diffusion models, meteorological data summaries, and atmospheric diffusion parameters must be presented for appropriate distances such as the site boundary distance and the outer boundary of the low population zone as specified in Section 2.3.4.2 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The methods of generating these distributions must be adequately described.

5. Relative concentrations used for assessment of consequences of radioactive releases for design basis (10 CFR Part 100) accidents, for onsite and offsite releases of toxic gases, and for radioactive releases resulting from other accidents must be presented.

III. REVIEW PROCEDURES

1. Atmospheric Diffusion Models

The applicant's diffusion models are compared to the general Gaussian models which are contained in Regulatory Guides 1.3 and 1.4 for elevated releases and ground level releases with a wake correction (see also Reference 3). The suitability of the models for mode of release, plant configuration, and site topography are reviewed. Accident Analysis Branch defines the modes of release and accidents to be considered.

A determination is made as to whether the release should be considered as an elevated point source or a ground level point source with a volumetric correction for turbulent mixing in the wake of buildings. Generally the release is considered to be elevated if the release point is at least twice as high as nearby solid structures. Otherwise, a ground level volumetric release formulation is usually used. The volumetric correction is usually based on 1/2 the minimum cross-sectional area of the structure from which the effluent is released.

Most accidental releases are considered as continuous releases (i.e., >5 minutes duration). However, in some instances, usually with explosions resulting in the release of toxic chemicals, the releases may be considered as instantaneous (puffs). For puff releases, instantaneous point source Gaussian diffusion equations are used with a correction for initial source volume (Ref. 10).

If a site is located such that the horizontal (or vertical) plume spread via diffusion is restricted by topography (or unusual meteorological conditions), the models are examined for appropriate modification. Some of these conditions are narrow, deep valleys, "fumigation" from elevated sources, and low level subsidence inversions of temperature in the vertical direction.

2. Meteorological Data Summaries

The data summaries in joint frequency distribution form are reviewed for compatability of data with the models utilized in the section above. General criteria are stated in Regulatory Guide 1.23 and in III.2 of Standard Review Plan 2.3.3.

3. Atmospheric Diffusion Parameters

Horizontal and vertical plume spread parameters, σ_y and σ_z, as functions of distance and atmospheric stability are reviewed. The current procedure is to relate $\sigma_y(X)$ and $\sigma_z(X)$ to vertical temperature difference classes as stated in Table 1 of Regulatory Guide 1.23. Departures from this procedure are reviewed for adequate justification. Such departures may be appropriate in the case of unusual sites (e.g., valley or coastal). The curves of σ_y and σ_z with distance, which appear in "Meteorology and Atomic Energy - 1968" are usually acceptable with the addition of a G stability class.

In instances when a puff diffusion equation is used, $\sigma_x = \sigma_y$ is usually a good assumption.

4. Cumulative Frequency Distributions of X/Q

A check is made of the cumulative frequency distributions for inclusion of pertinent modes and time periods of release, and adequacy of input data in accordance with the guidelines set forth in Section 2.3.4.2 of the Standard Format. The methods used to generate these distributions are reviewed for adequacy and conservatism.

5. Relative Concentrations Used for Accidents

The X/Q values used for assessment of consequences of radioactive releases for design basis accidents, for onsite and offsite releases of toxic gases, and for radioactive releases resulting from other accidents are reviewed for appropriateness and completeness of information.

An independent calculation of the probability distributions of X/Q is made for pertinent distances (usually the exclusion area boundary and the low population zone outer boundary, LPZ) using the computer program CHI/Q (Ref. 11) and the joint frequency distribution data for input. The most restrictive annual average X/Q values are also computed at the site boundary and the LPZ using the same program and input data. For assessment of the consequences of design basis accident releases, the value of X/Q at the "5% level" (value which is exceeded 5% of the time) is evaluated at the exclusion boundary and the LPZ. These values are assumed to represent conditions for a two-hour period. X/Q values for time periods greater than two hours are estimated for the LPZ distance by assuming a logarithmic relationship between the "two-hour" value and the annual average value.

Conservative (5%) values of X/Q from appropriate models for appropriate time intervals and distances are transmitted to AAB for dose assessment of design basis accidents.

For assessment of other accidents, the median (50%) values of X/Q for appropriate time intervals and distances (usually the site boundary) are evaluated and transmitted to AAB.

X/Q values based on site-specific meteorological data are calculated, as needed, for control room dose calculations and onsite and offsite releases of toxic gases. These estimates are made on a case-by-case basis since the mode of release and, therefore, the diffusion models vary.

IV. EVALUATION FINDINGS

The reviewer verifies that adequately conservative atmospheric diffusion models, with adequate onsite meteorological data as input to the models, have been used to calculate relative concentrations at appropriate distances and directions from postulated release points during accidental airborne releases of potentially hazardous materials. If adequate onsite meteorological data are not available for the construction permit review, the reviewer must assure that adequate conservatism has been applied to the calculated relative concentrations for accidental airborne effluent releases based on available data.

The reviewer's evaluation must support the following type of concluding statement, to be used in the staff's safety evaluation report:

"Conservative assessments of post-accident atmospheric diffusion conditions have been made by the staff from the applicant's meteorological data and appropriate diffusion models."

The input to the safety evaluation report will also include a brief summary of the relative concentrations (X/Q) calculated by the staff, reference to diffusion models used, and a comparison between the values computed by the staff and the applicant.

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

3. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

4. Regulatory Guide 1.23, "Onsite Meteorological Programs."

5. Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure."

6. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

7. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

8. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

9. D. H. Slade (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

10. G. R. Yanskey, E. H. Markee, and A. P. Richter, "Climatography of the National Reactor Testing Station," IDO-12048, Idaho Operations Office, USAEC (1966).

11. J. F. Sagendorf, "A Program for Evaluating Atmospheric Dispersion From A Nuclear Power Station," Technical Memorandum ERL ARL-42, National Oceanic and Atmospheric Administration (1974).

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SECTION 2.3.5 LONG TERM DIFFUSION ESTIMATES

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Information is presented by the applicant and reviewed by the staff concerning atmospheric diffusion estimates for routine releases of effluents to the atmosphere. The review covers the following specific areas:

1. Atmospheric diffusion models to calculate relative concentrations for routine radioactive gas release modes as determined by Effluent Treatment Systems Branch.

2. Meteorological data summaries used as input to diffusion models.

3. Derivation of diffusion parameters from meteorological data.

4. Relative concentrations used for assessment of consequences of routine airborne radioactive releases.

II. ACCEPTANCE CRITERIA

This section will be acceptable if the applicant has provided realistic estimates of atmospheric diffusion at appropriate distances from the source for routine releases of radioactive materials to the atmosphere. Guidelines for acceptability of models are Regulatory Guides 1.21, 1.42, and 1.DD (Refs. 1, 3, and 5); National Oceanic and Atmospheric Administration (NOAA) Technical Memorandum ERL ARL-42; standard references such as "Meteorology and Atomic Energy - 196;" and Effluent Treatment Systems Branch and Site Analysis Branch positions. Since the staff makes an independent evaluation of atmospheric diffusion estimates based on data from the onsite meteorological measurements program and other nearby meteorological data, it is not necessary for the applicant to duplicate the staff's final estimates. However, the applicants diffusion estimates should reasonably reflect staff positions and state-of-the-art atmospheric diffusion knowledge. Specifically the following information is required:

1. The atmospheric diffusion models used by the applicant to calculated concentrations resulting from routine airborne releases of radioactive gases must be documented in detail and substantiated so that the staff can evaluate their appropriateness to site and plant characteristics.

2. Meteorological data summaries to be used as input to the diffusion models must be presented in joint frequency distribution form. These summaries must have been generated from the best available annual periods of data on record and contain data acceptable to the staff which represent appropriate hourly values of wind direction, wind speed, and atmospheric stability for each mode of routine release.

3. The atmospheric diffusion parameters, such as vertical plume spread (σ_z) as a function of distance and wind speed, must be related to measured meteorological parameters and be substantiated as to their validity for use in estimating the consequences of routine releases from the site boundary to a radius of 50 miles from the plant.

4. Relative concentrations (X/Q) used for assessment of consequences of routine radioactive gas releases must be presented as described in Section 2.3.5.2 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

III. REVIEW PROCEDURES

1. Atmospheric Diffusion Models

The applicant's diffusion models are compared to the general Gaussian models which are contained in Regulatory Guide 1.DD (Ref.5) for elevated releases and ground level releases with a wake correction (see also Reference 3). The suitability of the models for mode of release, plant configuration, and site topography are reviewed. Effluent Treatment Systems Branch defines the modes of release to be considered.

A determination is made as to whether the release should be considered as an elevated point source or a ground level point source with a volumetric correction for turbulent mixing in the wake of buildings. Generally the release is considered to be elevated if the release point is at least twice as high as nearby solid structures. Otherwise, a ground level volumetric release formulation is usually based on 1/2 the height of the structure from which the effluent is released.

If a site is located such that the effluent trajectories (or vertical plume spread via diffusion) are restricted by topography (or unusual meteorological conditions), the models are examined for appropriate modification. Some of these conditions are narrow, deep valleys, "fumigation" from elevated sources, and low level subsidence inversions of temperature in the vertical direction.

2. Meteorological Data Summaries

The data summaries in joint frequency distribution form are reviewed for compatability of data with the models utilized in the section above. General criteria are stated in Regulatory Guide 1.23 and III.2 of Standard Review Plan 2.3.3.

3. Atmospheric Diffusion Parameters

The vertical plume spread parameter, σ_z, as a function of distance and atmospheric stability is reviewed. The current procedure is to relate σ_z (X) to vertical temperature difference classes as stated in Table 1 of Regulatory Guide 1.23 (Ref. 2). Departures from this procedure are reviewed for adequate reasons for the departures, such as in the case of unusual sites (e.g., valley or coastal). The curves of σ_z with distance, which appear in "Meteorology and Atomic Energy - 1968" are usually acceptable with the addition of a G stability class.

4. Relative Concentrations Used for Routine Releases

The X/Q values used for assessment of the consequences of routine radioactive releases are reviewed for appropriateness to site conditions and completeness of information.

An independent calculation of annual average X/Q values is made for 16 radial sectors from the site boundary to a distance of 50 miles from the plant using appropriate meteorological data in joint frequency distribution form and the computer program CHI/Q (Ref. 7). Adjustments of the X/Q output may be made through use of other offsite meteorological data when unusual topographic conditions surround the site or when the onsite meteorological data are found to be inadequate.

IV. EVALUATION FINDINGS

The reviewer verifies that adequate atmospheric diffusion models, with adequate onsite meteorological data as input to the models, have been used to calculate relative concentrations at appropriate distances and directions from postulated release points during routine airborne releases of radioactive gases. If adequate onsite meteorological data are not available for the construction permit review, the reviewer must assure that adequate conservatism has been applied to the calculated relative concentrations for routine airborne effluent releases based on available data. The reviewer's evaluation must support the following type of concluding statement, to be included in the staff's safety evaluation report:

> "Reasonable estimates of average atmospheric diffusion conditions have been made by the staff from the applicant's meteorological data and appropriate diffusion models."

The input to the safety evaluation report will also include a brief summary of the relative concentrations (X/Q) calculated by the staff, reference to diffusion models used, and a comparison between the values computed by the staff and the applicant.

V. REFERENCES

1. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1.

2. Regulatory Guide 1.23, "Onsite Meteorological Programs."

3. Regulatory Guide 1.42, "Interim Licensing Policy on As Low As Practicable for Gaseous Radioactive Releases from Light-Water-Cooled Nuclear Power Reactors."

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

5. Regulatory Guide 1.DD, "Methods for Estimating Atmospheric Dispersion of Gaseous Effluents from Routine Releases," attachment to Concluding Statement of Postion of the Regulatory Staff, Docket No. RM-50-2, February 20, 1974.

6. D. H. Slade, (ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC (1968).

7. J. F. Sagendorf, "A Program for Evaluating Atmospheric Dispersion From a Nuclear Power Station," Technical Memorandum ERL ARL-42, National Oceanic and Atmospheric Administration (1974).

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SECTION 2.4.1 HYDROLOGIC DESCRIPTION

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The areas of review under this plan are:

1. Identification of the interface of the plant with the hydrosphere.

2. Identification of hydrologic causal mechanisms that may require special plant design bases or operating limitations with regard to floods and water supply requirements.

3. Identification of surface and groundwater uses that may be affected by plant operation.

The review of Section 2.4.1.1 (Site and Facilities) of safety analysis reports (SAR) consists of comparing the independently verified or derived hydrologic design bases (see subsequent sections of 2.4) with the critical elevations of safety-related structures and facilities. The review of SAR Section 2.4.1.2 (Hydrosphere) requires identification of the hydrologic characteristics of streams, lakes (e.g., location, size, shape, drainage area, etc.), shore regions, the regional and local groundwater environments, and existing or proposed water control structures (upstream and downstream) influencing the type of flooding mechanisms which may adversely affect safety aspects of plant siting and operation.

II. ACCEPTANCE CRITERIA

Acceptance of the information presented in SAR Section 2.4.1.1 is based on a qualitative evaluation of the apparent completeness and quality of information, data, and maps. The description and elevations of safety-related structures, facilities, and accesses thereto should be sufficiently complete to allow evaluation of the impact of flood design bases. Site topographic maps must be of good quality and of sufficient scale to allow independent analysis of pre- and post-construction drainage patterns. All external plant structures and components should be identified on site maps. Data on surface water users, location with respect to the site, type of use, and quantity of surface water used are required.

The information presented in SAR Section 2.4.1.2 forms the basis for subsequent hydrologic engineering analysis. Therefore, completeness and clarity are of paramount importance. Maps must be legible and adequate in coverage to substantiate applicable data. Inventories of surface water users must be consistent with regional hydrologic inventories reported by applicable state and federal agencies. The description of the hydrologic characteristics of streams, lakes, and shore regions must correspond to those of the United States Geologic Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, or appropriate state and river basin agencies. Descriptions of all existing or proposed reservoirs and dams (both upstream and downstream) that could influence conditions at the site must be provided. Descriptions may be obtained from reports of the USGS, United States Bureau of Reclamation (USBR), Corps of Engineers, and others. Generally, reservoir descriptions of a quality similar to those contained in pertinent data sheets of a standard Corps of Engineers Hydrology Design Memorandum are adequate. Tabulations of drainage areas, types of structures, appurtenances, ownership, seismic and spillway design criteria, elevation-storage relationships, and short- and long-term storage allocations must be provided.

III. REVIEW PROCEDURES

The information presented in SAR Section 2.4.1.1 is not generally amenable to independent verification, except through cross-checks with other SAR sections and chapters, available publications relating to hydrologic characteristics of the site region, and by site visits. The review procedure consists of evaluating the completeness of the information and data by sequential comparison with information available from references. Based on the description of the hydrosphere (e.g., geographic location and regional hydrologic features) potential site flood mechanisms are identified. Subsequent SAR sections addressing the mechanisms are cross-checked to assure that data and information required therein for review and substantiation are available.

An important facet of the review procedure for this plan and for other plans in hydrologic areas is the site visit. The site visit provides the principal technical reviewer with independent confirmation of hydrologic characteristics of the site and adjacent environs. The site visit is discussed in Appendix A to this plan.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, findings will consist of a brief general description of the site with respect to the general hydrosphere, and the off-site uses of surface water. For operating license (OL) reviews, findings will consist of the same material, updated as required for new information available since preparation of the CP findings. A sample description for a CP review follows:

"The proposed site for the ABC Nuclear Plant is located about 26 miles SSE of Augusta, Maine, on the southwest bank of the DEF River at about river mile 152. Plant grade will be at about elevation 220 feet above mean sea level (MSL).

Significant hydrologically related plant features include the river intake structure, the natural draft cooling towers, mechanical draft nuclear service cooling towers (these are redundant towers and serve as the ultimate heat sink), and various groundwater wells."

V. REFERENCES

Because of the geographic diversity of plant sites and the large number of hydrologic references, no specific tabulation is given here. In general, maps and charts by the USGS, NOAA, Army Map Service (AMS), and Federal Aviation Administration (FAA); water-supply papers of the USGS; River Basin Reports of the Corps of Engineers; and other publications of state, federal, and other regulatory bodies, describing hydrologic characteristics and water utilization in the plant vicinity and region, are referred to on an "as available" basis. Other plans in the hydrology area (plans 2.4.2 through 2.4.14) contain references that are to be used in evaluating the hydrologic description of the site.

1. Appendix A, Standard Review Plan 2.4.1, "Hydrologic Engineering Site Visits," attached.

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

APPENDIX A

STANDARD REVIEW PLAN 2.4.1
HYDROLOGIC ENGINEERING SITE VISITS

I. PURPOSES

The purposes of hydrologic engineering site visits are as follows:

1. Acquaint the reviewer with general site and regional hydrologic characteristics and topography.

2. Confirm the applicant's general appraisal of the site/plant hydrologic interfaces.

3. Review specific hydrologic engineering problem areas with the applicant, his engineers, and his consultants.

The site visit objectives will have been achieved if, in addition to viewing pertinent hydrologic features, the reviewer has had the opportunity to discuss specific questions and concerns with the applicant's hydrologic engineers, and is assured that the questions and concerns are understood. In addition, generally acceptable techniques and procedures necessary to respond to staff concerns should be discussed.

II. PROCEDURES

Draft questions, or items of staff concern, are to be developed by the hydrologic engineering section reviewer and discussed in detail with the Section Leader 7-14 days before the scheduled site visit. For any unscheduled site visit (which may be necessary to resolve issues or prepare for hearings), similar draft questions or items of staff concern should be prepared at least 3 days prior to such site visit and also discussed in detail with the Section Leader.

Areas of overlap or interfaces with reviewers in other areas (such as geology, foundation engineering, auxiliary and power conversion systems, mechanical engineering, effluent treatment systems,and structural engineering) should be coordinated before final typing of drafts.

The Section Leader will discuss any unusual or potentially controversial areas of concern with the Chief, SAB, prior to transmittal of drafts to the Project Manager. Transmittal of the drafts will be by Memo Route Slip through the Section Leader.

Site visits are generally to consist of a detailed reconnaissance of site areas and environs with the applicant and technical counterparts, discussions of questions (or items of staff concern), discussions of acceptable methods of analysis, and a general summarization of the areas discussed and conclusions reached.

Normally, a small group composed of the staff reviewer and licensing project manager (LPM) should meet with an applicant representative responsible for responding to staff questions and the applicant's technical advisor. For verbal summarization during the site visit, the recommended method is to have the applicant or his technical advisor summarize the discussions to assure understanding.

III. <u>TRIP REPORT</u>

A trip report on a site visit should be prepared within 5 days of the reviewer's return. The report is to be as brief as possible and should summarize the trip and the areas of discussion and should list the participants in technical discussions. Within 7-10 days of returning, the reviewer should prepare final question lists, updating the draft for new areas of concern, deleting areas for which the site visit revealed that no safety or environmental problems remain, and clarifying draft questions based upon discussions and information obtained during the site visit.

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SECTION 2.4.2 FLOODS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

This section of the safety analysis report (SAR) identifies historical flooding (defined as occurrences of abnormally high water stage, or overflow from a stream, floodway, lake, or coastal area) at the proposed site or in the region of the site. It summarizes and identifies the individual types of flood-producing phenomena, and combinations of flood-producing phenomena, considered in establishing the flood design bases for safety-related plant features. It also covers the potential effects of local intense precipitation. Although the material may appear in another SAR section, the following matters are included with review of this subject.

The history and the potential for flooding from each of the following sources and events are reviewed:

1. Stream flooding;
   a. Probable maximum flood (PMF) with coincident wind-induced waves, considering dam failure potential due to inadequate capacity, inadequate flood-discharge capability, or existing physical condition.
   b. Ice jams, both independently and coincident with a winter probable maximum storm.
   c. Tributary drainage area PMF potential.
   d. Combinations of less severe river floods, coincident with surges and seiches.
2. Surges;
   a. Probable maximum hurricane (PMH) at coastal sites.
   b. PMH wind translated inland and resulting wave action coincident with runoff-induced flood levels.
   c. Probable maximum wind-induced (non-hurricane) storm surges and waves.
   d. Combinations of less severe surges, coincident with runoff floods.

3. Seiches;
   a. Meteorologically-induced in inland lakes (e.g. Great Lakes and harbors) and at coastal harbors and embayments.
   b. Seismically-induced in inland lakes.
   c. Seismically-induced by tsunamis (seismic sea waves) on coastal embayments.
   d. Combinations of less severe surges and seiches, coincident with runoff floods.
4. Tsunamis;
   a. Near field, or local, excitation.
   b. Far field, or distant, excitation.
5. Seismically-induced dam failures (or breaches), and maximum water level at site from;
   a. Failure of dam (or dams) during safe shutdown earthquake (SSE) coincident with 25-year flood.
   b. Failure during operating basis earthquake (OBE) coincident with standard project flood (SPF).
   c. Failure during other earthquakes, coincident with runoff, surge, or seiche floods where the coincidence is at least as likely as for 5.a. and 5.b. above.
6. Ice loadings from water bodies.

II. ACCEPTANCE CRITERIA

For SAR Section 2.4.2.1 (Flood History): the potential flood sources and flood response characteristics identified by the staff's review (described in Review Procedures) are compared to those of the applicant. If similar, the applicant's conclusions are accepted. If, in the staff's opinion, significant discrepancies exist, the applicant will be requested to provide additional data, reestimate the effects on the plant, or revise the applicable flood design bases, as appropriate.

For SAR Section 2.4.2.2 (Flood Design Considerations): the controlling flood levels independently determined (or verified) by the staff will be compared with the applicant's. The two levels, referenced to mean or normal water levels, should be within about 5 percent. If the SAR estimate is more than 5 percent low, the applicant should fully document and justify the SAR estimate of the controlling level, or accept the staff estimate and redesign applicable flood protection.

For SAR Section 2.4.2.3 (Effect of Intense Precipitation): the staff estimate of local probable maximum precipitation (PMP) and the capacity of site drainage facilities (including drainage from the roofs of buildings and site ponding) are compared with values in the SAR. The applicant's SAR estimates of PMP and capacity of site drainage facilities should be within about 5 percent of corresponding staff estimates; or, using the staff estimates, no hazard must be judged to exist to safety-related facilities for the SAR estimate to be acceptable. Similarly, conclusions relating to the potential for any adverse affects of blockage of site drainage facilities by debris, ice,or snow should be based upon conservative assumptions of storm and vegetation conditions likely to exist during storm periods. If a potential hazard does exist (e.g., the elevation of ponding exceeds the elevation of plant access openings) the applicant should document and justify his local PMP basis and analysis and redesign any affected facilities.

III. REVIEW PROCEDURES

For SAR Section 2.4.2.1 (Flood History): the staff will review publications of the U. S. Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, applicable state and river basin agencies, and others to ensure that historical maximum events and the flood response characteristics of the region and site have been identified. Similar material, in addition to applicant-supplied information, will be reviewed to identify independently the potential sources of site flooding.

For SAR Section 2.4.2.2 (Flood Design Considerations): the flood potential (level) from consideration of the worst single phenomenon, and combinations of less severe phenomena, as discussed in Regulatory Guides 1.27 (Ultimate Heat Sink) and 1.59 (Design Basis Floods), are compared with the proposed protection levels of safety-related facilities. Methods of analysis to determine the individual flood-producing phenomena are discussed in Standard Review Plans 2.4.3 through 2.4.7.

For SAR Section 2.4.2.3 (Effect of Local Intense Precipitation): staff estimates of flooding potential are based on 24-hour PMP estimates (from Hydrometeorological Report 33 and similar NOAA publications for western sites), with time distributions from the Corps of Engineers EM 1110-2-1411, and are developed by the staff for comparison with the applicant's estimate. Runoff models, such as the unit hydrograph if applicable, or other runoff discharge estimates presented in standard texts, are used to estimate discharge on the site drainage system. Where generalized runoff models are used, coefficients used for the site and region are compared to information available at documented locations to evaluate hydrologic conditions used in determining the probable maximum flood for the site-drainage system. Potential ponding on the site is also determined.

Construction permit (CP) stage reviews are carried out as indicated in this plan and to evaluate its significance with regard to the plant design basis for flood protection. At the operating license (OL) stage, a brief review is carried out to determine if new information has become available since the CP review and to evaluate its significance with regard to the plant design basis for flood protection. New information might arise, for instance, from the occurrence of a new maximum flood of record in the site region, from identification of a source of major flooding not previously considered, construction of new dams, flood plain incroachments, or from advances in predictive models and analytical techniques. If the CP-stage evaluation of flooding potential has been carefully done, all sources of major flooding should have been considered and any new floods of record should fall well within the design basis. Improvements in calculational methods may occur, but generally will be concerned with increased accuracy in stream flow and water level predictions rather than with substantive changes in the flows and levels predicted. It is not the intention of the staff to request adjustments in the flood design basis for a plant because "improved," OL-stage calculations of flows and water levels result in slightly different values than those accepted at the CP stage. Where the OL review does reveal significant differences from the CP evaluation, any supplemental provisions needed in the flood protection design basis should be directed primarily toward early warning measures and procedures for assuring safe shutdown of the plant.

Consultants may be employed by the staff in an advisory role in developing independent staff analyses of the potential for flooding, or in independently making other specific assessments, depending on the complexity of the analysis and the availability of staff manpower. The consultants may be from Coastal Engineering Research Center (CERC) or private contractors.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit reviews, the findings will consist of a statement indicating the completeness of the identification of site flood characteristics and flood design bases. For OL reviews, the flood history will be updated if necessary, with special attention to any new flood of record. Sample statements for CP reviews follow:

"The maximum flood known to have occurred on the A River was in 1796. The peak discharge at the B City, Montana, was estimated to be 360,000 cubic feet per second (cfs). The applicant estimated that a comparable flood would produce a water surface elevation at the site of 116 feet MSL. The maximum flood during the period since records were maintained (1883) at B City was 350,000 cfs and occurred on October 3, 1929. These floods occurred prior to construction of several upstream dams. Flood flows are now regulated by C and D Reservoirs as well as by upstream hydropower plants.

"The applicant has estimated potential flooding from rainfall over the E River basin upstream from the site. The probable maximum flood (PMF), the upper level of flooding the staff considers to be reasonably possible, was estimated to produce a flow of 5,000,000 cfs near the City of F. This estimate was made by using 165% of the Corps of Engineers project design flood (PDF) estimate of 3,030,000 cfs at the same location, as modified by upstream flood control reservoirs. The 3,030,000 cfs project design flood flow is estimated to be partially diverted to the leveed G and H Floodways upstream of the site, with 1,500,000 cfs continuing downstream within the levee system past the plant site. The applicant concluded that the PMF could result in overtopping of levees and flooding of the river valley well upstream from the site, thereby causing generally low level flooding in the plant area. The upstream levee overtopping and resulting valley flow during such an event would reduce the flow in the main levee channel adjacent to the site to levels equal to or less than those that would exist during a project design flood. We conclude that the combination of a runoff-type flood less severe than a PMF, but more severe than a PDF, and a coincident levee break in the vicinity of the site could occur before water approaches levee grade upstream. A failure or levee breach, when the levee is full to design capacity (3 feet below the top of the levee adjacent to the site plus the effects of any coincident wind-generated wave activity), would result in a higher water surface at the plant than a PMF spread over the valley as a result of levee failures upstream. At our request the applicant evaluated various modes of levee failure in the vicinity of the plant. One of the conditions postulated is that of a flood, approaching the severity of a PMF, causing a massive failure of the upstream left bank levee along the G Floodway, resulting in flooding around the plant, coincident with a failure of the levee adjacent to the plant site. The applicant estimated the resulting water level at the plant would reach elevation 22.5 feet MSL for this case. The case of an instantaneous levee failure adjacent to the plant, with no upstream levee failure, resulted in an estimated water level of 24.6 feet MSL. The staff concludes that the applicant should design for the conditions associated with the 24.6 feet MSL water level."

V. REFERENCES[1/]

1. "Surface Water Supply of the United States,"[2/] U.S. Geological Survey.

2. "Tide Tables," National Oceanic and Atmospheric Administration (similar situation as identified in footnote 2).

3. Reports of Great Lakes levels by Lake Survey Denver, National Oceanic and Atmospheric Administration.

4. Corps of Engineers records maintained in District and Division Offices, Coastal Engineering Research Center, and Waterways Experiment Station.

5. Regulatory Guide 1.27, "Ultimate Heat Sink," (Revision 2).

6. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

7. "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24, and 48 Hours," Hydrometeorological Report No. 33, U.S. Weather Bureau (1956).

8. "Standard Project Flood Determinations," Engineering Manual 1110-2-1411, Corps of Engineers, 26 March 1952 (rev. March 1965).

9. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

1/ References for PMP estimates, time distribution, etc., are in Standard Review Plan 2.4.3.

2/ "Surface Water Supply" is a continuing series of water discharge measurements by the USGS and others. It is not practical to list all the volumes (called "Water-Supply Papers") that are available. Numerous state and local authorities maintain river discharge, lake level, and tide data.

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SECTION 2.4.3 PROBABLE MAXIMUM FLOOD (PMF) ON STREAMS AND RIVERS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrometeorological design basis is developed to determine the extent of any flood protection required for safety-related plant systems, as discussed in Regulatory Guide 1.59. The areas of review include the precipitation potential, precipitation losses, the runoff response characteristics of the watershed, the accumulation of flood runoff through river channels and reservoirs, the estimate of the discharge rate trace (hydrograph) of the probable maximum flood (PMF) at the plant site, the determination of PMF water level conditions at the site, and the evaluation of coincident wind-generated wave conditions that could occur with the PMF. Included is a review of the details of design bases for site drainage (which is summarized in SAR Section 2.4.2) and a review of the probable maximum precipitation (PMP) potential and resulting runoff for site drainage and drainage areas adjacent to the plant site, and including the roofs of safety-related structures. The analyses involve modeling of physical rainfall and runoff processes to estimate the upper level of possible flood conditions adjacent to and onsite.

Regulatory Guide 1.59 describes two positions with respect to flood protection. While both require an estimate of the PMF in determining the controlling design basis conditions, Position 2 limits the applicability of the design bases to specific equipment and facilities. If Position 2 is applicable, the review will be limited to the equipment and facilities identified in the guide.

II. ACCEPTANCE CRITERIA

The probable maximum flood as defined in Regulatory Guide 1.59 has been adopted as one of the conditions to be evaluated in establishing the applicable stream and river flooding design basis referred to in General Design Criterion 2, Appendix A, 10 CFR Part 50. The criteria for accepting the applicant's PMF-related design bases depend on the relative significance of the flood. PMF estimates are required for all adjacent streams or rivers and site drainage (including the roofs of safety-related structures). One of three conditions may exist at the site under review, as follows:

1. The elevation attained by the PMF (with coincident wind waves) establishes a required protection level.

2. The elevation attained by the PMF (with coincident wind waves) is not controlling; the design basis flood protection level is established by another flood phenomena (e.g., the probable maximum hurricane).

3. The site is "dry", that is, the site is well above the elevation attained by a PMF (with coincident wind waves).

When condition 1 is applicable the staff will estimate the flood level as described below. The estimate may be made independently from basic data, by detailed review and checking of the applicant's analyses, or by comparison with estimates made by others which have been reviewed in detail. Acceptance is based on agreement of the staff and applicant estimates of static flood level to within about 2 feet or higher and of coincident wave action to within about 1 foot or higher.

When conditions 2 or 3 apply, the staff analyses may be less rigorous, as described below. For condition 2, acceptance is based on the protection level estimated for another flood-producing phenomenon exceeding the staff estimate of PMF water levels. For condition 3, the site grade must be well above the staff estimate of PMF water levels. The evaluation of the adequacy of the margin (difference in flood and site elevations) is generally a matter of engineering judgement. The judgement is based on the confidence in the flood level estimate and the degree of conservatism in each parameter used in the estimate.

III. REVIEW PROCEDURES

The review procedure is outlined in Figures 2.4.3-1 and -2, attached to this plan. In addition, Appendix A to Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," is a codification of techniques used by the staff during the review and verification of PMF estimates. The evaluation of flooding potential is, for review purposes, separated into two parts; PMF on adjacent streams and local PMF. The review procedure for the former is indicated on Figure 2.4.3-1. The review procedure for evaluating a local PMF is indicated on Figure 2.4.3-2. (The procedure for evaluating the adequacy of site drainage facilities based on a local PMF is outlined in Standard Review Plan 2.4.2.) PMF estimates approved by the Chief of Engineers, Corps of Engineers, and contained in published or unpublished reports of that agency, or generalized estimates may be used in lieu of staff-developed estimates. In the absence of such estimates, the staff will use both large and small basin PMP estimates by the National Oceanic and Atmospheric Administration (NOAA) and published techniques of the World Meteorological Organization in conjunction with Corps of Engineers' runoff, impoundment, and river routing models to estimate PMF discharge and water level at the site. These methods are used for conditions 1 and 2, described in the acceptance criteria. When detailed independent estimates are necessary (see acceptance criteria), the applicant will be requested to provide any necessary basic data. Wind-generated wave action will be independently estimated using Corps of Engineers criteria. Where sufficient water depth is available, the significant wave height and runup are used for structural design purposes, and the maximum (one percent) wave height and runup are used for flood level estimates. Where depth limits wave height, the breaking wave height and runup is used for both purposes.

When an applicant has chosen to demonstrate a "dry" site (i.e., condition 3, one not subject to stream flooding by virtue of local topographic considerations), the following procedures apply:

1. Use Corps of Engineers PMF estimates for other sites in the region to develop "regional drainage area vs. PMF discharge (cubic feet per second/square mile)" data, for extrapolation to the site.
2. Envelope the above data points to obtain an estimate of the PMF applicable to the site.
3. Increase the estimate based on a judgement as to the applicability of the basic estimates. An increase in the range of 10 to 50 percent is generally appropriate.
4. Estimate the flood level at the site using slope-area techniques or water surface profile computations, if warranted by relative elevation differences between the site and adjacent stream.
5. Estimate wind (40 mph over land) wave runup based on breaking, or maximum (one percent) wave.
6. Compare resultant water level with proposed plant grade and lowest safety-related facility that can be affected.

Consultants may be employed in an advisory role in developing independent staff flood effect estimates, depending on the complexity of the analysis required and available staff manpower. The consultants may be from the Hydrometeorological Branch of the U. S. Weather Service, the Corps of Engineers Coastal Engineering Research Center, or private contractors.

The above items of review are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings consist of a statement describing the applicant and staff estimates of the peak PMF runoff rate and water level (including allowance for coincident wind-generated wave activity) at the site. If the estimates are similar, staff concurrence will be stated. If the staff predicts substantially more severe flood conditions which may adversely affect the proposed plant, and the applicant has been unable to support his estimates, a statement requiring use of the staff bases will be made. If the flood conditions do not constitute a design basis, the statement will so indicate.

For operating license (OL) reviews which have received detailed PMF reviews during the CP review, the CP conclusions will be referenced. If no CP PMF review was undertaken (of the scope indicated), this fact will be indicated in the OL findings. Any flood potential not identified during the CP review will be noted.

If Regulatory Guide 1.59, Position 2, is elected by the applicant, a statement describing lesser design bases will be included in the findings with a staff conclusion of adequacy.

A sample statement for a CP review follows:

"The Probable Maximum Flood (PMF) resulting from the Probable Maximum Precipitation (PMP) on the ABC River drainage basin yielded an estimated maximum stillwater level at the intake structure on the D & E Canal of about 5.0 feet MSL.

"The PMF resulting from a local PMP storm on the drainage basins for the small streams near the site yielded an estimated maximum stillwater level of about 60 feet MSL, which is about 20 feet below plant grade.

"The local PMF resulting from the estimated local PMP was found not to cause flooding of safety-related facilities, since the site drainage system will be capable of functioning adequately during such a storm. Catch basins will be provided as part of the storm drainage system and will be located throughout the plant site to drain local areas. The plant yard will be graded with gentle slopes away from high points at the plant buildings, and storm water will drain away from the buildings into the local streams at lower elevations."

V. REFERENCES

In addition to the following specific references, Design Memoranda, Civil Works Investigations and research and development reports of the Corps of Engineers and reports of other federal and state agencies relevant to flood estimates at a specific site will be used on an "as available" basis.

1. Reports of the Corps of Engineers, Department of the Army:
   EM 1110-2-1411, "Standard Project Flood Determinations", 26 March 1952 (rev. March 1965).

   EC 1110-2-27, "Policies and Procedures Pertaining to Determination of Spillway Capacities and Freeboard Allowances for Dams," 19 February 1968.

   EM 1110-2-1405, "Flood Hydrograph Analysis and Computations," 31 August 1959.

   EM 1110-2-1408, "Routing of Floods Through River Channels," 1 March 1960.

   EM 1110-2-1406, "Runoff from Snowmelt," 5 January 1960.

   EM 1110-2-1603, "Hydraulic Design of Spillways," 31 March 1965.

   EM 1110-2-1409, "Backwater Curves in River Channels," 7 December 1959.

   Technical Bulletin No. 8, Sacramento District, "Generalized Snowmelt Runoff Frequencies," September 1962.

   EM 1110-2-1601, "Hydraulic Design of Flood Control Channels," 1 July 1970.

   EM 1110-2-1607, "Tidal Hydraulics," 2 August 1965.

CE 1308, "Stone Protection," January 1948.

EM 1110-2-1410, "Interior Drainage of Leveed Urban Areas: Hydrology," 3 May 1965.

Technical Report No. 4, Coastal Engineering Research Center, "Shore Protection, Planning and Design," 1966 and "Shore Protection Manual," (1973).

Waterways Experiment Station, "Hydraulic Design Criteria," continuously updated.

TSP37, "Riprap Stability on Earth Embankments Tested in Large and Small-Scale Wave Tanks," June 1972.

ETL 1110-2-120, "Additional Guidance for Riprap Channel Protection," May 1971.

2. Hydrometeorological Reports of the U. S. Weather Bureau (now U. S. Weather Service, NOAA), Hydrometeorological Branch:
No. 1., "Maximum Possible Precipitation Over the Ompompanoosuc Basin above Union Village, Vt." (1943).

No. 2., "Maximum Possible Precipitation over the Ohio River Basin above Pittsburgh, Pa." (1942).

No. 3., "Maximum Possible Precipitation over the Sacramento Basin of California" (1943).

No. 4., "Maximum Possible Precipitation over the Panama Canal Basin" (1943).

No. 5., "Thunderstorm Rainfall" (1947).

No. 6., "A Preliminary Report on the Probable Occurrence of Excessive Precipitation over Fort Supply Basin, Okla." (1938).

No. 7., "Worst Probable Meteorological Condition on Mill Creek, Butler and Hamilton Counties, Ohio" (1937), unpublished. Supplement (1938).

No. 8., "A Hydrometeorological Analysis of Possible Maximum Precipitation over St. Francis River Basin above Wappapello, Mo." (1938).

No. 9., "A report on the Possible Occurrence of Maximum Precipitation over White River Basin above Mud Mountain Dam Site, Wash." (1939).

No. 10., "Maximum Possible Rainfall over the Arkansas River Basin above Caddoa, Colo." (1939) Supplement (1939).

No. 11., "A Preliminary Report on the Maximum Possible Precipitation over the Dorena, Cottage Grove, and Fern Ridge Basins in the Willamette Basin, Oreg." (1939).

No. 12., "Maximum Possible Precipitation over the Red River Basin above Denison, Tex." (1939).

No. 13., "A Report on the Maximum Possible Precipitation over Cherry Creek Basin in Colorado"(1940).

No. 14., "The Frequency of Flood-Producing Rainfall over the Pajaro River Basin in California" (1940).

No. 15., "A Report on Depth-Frequency Relations of Thunderstorm Rainfall on the Sevier Basin, Utah" (1941).

No. 16., "A Preliminary Report on the Maximum Possible Precipitation over the Potomac and Rappahannock River Basins" (1943).

No. 17., "Maximum Possible Precipitation over the Pecos Basin of New Mexico" (1944), unpublished.

No. 18., "Tentative Estimates of Maximum Possible Flood-Producing Meteorological Conditions in the Columbia River Basin" (1945).

No. 19., "Preliminary Report on Depth-Duration-Frequency Characteristics of Precipitation over the Muskingum Basin for 1- to 9-week Periods" (1945).

No. 20., "An Estimate of Maximum Possible Flood-Producing Meteorological Conditions in the Missouri River Basin above Garrison Dam Site" (1945).

No. 21., "A Hydrometeorological Study of the Los Angeles Area" (1939).

No. 21A., "Preliminary Report on Maximum Possible Precipitation, Los Angeles Area, California" (1944).

No. 21B., "Revised Report on Maximum Possible Precipitation, Los Angeles Area, California" (1945).

No. 22., "An Estimate of Maximum Possible Flood-Producing Meteorological Conditions in the Missouri River Basin Between Garrison and Fort Randall" (1946).

No. 23., "Generalized Estimates of Maximum Possible Precipitation over the United States East of the 105th Meridian, for Areas of 10, 200, and 500 Square Miles" (1947).

No. 24., "Maximum Possible Precipitation over the San Joaquin Basin, Calif." (1947).

No. 25., "Representative 12-hour Dewpoints in Major United States Storms East of the Continental Divide" (1947).

No. 25A., "Representative 12-hour Dewpoints in Major United States Storms East of the Continental Divide," 2d edition (1949).

No. 26., "Analysis of Winds over Lake Okeechobee during Tropical Storm of August 26-27, 1949" (1951).

No. 27., "Estimate of Maximum Possible Precipitation, Rio Grande Basin, Fort Quitman to Zapata" (1951).

No. 28., "Generalized Estimate of Maximum Possible Precipitation over New England and New York" (1952).

No. 29., "Seasonal Variation of the Standard Project Storm for Areas of 200 and 1,000 Square Miles East of the 105th Meridian" (1953).

No. 30., "Meteorology of Floods at St. Louis" (1953), unpublished.

No. 31., "Analysis and Synthesis of Hurricane Wind Patterns over Lake Okeechobee, Florida" (1954).

No. 32., "Characteristics of United States Hurricanes Pertinent to Levee Design for Lake Okeechobee, Florida" (1954).

No. 33., "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24, and 48 Hours" (1956).

Draft Report, "All-Season Probable Maximum Precipitation, United States East of the 105th Meridian for Areas From 1,000 to 20,000 Square Miles and Durations From 6 to 72 Hours" (1972).

No. 34., "Meteorology of Flood-Producing Storms in the Mississippi River Basin" (1956).

No. 35., "Meteorology of Hypothetical Flood Sequences in the Mississippi River Basin" (1959).

No. 36., "Interim Report, Probable Maximum Precipitation in California" (1961), revised (1969).

No. 37., "Meteorology of Hydrologically Critical Storms in California" (1962).

No. 38., "Meteorology of Flood-Producing Storms in the Ohio River Basin" (1961).

No. 39., "Probable Maximum Precipitation in the Hawaiian Islands" (1963).

No. 40., "Probable Maximum Precipitation, Susquehanna River Drainage above Harrisburg, Pa." (1965).

No. 41., "Probable Maximum and TVA Precipitation over the Tennessee River Basin above Chattanooga" (1965).

No. 42., "Meteorological Conditions for the Probable Maximum Flood on the Yukon River above Rampart, Alaska" (1966).

No. 43., "Probable Maximum Precipitation, Northwest States" (1966).

No. 44., "Probable Maximum Precipitation over South Platte River, Colorado, and Minnesota River, Minnesota" (1969).

No. 45., "Probable Maximum and TVA Precipitation for Tennessee River Basin up to 3,000 Square Miles in Area and Durations to 72 Hours" (1969).

No. 46., "Probable Maximum Precipitation, Mekong River Basin" (1970).

No. 47., "Meteorological Criteria for Extreme Floods For Four Basins in the Tennessee and Cumberland River Basins" (1973).

No. 48., "Probable Maximum Precipitation and Snowmelt Criteria for Red River of the North Above Pembinz, and Souris River Above Minot, North Dakota" (1973).

3. Technical Papers of the U. S. Weather Bureau (Now U. S. Weather Service, NOAA):
No. 2., "Maximum Recorded United States Point Rainfall for 5 Minutes to 24 Hours at 207 First Order Stations," Rev. (1963).

No. 5., "Highest Persisting Dewpoints in the Western United States" (1948).

No. 10., "Mean Precipitable Water in the United States" (1949).

No. 13., "Mean Monthly and Annual Evaporation Data from Free Water Surface for the United States, Alaska, Hawaii, and the West Indies" (1950).

No. 14., "Tables of Precipitable Water and Other Factors for a Saturated Pseudo-Adiabatic Atmosphere" (1951).

No. 15., "Maximum Station Precipitation for 1, 2, 3, 6, 12, and 24 Hours:" Part I: Utah (1951); Part II: Idaho (1951); Part III: Florida (1952); Part IV: Maryland, Delaware, and District of Columbia (1953); Part V: New Jersey (1953); Part VI: New England (1953); Part VII: South Carolina (1953); Part VIII: Virginia (1954); Part IX: Georgia (1954); Part X: New York (1954); Part XI: North Carolina (1955); Part XII: Oregon (1955); Part XIII: Kentucky (1955); Part XIV: Louisiana (1955); Part XV: Alabama (1955); Part XVI: Pennsylvania (1956); Part XVII: Mississippi (1956); Part XVIII: West Virginia (1956); Part XIX: Tennessee (1956); Part XX: Indiana (1956); Part XXI: Illinois (1958); Part XXII: Ohio (1958); Part XXIII: California (1959); Part XXIV: Texas (1959); Part XXV: Arkansas (1960); Part XXVI: Oklahoma (1961).

No. 16., "Maximum 24-Hour Precipitation in the United States" (1952).

No. 25., "Rainfall Intensity-Duration-Frequency Curves for Selected Stations in the United States, Alaska, Hawaiian Islands, and Puerto Rico" (1955).

No. 28., "Rainfall Intensities for Local Drainage Design in Western United States for Durations of 20 Minutes to 24 Hours and 1- to 100-Year Return Periods" (1956).

No. 37., "Evaporation Maps for the United States" (1959).

No. 38., "Generalized Estimates of Probable Maximum Precipitation for the United States West of the 105th Meridian for Areas to 400 Square Miles and Durations to 24 Hours" (1960).

No. 40., "Rainfall Frequency Atlas of the United States for Durations from 30 Minutes to 24 Hours and Return Periods from 1 to 100 Years" (1961).

No. 42., "Generalized Estimates of Probable Maximum Precipitation and Rainfall-Frequency Data for Puerto Rico and Virgin Islands" (1961).

No. 43., "Rainfall-Frequency Atlas of the Hawaiian Islands for Areas to 200 Square Miles, Durations to 24 Hours, and Return Periods from 1 to 100 Years" (1962).

No. 47., "Probable Maximum Precipitation and Rainfall-Frequency Data for Alaska for Areas to 400 Square Miles, Durations to 24 Hours, and Return Periods from 1 to 100 Years" (1963)

No. 48., "Characteristics of the Hurricane Storm Surge" (1963).

4. Unpublished Hydrometeorological Reports of the U. S. Weather Bureau (now U. S. Weather Service, NOAA):
"Rappahannock River above Salem Church Dam Site, Va." (11/28/50).

"Potomac River, Va., Md., W. Va., (12 sub-basins)" (6/29/56).

"Delaware River above Trenton, Chestnut Hill, and Belvidere Dam Sites" (11/19/56).

"Delaware River above Tock's Island Dam Site" (12/16/65).

"St. John River above Dickey Dam Site, and Between Dicky and Lincoln School Dam Sites, Maine" (12/20/66).

"Coosa River above Howell Mill Shoals Dam Site, Ala." (3/3/50).

"Cape Fear River above Smiley Falls Dam Site, N.C." (11/16/50).

"Savannah River above Hartwell Dam Site, N.C." (1/5/51).

"Alabama and Appalachicola Rivers, Ala. and Fla." (3/19/52).

"Black Warrior River above Holt Lock Dam Site, Ala." (12/10/59).

"South Fork of Holston River above Boone Dam Site, Tenn." (8/14/50).

"Allegheny River above Allegheny River Reservoir, Pa." (9/28/56).

"Kentucky River, Ky. (2 basins)" (3/12/58).

"New River above Moores Ferry Dam Site, Va." (5/13/63).

"Licking River, Ky, and White River, Ind." (11/9/64).

"Iowa River above Coralville Dam Site, Iowa" (11/20/47).

"Des Moines River above Saylorville, Iowa and Howell Dam Site, Iowa" (3/19/48).

"Salt River, Mo." (1/21/55).

"James River above Jamestown Dam Site, N. Dak." (9/16/48).

"Big Blue River above Tuttle Creek Dam Site, Kans." (10/23/51).

"Republican River at (a) above proposed Milford Dam Site, Kan.; and (b) between Harlan Co. Dam and proposed Milford Dam Site, Kans." (11/24/58).

"Meramec River Basin, Missouri" (12/21/61).

"Republican River above Harlan Co. Res., Neb." (3/7/69).

"Canadian River above Eufaula Dam Site, Okla." (12/19/47).

"White River above Table Rock Dam Site, Mo." (3/19/48).

"Eleven Point River above Water Valley Dam Site, Ark." (3/19/48).

"Kiamichi River above Hugo Dam Site, Okla." (4/9/48).

"Boggy Creek above Boswell Dam Site, Okla." (4/9/48).

"North Canadian River above Optima (Hardesty) Dam Site, Okla." (12/22/49).

"Lower Canadian River, Okla." (6/10/48).

"Gaines Creek Dam Site, Okla." (5/13/48).

"Onapa-Canadian (combined) Dam Site, Okla." (5/13/48).

"Verdigris River above Oologah Dam Site, Okla." (5/4/50).

"Little Red River above Green Ferry, Ark." (7/24/50).

"Grand (Neosho) River above Strawn Dam Site, Kans." (11/14/51).

"Pinon Canyon above Trinidad, Colo." (4/10/52).

"Beaver Reservoir, White River, Ark." (12/1/55).

"Kisatchie Dam Site on Kisatchie Bayou, La." (3/1/56).

"Cypress Creek above Mooringsport, La." (8/27/56).

"Little River above at (a) Millwood Dam Site, Ark.; and (b) Broken Bow, Okla." (5/14/59).

"White River Drainage above Wolf Bayou, Ark." (3/31/66).

"Upper Arkansas River, Colorado (sub-basins)" (2/13/67).

"Arkansas River Drainage Between John Martin Dam, Colo. and Great Bend, Kans." (9/23/69).

"Leon River above Belton Dam Site, Tex." (12/9/47).

"Jemez Creek, N. Mex." (12/9/49).

"Chama River above Chamita Dam Site, N. Mex." (1/18/50).

"Rio Hondo above Two Rivers Reservoir, N. Mex." (12/19/56).

"Richland Creek, Tex." (4/6/56).

"Basque River above Waco Reservoir, Tex." (4/6/56).

"Leon River above Proctor Reservoir Project near Hasse, Tex." (12/5/56).

"Pecos River above Alamogardo Reservoir, N. Mex." (7/24/57).

"Pecos River above Los Esteros, N. Mex." (7/24/57).

"Intervening Drainage between Los Esteros and Alamogardo, N. Mex." (7/24/57).

"Rio Grande between Cerro and Cochiti Dam Site, N. Mex." (2/26/58).

"Combined Drainage of Santa Fe Creek and Rio Galisto above Galisto Dam Site, N. Mex." (2/26/58).

"Lamposas River above proposed Lamposas Dam Site, Tex." (4/17/58).

"Navasota River, Tex. (7 sub-basins)" (11/2/59).

"Colorado River above Fox Crossing, Tex." (11/12/63).

"Lower Rio Grande, United States and Mexico (between Falcon and Anzalduas Dams)" (7/68).

"Gila River above Coolidge Dam Site, Ariz." (9/14/53).

"Queen Creek, Gila River Basin, Ariz." (4/26/55).

"Bill Williams River above proposed Alamo Dam Site, Ariz." (1/14/58).

"Santa Rosa Wash Basin, Ariz." (8/2/68).

"Black Creek, Ariz." (6/20/69).

"Preliminary Estimate for Drainages North of Phoenix, Ariz." (9/29/72).

"Humboldt River, Devils Gate Dam Site, Nev." (11/20/51).

"Mathews Canyon Dam Site (Virgin River), Nev. and Pine Canyon Dam Site (Virgin River), Nev." (8/9/54).

"Dell Canyon Reservoir, Utah" (8/26/57).

"Las Vegas Wash, Nev." (11/22/60).

"Henderson Wash, Nev." (11/22/60).

"West Fork (Mojave River), Calif." (11/22/60).

"Tahchevah Creek, Calif." (11/22/60).

"San Gorgonio River above Cabazon Dam Site, Calif." (4/13/62).

"Whitewater River above Garnet Dam Site, Calif." (4/13/62).

"Martis Creek, Calif." (3/18/64).

"Merced River, Calif." (6/4/62).

"American River above Folsom Dam, Calif." (8/1/68).

"North and Middle Forks of American River above Auburn Dam Site, Calif." (8/1/68).

"Intervening Drainage between Auburn Dam Site and Folsom Dam" (8/1/68).

"Yuba River above Marysville, Calif." (11/29/68).

"Los Angeles District, Calif. (18 basins in Calif, Nev. and Ariz.)" (12/2/68).

"San Diego River Watershed, Calif. (13 sub-basins)" (3/16/73).

"Skagway River, Alaska" (7/8/47).

"Bradley Lake Basin, Alaska" (5/19/61).

"Chena River, Alaska" (8/1/62).

"Long Lake portion of the Snettisham Project" (4/19/65).

"Takatz Creek, Baranof Island, Alaska" (2/21/67).

"Tanana River Basin for (a) Chena River above Chena Dam Site; (b) Little Chena River above Little Chena Dam; and (c) Tana River between Tanacross and Nenana, Alaska" (6/5/69).

"Preliminary Estimates, Vicinity of Junea: Mendenhall River, Lemon Creek, and Montana Creek" (11/7/69).

"Preliminary Estimates, Vicinity of Ketchikan: Whipple Creek near Wards Cove, Carlanna Creek near Ketchikan, Hoadley Creek near Ketchikan, and Ketchikan Creek near Ketchikan" (1/7/74).

"Eastern Panama and Northwest Colombia" (9/65).

"Hypothetical Rainstorms over Rio Atrato Basin, Colombia, South America" (7/67).

"Probable Maximum Thunderstorm Precipitation Estimates Southwest States" (3/30/73).

5. J. R. Weggel, "Maximum Breaker Height," Jour. Waterways, Harbors and Coastal Engineering Division, Proc. Am. Soc. of Civil Engineers, Vol. 98, No. WW4, pp. 529-548 (1972).

6. Technical Note 98, "Estimation of Maximum Floods," WMO-No. 233, World Meteorological Organization (1969).

7. C. O. Clark, "Storage and the Unit Hydrograph," Trans. Am. Soc. Civil Engineers, Vol. 110, No. 2261, pp. 1419-1488 (1945).

8. U.S. Department of Commerce, "Snow Hydrology," PB-151660, undated.

9. Bureau of Reclamation, "Effect of Snow Compaction from Rain on Snow," Engineering Monograph No. 35, U. S. Department of the Interior (1966).

10. Bureau of Reclamation, "Design of Small Dams," Second Edition, U. S. Department of the Interior (1973).

11. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

12. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

FIGURE 2.4.3-1

STANDARD REVIEW PLAN 2.4.3
FLOODS ON STREAMS & RIVERS

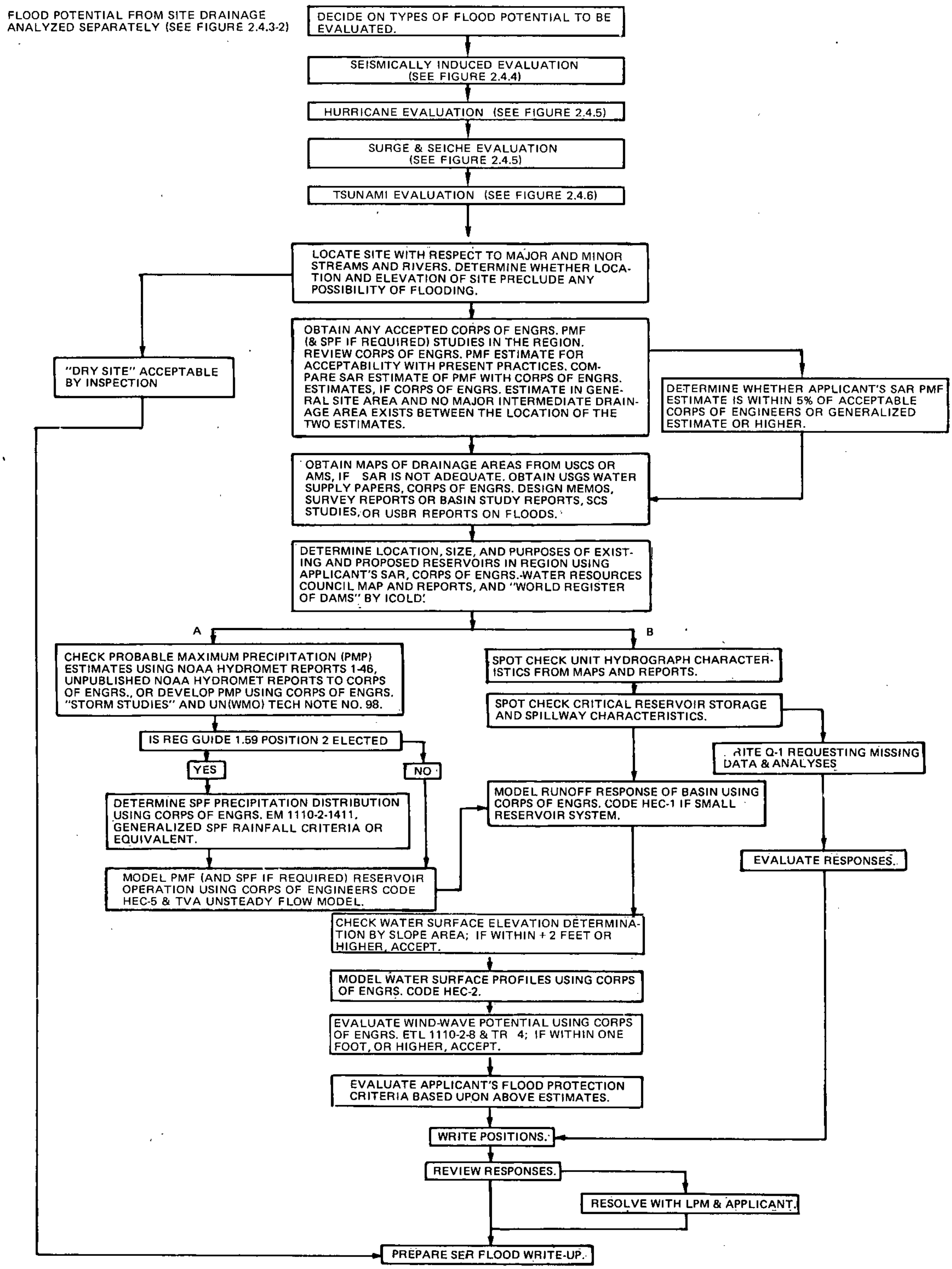
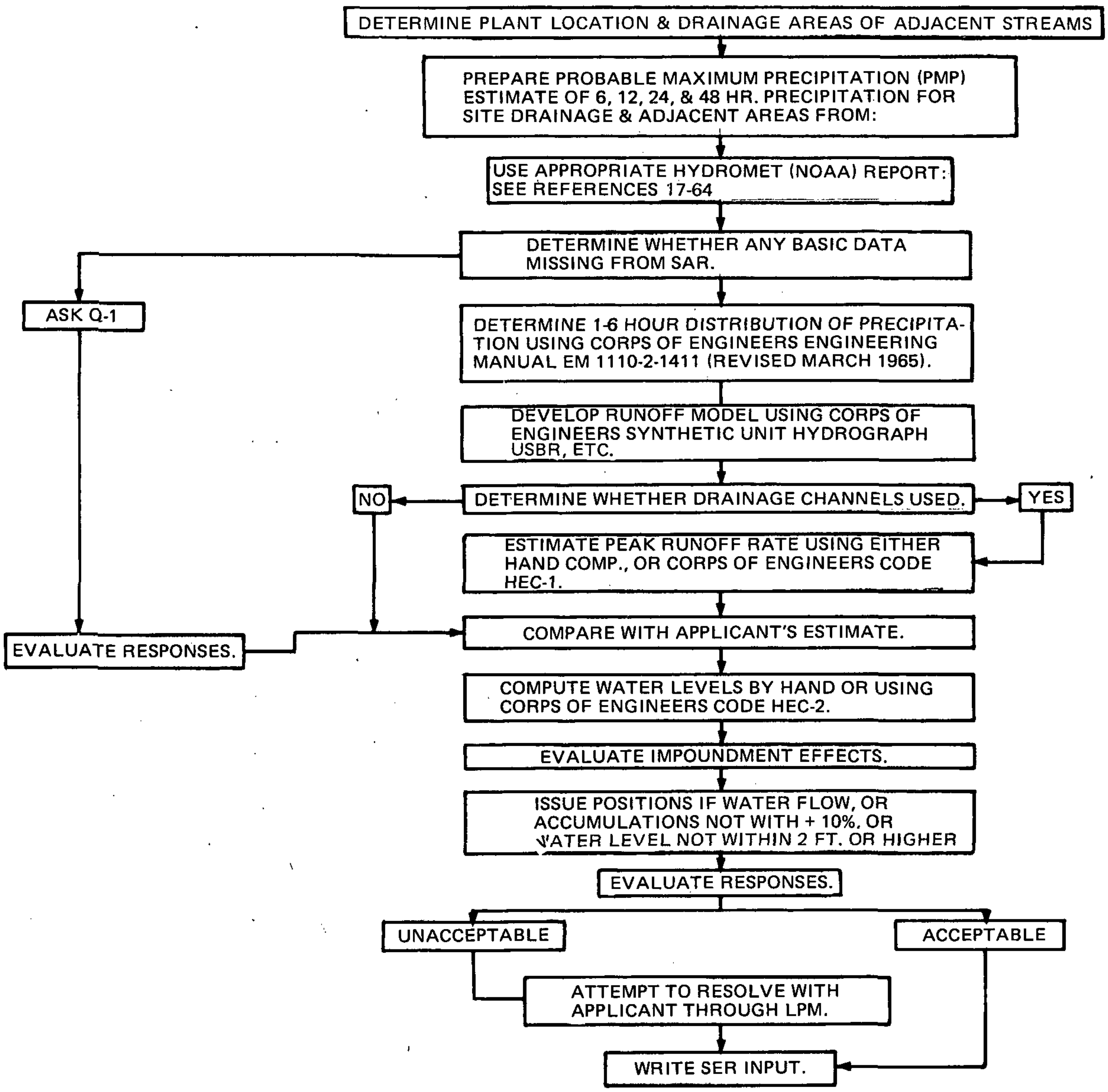


FIGURE 2.4.3-2

STANDARD REVIEW PLAN 2.4.3
SITE DRAINAGE & ADJACENT DRAINAGE



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SECTION 2.4.4 POTENTIAL DAM FAILURES (SEISMICALLY INDUCED)

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrogeologic design basis is developed to assure consideration in plant design of any potential hazard due to the failure of upstream and downstream water control structures from seismic causes. These hazards include flood waves (bores) from severe breaching of upstream dams and the potential loss of water supply due to failure of a downstream dam.

When data are provided to show that seismic events will not cause failures of upstream dams that could produce the governing flood at the plant, this section may contain additional data and other information to support a contention that the dams are equivalent to seismic Category I structures and will survive a local equivalent of the safe shutdown earthquake (SSE) or will survive the operating basis earthquake (OBE). In such cases the areas of review will include items necessary to justify such a classification. Such review would be referred to the SAB Geology, Seismology, and Foundation Engineering Section for evaluation. The balance of this review plan applies to non-Category I structures, and to the hydrologic analysis of those Category I structures that could be affected by flood waves caused by upstream failures.

Where analyses are provided in support of either a conclusion that a probable maximum flood (PMF) should be the design basis flood for a stream, or that a postulated or arbitrarily assumed seismically-induced flood is the design basis flood for a stream, the areas of review consist of the following:

1. Conservatism of modes of assumed dam failure and deposition of debris downstream.

2. Consideration of full flood control reservoirs.

3. Conservatism of downstream flow rates and levels depending on whether failure is postulated with an equivalent SSE coincident with a 25-year flood, or an OBE coincident with a flood approximately half as severe as a PMF.

4. Flood wave attenuation to downstream dams, or to the site, whichever would be encountered first.

5. Potential for multiple dam failures; flood wave effects and potential for failure of downstream dams.

6. Hydraulic failure of downstream dams induced by upstream failures.

7. Dynamic effects on exposed plant facilities of possible bores.

8. Conservatism (see item 3 above) of ambient flow conditions for downstream dam failures that can influence safety-related water supplies.

II. ACCEPTANCE CRITERIA

The staff will review the applicant's analyses and independently estimate the coincident river flows at the site and at the dams being analyzed (see Figure 2.4.4). The acceptable "worst conditions" to be postulated for analysis of upstream failures in lieu of substantiation of seismic resistance capability are: (1) a 25-year flood on a full reservoir coincident with the dam-site equivalent of the SSE, and (2) a standard project flood (a flood about half the severity of a PMF) on a full reservoir coincident with the dam site equivalent of the OBE.

For SAR Section 2.4.4.1 (Dam Failure Permutations): the location of dams and potentially "likely" or severe modes of failure must be identified. The potential for multiple, seismically-induced dam failures (of closely spaced dams) and the domino failure of a series of dams, including the resulting flood surge-caused failure of intermediate structures, must be discussed. First-time use of analytical hydraulic failure models will require complete model description and documentation. Acceptance of the model (and subsequent analyses) is based on the staff review of model theory, available verification, and application. A determination of the peak flow rate and water level at the site for the worst possible combination of dam failures and a summary analysis (that substantiates the condition as the critical permutation) must be presented, along with a description (and the bases) of all coefficients and methods used. Also, the effects of other concurrent events on plant safety, such as blockage of the river and waterborne missiles must be considered.

For SAR Section 2.4.4.2 (Unsteady Flow Analysis of Potential Dam Failures): the effects of coincident and antecedent flood flows (or low flows for downstream structures) on initial pool levels must be considered. Use of the methods given in References 1 or 3 is acceptable for determination of initial pool levels. Depending on the estimated failure mode, the "gradually varied unsteady flow profiles" program (Ref. 9) used by the Corps of Engineers or the Tennessee Valley Authority model (Ref. 8) may provide an acceptable analysis.

For SAR Section 2.4.4.3 (Water Level at Plant Site): computations, coefficients, and methods used to establish the water level at the site for the most critical dam failures must be summarized. Comparison with the HEC-2 program (Ref. 2) or unsteady flow models (Refs. 8 and 9) with adequate site-related coefficients, serves as a basis for acceptance. Coincident wind-generated wave activity should be considered in a manner similar to that discussed in Standard Review Plan 2.4.3.

III. REVIEW PROCEDURES

The review procedures are outlined in Figure 2.4.4. In general, the conservatism of the applicant's estimate of flood potential and low water levels from seismically-induced structure failures is judged against the criteria indicated above. When required, an analysis is performed using simplified, conservative procedures (such as instantaneous failure, coincident one-half PMF flows, minimal flood wave attenuation, and extrapolated site discharge-rating curves). Techniques for such analyses are identified in standard hydraulic design references and text books, such as those listed in the reference section. If no potential flood problem exists, the staff safety evaluation report (SER) input is written accordingly. If the simplified analysis indicates a potential flooding problem, the analysis is repeated using a more refined technique, and additional information and data are requested from the applicant if necessary. Detailed failure models, such as those of the Corps of Engineers and the Tennessee Valley Authority, are utilized to identify the outflows from various failure modes. Models of the Corps of Engineers or the Tennessee Valley Authority are used to identify the outflow characteristics and resultant water level at the site (Refs. 4, 8, 13, 14, 15).

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will consist of a statement comparing the applicant and staff evaluations of the design-basis maximum and minimum water levels caused by seismically-induced dam failures. If staff findings are similar to the applicant's, staff concurrence in the applicant's estimates will be stated. If the staff estimates substantially higher or lower water levels or flows, and if the plant may be adversely affected, a position requiring use of the staff bases will be stated. If no seismically-induced dam failure review was undertaken at the construction permit stage (of the scope described), this fact will be indicated.

For operating license (OL) reviews of cases for which detailed seismically-induced dam failure analyses were made during the CP review, the CP-stage conclusions will be referenced. In addition, any further review done to reaffirm the maximum or minimum water levels based on any new information will be described and the results and conclusions stated.

Sample statements for CP reviews follow:

"The distance (more than 300 miles) to upstream reservoirs of appreciable size is such that the staff considers their arbitrarily assumed failure, under AEC criteria of reasonably postulated combinations of floods and earthquakes, would not constitute a threat to the plant worse than that due to a severe runoff-type flood or to hurricane-induced surge.

"Dam failure-caused 'worst case' floods were evaluated by the applicant based upon failures with consideration of only the location and sizes of upstream impoundments, and not on inherent capability of such structures to resist earthquakes, volcanic activity and severe landslide-induced floods. The most severe flood of this kind was estimated based upon an assumed catastrophic failure of Dam A some 420 miles upstream. The peak flow at the site from such a flood was estimated to be 3,000,000 cfs. This flow is estimated to occur about two days after the dam failure and reach elevation 41 feet MSL. Smaller dams on the river between Dam A and the site were also evaluated for such a flood and, it was concluded, would probably also fail.

"A volcanically-induced flood was assumed to cause a domino-type failure of the three dams on the tributary B River from a volcanic eruption of Mt. D. The evaluation indicated such an event could cause the second most severe artificial flood that would reach the site. This event was estimated to produce a peak flow at the site of 3,300,000 cfs and a water level of 39 feet MSL."

V. REFERENCES

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2. "Water Surface Profiles," HEC-2, Corps of Engineers Hydrologic Engineering Center, Davis, California, February 1972.

3. "Reservoir System Operation for Flood Control," HEC-5, Corps of Engineers Hydrologic Engineering Center, Davis, California, May 1973.

4. "Routing of Floods Through River Channels," EM 1110-2-1408, Corps of Engineers, March 1960.

5. Hunter Rouse, ed., "Engineering Hydraulics," John Wiley & Sons, Inc., New York (1950).

6. Ven Te Chow, "Open-Channel Hydraulics," McGraw-Hill Book Co., New York (1959).

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8. J. M. Garrison, J. P. Granju, and J. T. Price, "Unsteady Flow Simulation in Rivers and Reservoirs," Jour. Hydraulics Division, Proc. Am. Soc. of Civil Engineers Vol. 95, No. HY5, pp. 1559-1576 (1969).

9. "Gradually Varied Unsteady Flow Profiles," 723-62-L2450, Corps of Engineers Hydrologic Engineering Center, Davis, California, March 1969.

10. R. A. Baltzer and C. Lai, "Computer Simulation of Unsteady Flows in Waterways," Hydraulics Division, Proc. Am. Soc. of Civil Engineers, Vol. 94, No. HY4, pp. 1083-1117 (1968).

11. J. J. Stoker, "Numerical Solution of Flood Prediction and River Regulation Problems," Reports I and II, New York Univ. (1953-54).

12. V. L. Streeter and E. B. Wylie, "Hydraulic Transients," McGraw-Hill Book Co., New York, pp. 239-259 (1967).

13. W. A. Thomas, "A Method for Analyzing Effects of Dam Failures in Design Studies," Corps of Engineers Hydrologic Engineering Center, Davis, California, (for presentation at the ASCE Hydraulics Division Specialty Conference, Cornell University, August 1972).

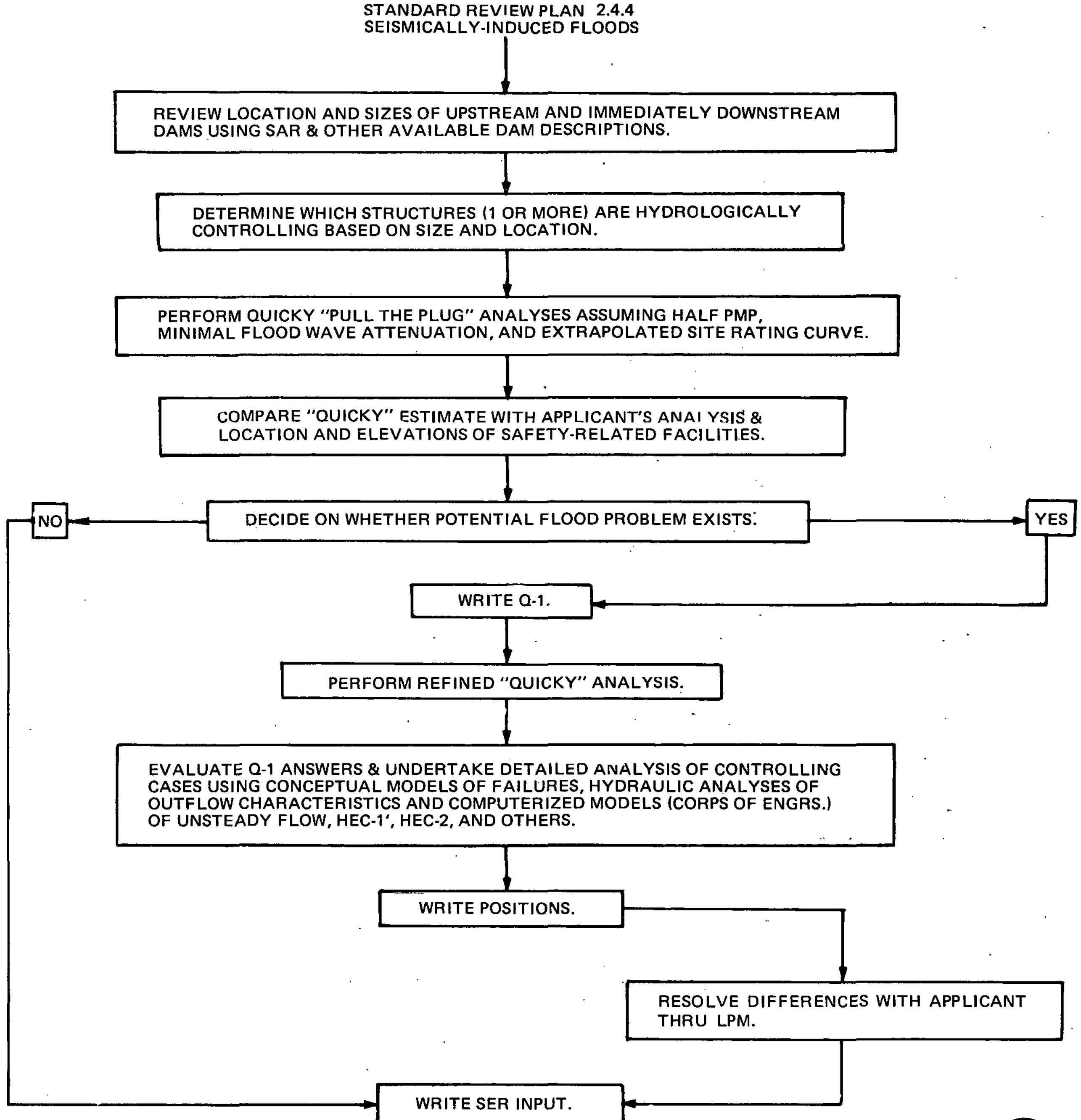
14. "Flow Through a Breached Dam," Military Hydrology Bulletin No. 9, Corps of Engineers (1957).

15. "Floods Resulting from Suddenly Breached Dams, Conditions of High Resistance," Misc. Paper No. 2-374, Report 2, Corps of Engineers (1961).

16. Bureau of Reclamation, "Flood Routing," Chapter 6/0 in "Flood Hydrology," Part 6 in "Water Studies," Volume IV, U. S. Department of the Interior (1947).

17. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

FIGURE 2.4.4


STANDARD REVIEW PLAN 2.4.4
SEISMICALLY-INDUCED FLOODS
REVIEW LOCATION AND SIZES OF UPSTREAM AND IMMEDIATELY DOWNSTREAM DAMS USING SAR & OTHER AVAILABLE DAM DESCRIPTIONS.
DETERMINE WHICH STRUCTURES (1 OR MORE) ARE HYDROLOGICALLY CONTROLLING BASED ON SIZE AND LOCATION.
PERFORM QUICKY "PULL THE PLUG" ANALYSES ASSUMING HALF PMP, MINIMAL FLOOD WAVE ATTENUATION, AND EXTRAPOLATED SITE RATING CURVE.
COMPARE "QUICKY" ESTIMATE WITH APPLICANT'S ANALYSIS & LOCATION AND ELEVATIONS OF SAFETY-RELATED FACILITIES.
DECIDE ON WHETHER POTENTIAL FLOOD PROBLEM EXISTS.
NO
YES
WRITE Q-1.
PERFORM REFINED "QUICKY" ANALYSIS.
EVALUATE Q-1 ANSWERS & UNDERTAKE DETAILED ANALYSIS OF CONTROLLING CASES USING CONCEPTUAL MODELS OF FAILURES, HYDRAULIC ANALYSES OF OUTFLOW CHARACTERISTICS AND COMPUTERIZED MODELS (CORPS OF ENGRS.) OF UNSTEADY FLOW, HEC-1', HEC-2, AND OTHERS.
WRITE POSITIONS.
RESOLVE DIFFERENCES WITH APPLICANT THRU LPM.
WRITE SER INPUT.

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SECTION 2.4.5 PROBABLE MAXIMUM SURGE AND SEICHE FLOODING

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

In this section of the safety analysis report (SAR) the hydrometeorological design basis is developed to determine the extent of flood protection required for safety-related plant systems. The areas of review include the probable maximum hurricane or other probable maximum wind storms, antecedent water levels, storm tracks, methods of analysis, coincident wind-generated wave action and wave runup on safety-related structures, potential for wave oscillation at the natural periodicity, and the resultant design bases for surge and seiche flooding.

II. ACCEPTANCE CRITERIA

Hydrometeorological estimates and criteria for development of probable maximum hurricanes for east and Gulf coast sites, squall lines for the Great Lakes, and severe cyclonic wind storms for all lake sites by the Corps of Engineers, National Oceanic and Atmospheric Administration (NOAA), and the staff are used as standards for evaluating the conservatism of the applicant's estimates of severe windstorm conditions, as discussed in Regulatory Guide 1.59. The Corps of Engineers and NOAA criteria require variation of the basic meterological parameters within given limits to determine the most severe combination that could result. The applicant's estimates should be at least as conservative as the most critical combination of these parameters.

Data from publications of NOAA, the Corps of Engineers, and other sources (such as tide tables, tide records, and historical lake level records) are used to substantiate antecedent water levels. These antecedent water levels must be as high as the "10 percent exceedence" monthly spring high tide plus a sea level anomaly based on the maximum difference between recorded and predicted average water levels for durations of two weeks or longer for coastal locations or the average monthly recorded high water for the Great Lakes. In a similar manner, the storm track, wind fields, effective fetch lengths, direction of approach, and frictional surface and bottom effects are evaluated by independent staff analysis to assure that the most critical values have been selected. Models used to estimate surge hydrographs that have not previously been reviewed and approved by the staff are verified by reproducing historical events, with any discrepancies in the model being on the conservative (i.e., high) side.

Criteria and methods of the Corps of Engineers as generally summarized in Reference 30 are used as a standard to evaluate the applicant's estimate of coincident wind-generated wave action and runup.

Criteria and methods of the Corps of Engineers and other standard techniques are used to evaluate the potential for oscillation of waves at natural periodicity.

Criteria and methods of the Corps of Engineers (Ref. 30) are used to evaluate the adequacy of protection from flooding, including the static and dynamic effects of broken, breaking, and nonbreaking waves.

The analysis will be considered complete and acceptable if the following areas are addressed and can be independently and comparably evaluated from the applicant's submission (the following presumes that it has been determined that surge and seiche flooding estimates are necessary to identify flood design bases):

1. All reasonable combinations of probable maximum hurricane, moving squall line, or other cyclonic wind storm parameters are investigated, and the most critical combination is selected for use in estimating a water level.

2. Models used in the evaluation are verified, or have been previously approved.

3. Detailed descriptions of bottom profiles are provided (or are readily obtainable) to enable an independent estimate of surge levels to be made.

4. Detailed descriptions of shoreline protection and safety-related facilities are provided to enable an independent estimate of wind-generated waves, runup, and potential erosion to be made.

5. Ambient water levels, including tides and sea level anomalies, are estimated as described above.

6. Combinations of surge levels and waves that may be critical to plant design are considered, and adequate information is supplied to allow a determination that no adverse combinations have been omitted.

7. If Regulatory Guide 1.59, Position 2, is elected by the applicant, the design basis for flood protection of all safety-related facilities identified in Regulatory Guide 1.29 must be shown to be adequate in terms of time required for implementation of any emergency procedures. The applicant must also demonstrate that the less severe design basis

selected will provide for all potential flood situations that could negate the time and capability to initiate flood emergency procedures.

In general, the staff will make an independent estimate of surge, seiche, and wave action effects (static and dynamic). If the estimated effects are comparable with those of the applicant, or if the applicant's estimates are greater, the proposed design basis will be considered confirmed.

III. REVIEW PROCEDURES

The review procedure is outlined on Figure 2.4.5. In general, the conservatism of the applicant's estimate of flood potential from surges and seiches is judged against the criteria indicated above and as discussed in Regulatory Guide 1.59. If the site is not near a large body of water the staff findings may be prepared a priori. Methods of the Corps of Engineers and National Oceanic and Atmospheric Administration (NOAA) (HUR 7-97 and amendments) are used to develop the critical probable maximum hurricane (PMH) parameters for the site. The Corps of Engineers model SURGE (or other verified models) may be used to estimate the maximum surge stillwater elevations at coastal sites. Coincident wind-generated waves and runup are estimated from publications by the Corps of Engineers (Ref. 30). Reports of NOAA and the Corps of Engineers are used to estimate probable maximum wind fields over the Great Lakes. Models such as Platzmann's, or other verified models, are used to estimate the maximum surge or seiche stillwater elevation for Great Lakes sites; coincident wind-generated waves and runup are estimated as above.

Seiching potential is evaluated by comparing the natural period of oscillation (resonance) of the water body with the estimated meteorologically-induced wave periods. Resonance of a water body may be calculated by the methods presented in Ref. 30, or standard texts. Generally, a demonstration that the water body cannot generate or sustain waves of the required period (for resonance) is satisfactory to conclude that atmospheric pressure and wind-generated wave amplification is not possible. If resonance is possible, the maximum seiche must be considered in the selection of the critical flood design bases.

Consultants may be employed by the staff in either an advisory role on specific aspects of the analysis, or to make a separate independent analysis, depending upon the complexity of the analysis and available staff manpower. The consultants may be from the Corps of Engineers Coastal Engineering Research Center (CERC) or private contractors.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement summarizing the applicant and staff estimates of critical water level (including wind-generated wave levels) at the site. If the estimates are similar, staff concurrence will be stated. If the staff predicts substantially higher water levels, and the proposed plant may be adversely affected, a statement requiring use of the staff estimate for the design basis will be made. If the flood conditions do not constitute a design basis, the statement will so indicate.

For operating license (OL) reviews of plants which have received detailed reviews during the CP review, the CP conclusions will be referenced. However, a review will be made to assure that protection against the design-basis water level conditions established in the CP review has been properly implemented. In addition, a review of surge and seiche history since the CP review will be made. Any new information or improvements in predictive models will be noted. If no detailed CP review was undertaken, this fact will be indicated in the OL findings.

If Regulatory Guide 1.59, Position 2, is elected by the applicant for protection, a statement describing lesser design bases will be included in the findings with the staff conclusion of adequacy.

A sample statement for an OL review follows:

> "The design basis hurricane-induced high and low stillwater levels were established during the CP review at elevations 22.0 feet MSL and -7.5 feet MSL, respectively. These levels are based upon the estimated water levels, exclusive of wave action, that would occur during passages of a probable maximum hurricane (PMH)[1/] to the south and north, respectively, of the plant. At the request of the staff, the applicant analyzed the wave conditions on safety-related facilities that could accompany the 22 foot MSL surge level. The results of these analyses indicate the most severe wave action would be restricted to the canal, and that high ground levels would limit wave heights in the vicinity of exposed safety-related buildings, except the service water intake, to 1.6 feet. For the intake, the applicant has estimated waves 3 feet high. The resulting wave runup levels were estimated to reach a maximum elevation of 28.3 feet MSL on the intake, and 25.6 feet MSL on other exposed buildings."

"[1/]A PMH is considered to be the worst hurricane reasonably possible of occurrence."

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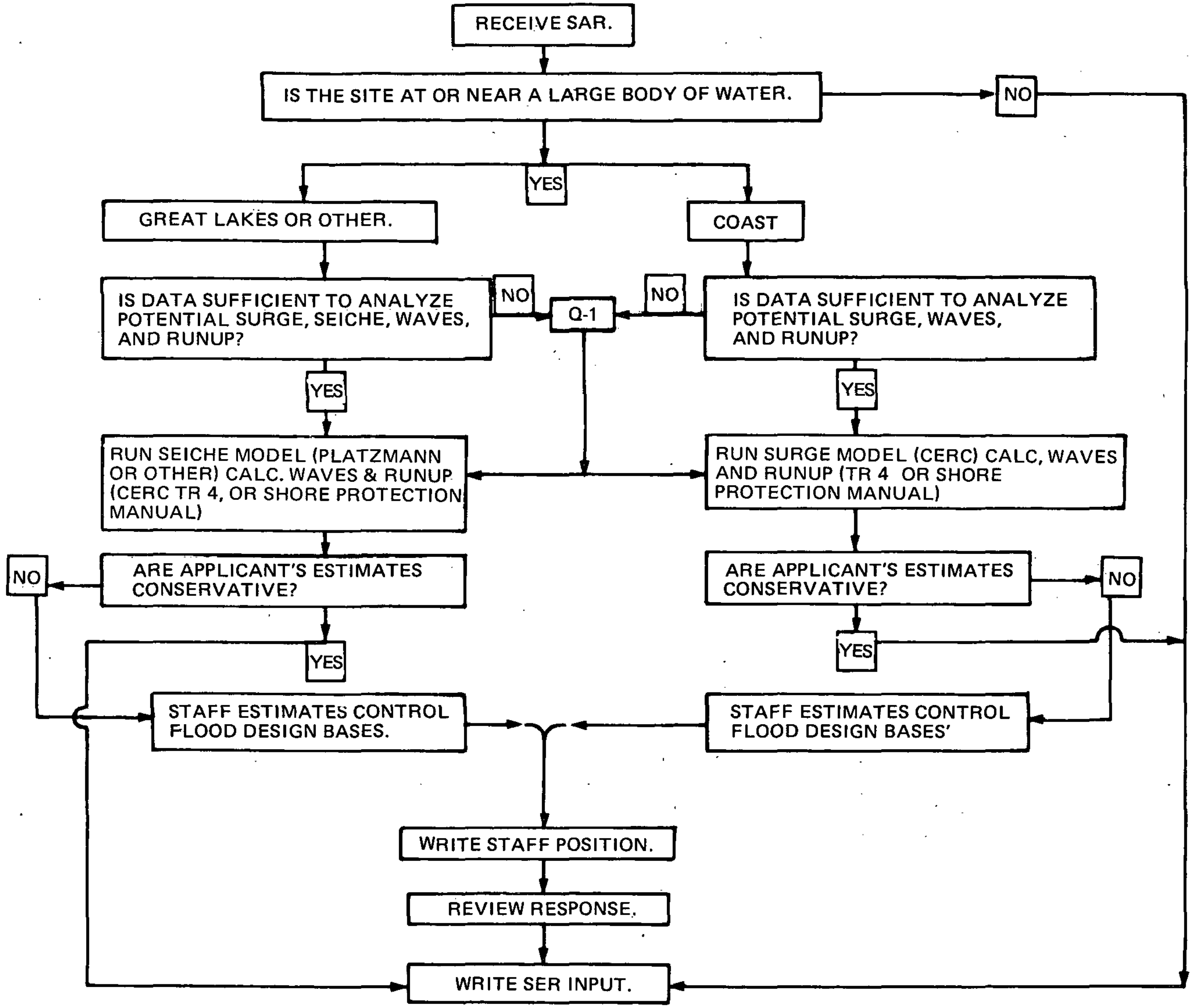
FIGURE 2.4.5

STANDARD REVIEW PLAN 2.4.5
PROBABLE MAXIMUM SURGE AND SEICHE FLOODING



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SECTION 2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The geohydrological design basis of the plant is developed in this section of the safety analysis report (SAR) to determine the extent of plant protection required for tsunami flooding, if any. The areas of review consist of (1) identifying the most severe distant and local sources of tsunami generation, and (2) identifying the maximum magnitudes, focal depths, vertical displacement components, and periodicities of seismic activity for these most severe distant and local potential generators. Based on the results of (1) and (2), sea disturbances at those locations of maximum tsunami wave train propagation potential with respect to the plant site are modeled by postulating various initiating sea wave train forms at the generator locations, and analytically propagating the initiating wave train to the site for determination of the worst case. Coincident astronomical tide, storm surge or sea level anomaly, and storm waves (all of approximately annual severity) are postulated coincident with the tsunami. Tsunami wave runup and runout (drawdown), including superposition of the effects of coincident water level and wave action, are evaluated for each safety-related structure that may be affected. Design criteria for structures provided to protect safety-related facilities, such as seawalls and breakwaters, are reviewed for seismic design basis and, separately, for tsunami wave train design basis. Predictive deterministic models used in making probable maximum tsunami wave train propagation estimates are compared with historical events for model verification. Nearshore wave propagation is analyzed for wave form changes due to local hydrography and harbor or breakwater influences, including resonance (wave amplification) effects.

II. ACCEPTANCE CRITERIA

The general criteria to be used in estimating tsunami static and dynamic effects are contained in Regulatory Guide 1.59, which is based, in part, upon General Design Criteria 2, Appendix A to 10 CFR Part 50. The analysis will be considered complete and acceptable if (a) reasonably severe seismic generating mechanisms of safe shutdown earthquake (SSE) severity have been postulated for the known most severe local and distant generator locations, (b) wave train propagation accounts for bathymetric influences, (c) coincident ambient tide and wave conditions of about annual severity have been assumed, and (d) resonance effects are considered. In general, the staff will make an independent estimate of the effects of at least one controlling tsunami wave train case. If the applicant's water level estimates, referenced to mean or normal levels, are within about 5-10 percent of those made by the staff, or are greater, the applicant's proposed design basis will be considered confirmed.

III. REVIEW PROCEDURES

The review procedures, as described on the attached Figure 2.4.6, consist of evaluating the potential for maximum tsunami generation from both distant and local sources, and may employ the use of analytical models of the generating mechanisms, a range of generating wave forms, and wave propagation analyses to the edge of the continental shelf and to safety-related near-shore facilities. The references used are general geophysical, seismological, and hydrodynamic publications, such as published data by the National Oceanic and Atmospheric Administration (NOAA), and wave propagation models such as those developed by NOAA and Tetra Tech. Tsunami analyses, with respect to both generator identification and evaluation and wave generation models, are currently in a state of flux. The reviewer and applicant must stay abreast of the availability of the rapidly developing analytic techniques which apply to each site and region.

Because tsunami estimates are site-specific, each application requires identification of the physical parameters and data associated with the specific site in question on an individual basis.

Consultants may be employed in either an advisory role in developing independent staff tsunami effect estimates, or in making independent estimates of specific effects, depending on the complexity of the analysis required and available staff manpower. The consultants may be from the NOAA Tsunami Research Center, the Corps of Engineers Coastal Engineering Research Center (CERC), or private contractors.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement summarizing applicant and staff estimates of the maximum and minimum tsunami water levels, static and dynamic effects of wave action, and a statement of acceptability of the tsunami-induced design basis. For operating license (OL) reviews, the findings will consist of the evaluation of any new information on tsunami potential, improvements in predictive models acceptability of specific design bases, and the acceptability of design provisions.

A sample statement for an CP review follows:

> "Floods caused by wind-driven surges up the tidal portion of the estuary between the Pacific and the site, and seismically-induced sea waves (tsunamis) being propagated up the A River estuary were also evaluated by the applicant. Both

phenomena are believed to have historically caused water level variations in the site vicinity. The applicants' analysis, however, indicates the physical characteristics of the A River estuary between the site and the Pacific tend to limit water level effects at the site to only relatively small water level excursions."

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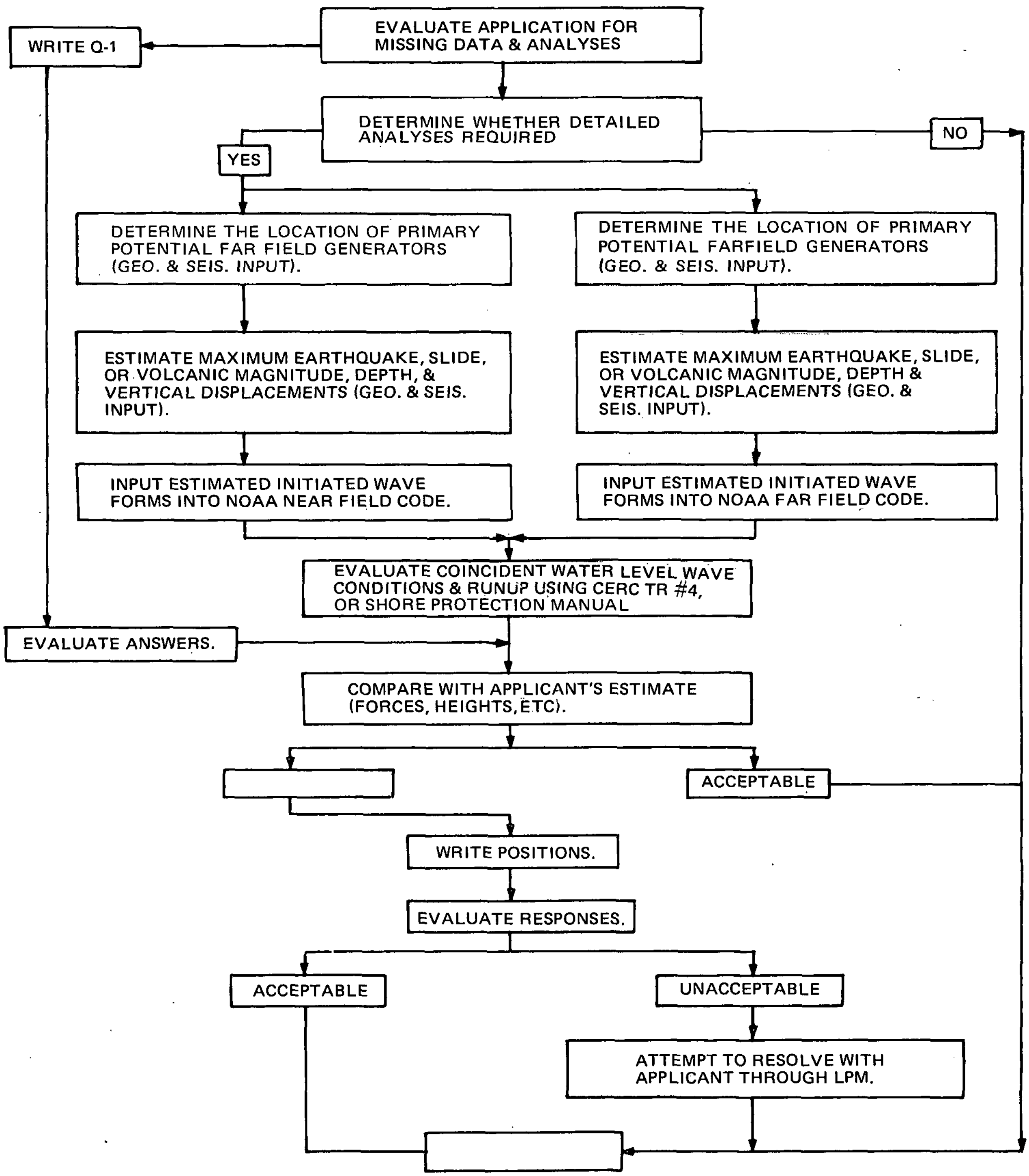
FIGURE 2.4.6

STANDARD REVIEW PLAN 2.4.6—TSUNAMIS



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SECTION 2.4.7 ICE EFFECTS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branh (SAB)

Secondary - None

I. AREAS OF REVIEW

The hydrometeorologic design basis is developed in this section of the safety analysis report (SAR) to assure that safety-related facilities and water supply are not affected by ice flooding or blockage. The areas of review include:

1. The regional history and types of historical ice accumulations (i.e., ice jams, wind-driven ice ridges, floes, etc.).
2. The potential for ice-produced forces on, or blockage of, safety-related facilities.
3. The potential effects of ice-induced high or low flow levels on safety-related facilities and water supplies.

II. ACCEPTANCE CRITERIA

Publications of the National Oceanic and Atmospheric Administration (NOAA), the United States Geologic Survey (USGS), the Corps of Engineers, and other sources are used to identify the history and potential for ice formation in the region. Historical maximum depths of icing should be noted, as well as mass and velocity of any large floating ice bodies. The phrase "historical low water ice affected," or similar phrases in streamflow records (USGS and state publications) will alert the reviewer to the potential for ice effects. The following items must be considered and evaluated, if found necessary, in the design of protection of safety-related facilities and water supplies.

1. The regional ice and ice jam formation history must be described to enable an independent determination of the need for including ice effects in the design basis.
2. If icing has not been severe, based on regional icing history, design considerations must be presented (e.g., return of a portion of low-grade heat to the intake) to assure that icing or ice blockage of intake screens and pumps will not adversely affect safety-related facilities and water supplies.
3. If the potential for icing is severe, based on regional icing history, it must be shown that water supplies capable of meeting safety-related requirements are available from under the ice formations postulated and that safety-related equipment is protected from

icing as in 2., above. If not, it must be demonstrated that alternate sources of water are available, that they are protected from freezing, and that the alternate source is capable of meeting safety-related requirements in such situations. Ice loading must have been included in the structural design basis, if severe icing is possible.

4. If floating ice is prevalent, based on regional icing history, consideration of impact forces on the safety-related intakes must be a consideration in the design basis. The dynamic loading caused by floating ice must be included in the structural design basis.
5. If ice blockage of the river or estuary is possible, it must be demonstrated that the resulting water level in the vicinity of the site has been considered in establishing the flood and water supply design bases. If this water level would adversely affect the intake structure, or other safety-related facilities, it must be demonstrated that an alternate safety-related water supply will not also be adversely affected.

III. REVIEW PROCEDURES

Applicable literature describing historical occurrences of icing in the region is reviewed to determine if icing protection should be considered in the design of safety-related facilities. If so, the most likely types of icing conditions (floating ice, river blockage by ice buildup, frazil, etc.) are listed, and the impact on plant design of each type is identified. Criteria of the Corps of Engineers and others provide a means of assessing icing impact and methods of mitigating adverse effects. For each type of icing condition, independent estimates of the "worst case" will be made by either statistical or deterministic techniques. Evidence, if any, of potential structural effects will be furnished the Structural Engineering Branch (SEB); similarly, mechanical impairment potential will be furnished the Auxiliary and Power Conversion Systems Branch (A&PCSB) or the Mechanical Engineering Branch (MEB).

The above reviews are performed only when applicable to the site or site regions. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will consist of a statement of the applicant and staff estimates of the potential for ice flooding. If applicable, the minimum low water levels (from upstream ice blockage) will be included. If the estimates are similar, staff concurrence with the applicant's estimate will be stated. If the staff predicts substantially higher or lower controlling water levels, or blockage of the intake, and if the proposed plant may be adversely affected, a statement of the staff bases will be made. If the icing conditions do not constitute a design basis, the statement will so indicate.

For operating license (OL) reviews of plants for which detailed icing reviews were made at the CP stage, the CP conclusions will be referenced. However, a review will be made to assure that the design basis established in the CP review has been implemented properly. In addition, a review of icing records since the CP review will be made. If no CP review was undertaken (of the scope indicated), this fact will be noted in the OL findings.

A sample CP statement follows:

"Ice Flooding, which is common on the A River at the makeup intake structure, could only affect the river intake structure which would not result in any adverse effects to the plant's safety-related facilities. The applicant states that ice flooding may possibly raise the water surface near the A River intake to a maximum elevation of about 555 feet MSL. The applicant further states that ice and ice flooding on the A River tributaries outside the cooling lake will not affect the plant facilities. The major tributary nearest the plant is the B Creek with the closest point located about one mile to the southeast of the site. The applicant concludes that, because of the distance from the proposed site and the wide floodplain of the river, there will be no adverse effects at the plant site due to ice in the river and consequent flooding. We concur with this conclusion.

"The safety-related pumps from the cooling lake are to be protected from ice blockage by means of traveling screens, stop logs, and trash racks located at the front of the lake screenhouse. In addition, the applicant proposes a warm-up line from the circulating water discharge which will keep the inlet water temperature 40° F. during winter operation. An essential cooling water screen bypass pipe is also available. We concur with the applicant that icing or ice flooding should not adversely affect the plant's safety-related facilities."

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SECTION 2.4.8 COOLING WATER CANALS AND RESERVOIRS

REVIEW RESPONSIBILITY

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

This section of the applicant's safety analysis report (SAR) presents the basis for the hydraulic design of canals and reservoirs used to transport and impound plant cooling water. In addition, the hydraulic design basis for protection of structures (e.g., riprap) is reviewed. For canals, the review covers the design basis for capacity, protection against wind waves, erosion, and freeboard, and (where applicable) the ability to withstand a probable maximum flood (PMF), surges, etc. For reservoirs, the areas of review include the design basis for capacity, probable maximum flood design basis, wind wave and runup protection, discharge facilities (low level outlet, spillway, etc.), outlet protection, and freeboard.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the protection of cooling water canals from wind waves, PMF, surges, etc., are the same as those outlined in Standard Review Plans 2.4.3, 2.4.4, 2.4.5, and 2.4.7. The criterion for canal capacity is that the canal must be capable of transmitting to the plant sufficient water to meet all safety requirements during postulated extreme hydrologic events (i.e., both floods and droughts). Where canals comprise a part of the ultimate heat sink, Regulatory Guide 1.27 is used as a basis for the adequacy of design criteria and provisions. The design basis for canal capacity is analyzed, in any case, to assure that safety-related water requirements can be supplied under all reasonably severe conditions, or that alternative conveyance systems are designed to be available during the postulated conditions. The potential need for Technical Specifications to limit plant operation if normal plant water requirements may be adversely affected by extreme hydrologic phenomena is determined. Techniques developed by the Bureau of Reclamation (USBR) and Corps of Engineers are used to analyze the hydraulic design.

The acceptance criteria for the hydraulic design of reservoirs are as follows:

1. For protection of structures against wind waves, input from SAR Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7 for PMF, probable maximum hurricane (PMH), surge, seiche, or tsunami levels and coincident waves and runup must be considered to establish the maximum and minimum water level and wave conditions. Also, normal pool level and coincident probable maximum wind-wave activity must be considered. Criteria and methods as reported in Corps of Engineers publications are generally acceptable for design of embankment protection (riprap, grass, soil cement, tetrapods, dolosse, etc.) and freeboard.

2. For emergency storage evacuation, the spillways are acceptable if they can safely pass the PMF, or controlling design basis flood, without endangering safety-related facilities or increasing the hazard to downstream residents. In addition, a low level outlet should be provided to evacuate the storage in an emergency.

3. For reservoir routings, the maximum still water level is acceptable if the spillway design flood has been routed through the spillway (and outlet works, if applicable) using standard methods as suggested by the Corps of Engineers, USBR, and others, and a minimum of three feet of freeboard (including waves) is available. However, the antecedent reservoir level to be used with the flood routing must be at least as high as that suggested by Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

   The probable minimum low water level is acceptable if the flow during the design basis drought (from SAR Section 2.4.11) has been routed through the reservoir[1/] using standard methods as suggested by the Corps of Engineers, USBR, and others. The antecedent reservoir level for this routing, if reservoir storage is the sole water supply source, must be the lowest reasonably possible, considering regional conditions at the beginning of the drought and water demands, including plant requirements. In no case should the antecedent reservoir level be greater than the established normal operating level.

4. Where not covered above, the hydraulic design for low level outlets, conduits, spillways (gated and ungated, regulating and emergency), and embankment protection is required where the failure of such items could constitute a threat to essential plant facilities or to safety-related water supplies. The design is acceptable if standard techniques have been used as suggested by the Corps of Engineers, USBR, and others such that the minimum design water level for safety-related pumps would not be violated.

III. REVIEW PROCEDURES

In general, the conservatism of the applicant's design basis is judged against the criteria indicated above. SAR Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.7 should provide the basic data for analyzing the high flow hydraulic design basis of the facility. The applicant's hydraulic design basis is judged against standard design practices discussed in Corps of

[1/] For those plants proposing multiple reservoirs for water supply, analyses must be provided to assure that storage allocated for safety-related water supply in alternate reservoirs will be available during postulated drought conditions. Additionally, evidence of the right to use the water consumptively must be documented.

Engineers (Waterway Experiment Station) or USBR publications. Low flow input data are taken from SAR Section 2.4.11. The review procedure consists of independently "designing" (hydrologically and hydraulically), when necessary, the applicant's facilities (e.g., dams, canals, spillways) using the above methods and comparing the resultant "design" with the applicant's. Wave and runup protection is evaluated using the methods of References 20 and 21.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement of the applicant and staff estimates of the type and adequacy of required structure protection and the hydraulic design basis of canals and reservoirs. Because of the advanced design required for the CP and where the design has received a detailed review at the CP stage, the operating license (OL) findings will only be an acknowledgement of any changes and a statement of acceptability. If a design or flooding potential was not reviewed in detail at the CP stage, it will be done at the OL stage.

Sample statements from CP reviews follow:

"Although postulated flood waters are not expected to reach plant grade, protection of the essential auxiliary and main dams against their respective probable maximum floods is to be provided by riprap protection of exposed embankment surfaces (including areas in the plant site vicinity along the auxiliary reservoir intake channel) and concrete overflow spillways. At our request, the applicant provided design bases for riprap protection and the hydraulic design criteria for the two spillways. The applicant at our request, in Amendment No. 31 to the PSAR, provides criteria for the windwave riprap protection based upon an empirical relationship for the median size stone to be placed in a blanket approximately two feet thick and indicated its specifications for stone gradation. A filter blanket approximately one foot thick is to be placed under the riprap to prevent piping (removal of smaller material) through the larger armor riprap cover layer. Criteria were provided for the filter gradation, angularity, durability of the riprap, and placement which provides assurance that erosive failure of safety-related embankments should not occur. An armor protection layer also is provided. We find these riprap design bases and spillway hydraulic design criteria to be acceptable.

"Storage in the three reservoir system, runoff from the contributing drainage area, and diversion of A River flows to the main reservoir during periods of low runoff and high reservoir evaporation, will constitute the water supply for the four unit once-through cooling systems.

"The applicant has provided analyses of the capability of the main and auxiliary reservoirs to supply water during emergency conditions requiring emergency shutdown and cooldown of one unit and the simultaneous normal shutdown and cooldown of the remaining three units as suggested in Regulatory Guide 1.27 - Ultimate Heat Sink. In addition, the applicant has provided analyses of the operation of the plant and the main reservoir under historical and a synthesized 100-year drought conditions. For the shutdown conditions the applicant has demonstrated that the two reservoir - A River diversion system constituting the ultimate heat sink would have a water supply available in excess of thirty days in the auxiliary reservoir if water were not available from the main reservoir - auxiliary reservoir - A River diversion facilities. The operation of the sink as a whole will require that the auxiliary reservoir be kept at its normal operating level of elevation 250 feet MSL at all times by pumping water from the main reservoir to make up for water lost to normal evaporation.

"For the analyses of evaporation under normal plant operation during periods of assumed reoccurrence of historical droughts, the applicant has used historical flow records for the A River and synthesized flow data for the drainage area contiguous to the reservoir system. For the analysis of evaporation during a more extreme drought than has occurred historically, the applicant has synthesized flows from both the A River and the contiguous drainage areas for what is called a 100-year frequency drought. The staff, in consonance with our consultant (the U.S. Geological Survey), independently developed and analyzed synthesized flows from both drainage areas. We concluded that it is likely that flows from both areas could be substantially less than estimated by the applicant. The applicant is installing a streamflow gage near the plant to determine runoff characteristics from the contiguous drainage which should allow more accurate analysis of the operating capability of the reservoir system prior to plant operation. Inaccuracies in estimation of runoff are considered to be only indirectly safety related since an adequate shutdown and cooldown water supply will be available in the auxiliary reservoir should evaporation and the lack of runoff prevent replenishment of main reservoir storage above the minimum operating level of elevation 244 feet MSL."

V. REFERENCES

1. Am. Soc. Civil Engineers "Hydraulic Models," Manual of Engineering Practice No. 25 (1963).

2. Leo R. Beard, "Flood Control Operation of Reservoirs," Jour. Hydraulics Division, Proc. Am. Soc. Civil Engineers, Vol. 88, No. HYI, pp. 1-25 (1963).

3. Leo R. Beard, "Methods for Determination of Safe Yield and Compensation Water from Storage," Seventh International Water Supply Conference, Barcelona, Spain (1966).

4. E. F. Brater and H. W. King, "Handbook of Hydraulics for the Solution of Hydrostatic and Fluid-Flow Problems," McGraw-Hill Book Company, New York (1963).

5. V. T. Chow (ed), "Handbook of Applied Hydrology," McGraw-Hill Book Company, New York (1964).

6. V. T. Chow (ed), "Open Channel Hydraulics," McGraw-Hill Book Company, New York (1959).

7. C. V. Davis (ed), "Handbook of Applied Hydraulics," McGraw-Hill Book Company, New York (1964).

8. G. W. Fair, J. C. Geyer, and D. A. Okien, "Water Supply and Waste Water Removal," John Wiley & Son, Inc., New York (1966).

9. G. A. Hathaway, "Determination of Spillway Requirements for High Dams," Proc. Fourth International Conference on Large Dams, New Delhi, Vol. 2, pp. 301-347 (1951).

10. H. W. King and E. F. Brater, "Handbook of Hydraulics," McGraw-Hill Book Company, New York (1963).

11. R. K. Linsley and J. B. Franzini, "Water-Resources Engineering," McGraw-Hill Book Company, New York (1964).

12. H. Rouse (ed), "Engineering Hydraulics," John Wiley & Son, Inc., New York (1951).

13. "Hydraulic Design Criteria," prepared by the Corps of Engineers Waterways Experiment Station, loose-leaf by serials.

14. "Hydraulic Design of Flood Control Channels," Engineer Manual 1110-2-1601, Corps of Engineers, July 1970.

15. "Hydraulic Design of Spillways," Engineer Manual 1110-2-1603, Corps of Engineers, March 1965.

16. "Hydraulic Tables," Corps of Engineers (1944).

17. "Hydrologic Engineering Methods for Water Resources Development," Volumes 1 through 12, Corps of Engineers Hydrologic Engineering Center, Davis, California (1971).

18. "Reservoir Regulation," Engineer Manual 1110-2-3600, Corps of Engineers, May 1959.

19. "Reservoir Storage-Yield Procedures," Corps of Engineers Hydrologic Engineering Center (1967).

20. "Shore Protection Manual," Technical Report No. 4, Third Edition, Corps of Engineers Coastal Engineering Research Center (1966).

21. "Shore Protection Manual," Corps of Engineers Coastal Engineering Research Center (1973).

22. Hydraulic Model Studies of the Corps of Engineers Waterways Experiment Station.[2/]

23. "Design of Small Dams," Second Edition, Bureau of Reclamation, U.S. Dept. of the Interior (1973).

24. "Design Standards No. 3, Canals and Related Structures," Chapter 2 of "General Design Information for Structures," Bureau of Reclamation, U.S. Dept. of the Interior, April 1962.

25. "Hydraulic Model Studies"[2/] of the Bureau of Reclamation, U.S. Dept. of the Interior.

26. "Hydraulic Model Studies"[2/] of the Dept. of Water Resources, State of California.

27. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

[2/] A series of such studies exists in the literature too numerous to mention here. In addition to the three specifically cited series studies by others will be utilized on an "as available" basis.

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SECTION 2.4.9 CHANNEL DIVERSIONS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

In this section of the applicant's safety analysis report (SAR) the geohydrologic design basis is developed to assure that the plant and essential water supplies will not be adversely affected by natural stream channel diversion, or that in such an event, alternate water supplies are available to safety-related equipment.

The review includes:

1. Historical channel diversions, including cutoffs and subsidence.
2. Regional topographic evidence which suggests that future channel diversion may or may not occur (used in conjunction with evidence of historical diversions).
3. Alternate water sources and operating procedures (coordinate review with that of SAR Section 2.4.11.6).

II. ACCEPTANCE CRITERIA

The analyses will be considered acceptable if at least the following are addressed:

1. A description of the applicability (potential adverse effects) of stream channel diversions.
2. Historical diversions and realignments.
3. The topography and geology of the basin and its applicability to natural stream channel diversions.
4. If applicable, the safety consequences of diversion and the potential for high or low water levels caused by upstream or downstream diversion adversely to affect safety-related facilities or water supply.

III. REVIEW PROCEDURES

Site-specific publications and maps are reviewed to identify historical channel diversions and evaluate (by independent conservative calculations and professional judgement) the potential for future diversions. Where an alternate safety-related cooling water supply is provided, the criteria for SAR Section 2.4.11.6 apply and are checked for consistency.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews and when applicable, findings will consist of a brief general description of historical channel diversions. If the staff concurs with the applicant that channel diversion is unlikely or that the plant is protected and alternate essential water supplies meet the criteria of Regulatory Guide 1.27, the findings will so indicate. If the staff evaluation does not support the applicant's contention of channel stability, an alternate source of water may be required.

For operating license reviews, findings will consist of the same material, updated as required to reflect new information available since preparation of the CP findings.

A sample CP-stage statement follows:

> "Diversions of the A River are well-documented in historical and topographic data. Oxbow lakes, low-lying swamps, sand bars, and chutes provide eloquent evidence of historical diversion. Others are planning a further bank protection measures, additional to the existing levee system, in the vicinity of the plant intake structure. However, the diversion of the main channel by degradation/aggradation within the confines of the levee system, or by breaching the west levee during major floods, cannot be discounted. The ultimate heat sink (as discussed in Section 2.4.9) is not directly dependent on the river intake. We conclude that channel diversions present no safety-related hazard to the plant."

V. REFERENCES

No specific publications can be cited for general use; however, site-specific publications and maps can be obtained from the United States Geologic Survey, Soil Conservation Service, National Oceanic and Atmospheric Administration, Corps of Engineers, and state and other agencies and organizations, to identify historical and potential future channel diversions.

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.4.10 FLOODING PROTECTION REQUIREMENTS

REVIEW RESPONSIBILITY

Primary - Site Analysis Branch (SAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Electrical, Instrumentation, and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The locations and elevations of safety-related facilities and of structures and components required for protection of safety-related facilities are compared with the estimated static and dynamic effects of design basis flood conditions identified in safety analysis report (SAR) Section 2.4.2.2, to determine whether flood effects need be considered in plant design or emergency procedures.

If flood protection is required, the type of flood protection ("hardened facilities," sandbags, flood doors, bulkheads, etc.) is reviewed. Any emergency procedures required to implement flood protection and warning times available for implementation thereof are reviewed, based on the flood conditions identified in other sections.

II. ACCEPTANCE CRITERIA

The flood design basis for each facility must be comparable with the positions in Regulatory Guide 1.59. For construction permit (CP) reviews, the types of flood protection proposed must be capable of protecting those safety-related structures, systems, and components identified in Regulatory Guides 1.59 and 1.29.

For operating license (OL) reviews, the specific designs of flood protection measures are reviewed to assure the protection levels are adequate (including static and dynamic effects) for the controlling flood conditions and that any necessary technical specifications are considered.

Standard engineering practice in positive flood control and shore protection, such as that developed by the Corps of Engineers, provides the basis for acceptance of methods to be employed for protection. Where sites are not "hardened," that is, where emergency action is required, the time available to implement emergency procedures must be estimated by analysis of the hydrologic design event. The environmental conditions likely to prevail during all potential flooding events up to and including events of the severity of the controlling event are compared with the requirements for implementing flood emergency procedures. If the environmental conditions likely are such that the procedures can be carried out, they will be considered acceptable. An appropriate item in the plant Technical Specifications will be required in cases where emergency procedures are required to assure adequate flood protection.

"Hardened" flood protection (as discussed in Regulatory Guide 1.59, for facilities identified in Regulatory Guide 1.29) will be interpreted to mean "almost always in place".

III. REVIEW PROCEDURES

The estimated design basis flood level is compared with the locations and elevations of safety-related components. The staff will independently determine from analyses of postulated individual hydrologic events whether flood protection is required, and if so, what protective levels (including static and dynamic effects) are applicable. These data are transmitted to Structural Engineering Branch for determination of structural competence and to Auxiliary and Power Conversion Systems Branch (APCSB) and Electrical, Instrumentation, and Control Systems Branch (EICSB) for determination of safety system adequacy. For flood protection requiring emergency action, the design basis flood conditions, and other, less severe events, are reviewed to establish the minimum time available for implementation of emergency procedures. Physical parameters such as rate-of-rise (of river or lake levels), as well as evaluation (based on experience and engineering judgment) of flood warning networks provide the staff with an independent estimate of available time. These data are provided APCSB and EISCB for their independent evaluation of the time required to implement shutdown and flood protective measures.

For OL reviews, the design of flood protection measures is reviewed to assure compatability with the original design basis. For those plants for which shutdown (if required under Regulatory Guide 1.59, position 2) and installation of protective measures is required in the event of a major flood, the procedures for carrying out these measures are reviewed for compatibility of available and required times as established above. The Technical Specifications must reference an emergency plan which allows for the orderly installation of required flood protection.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For CP reviews, the findings will consist of statements of flood design bases for safety-related facilities. If emergency procedures are required, the findings will indicate staff conclusions that time for implementation and methods of providing flood protection provide the necessary protection.

For OL reviews the findings will indicate the flood protection measures provided for safety-related facilities, and will indicate the type of technical specifications required to assure that the protection will be in place.

If Regulatory Guide 1.59, position 2, is elected by the applicant, a statement describing lesser design bases will be included in the findings with the staff's conclusion of adequacy.

A sample CP-stage statement follows:

> "The applicant states, and we concur, that the station is above the flood level of a Probable Maximum Flood (PMF), either on the A River or the two intermittent streams crossing the site.
>
> "Further, the applicant has stated that the roofs of safety-related buildings will be constructed to safely dispose of, or store, local precipitation as severe as the Probable Maximum Precipitation (PMP). Further, we conclude that the bases for plant grading and drainage will be sufficient to prevent a threat to safety-related facilities by a localized PMP."

V. <u>REFERENCES</u>

Other sections of 2.4 provide hydrologic design basis flood levels and environmental condition descriptions. Reports of the Corps of Engineers, United States Geologic Survey, Bureau of Reclamation, National Oceanic and Atmospheric Administration, and others will be used on an "as available" basis to evaluate flood warning systems, if applicable. The references for acceptability of protection will be completed projects of the Corps of Engineers and other federal, state, and local agencies, and similar types of protection previously reviewed and found acceptable for other nuclear plants.

1. Regulatory Guide 1.70, "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.4.11 LOW WATER CONSIDERATIONS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The purpose of this section of the applicant's safety analysis report (SAR) is to identify natural events that may reduce or limit the available cooling water supply, and to assure that an adequate water supply will exist to operate or shut down the plant, as required.

Depending on the site, the areas of review include:

1. The worst drought considered reasonably possible in the region.
2. Low water (setdown) resulting from surges, seiches, or tsunamis.
3. The effect of existing and proposed water control structures (dams, diversions, siltation, dam failures, etc.).
4. The intake structure and pump design basis in relation to the events described in SAR Sections 2.4.11.1, 2.4.11.2 and 2.4.11.4, and historical low water conditions.
5. The use limitations imposed or under discussion by federal, state, or local agencies authorizing the use of the water.
6. The range of water supply required by the plant, including minimum operating and shutdown flows, compared to availability.

II. ACCEPTANCE CRITERIA

Acceptance is based principally on the adequacy of the intake design basis for safe shutdown, cooldown (first 30 days), and long-term cooldown (periods in excess of 30 days) in the event of adverse natural phenomena or plant accidents. Where the specific design bases preclude plant operation during severe hydrologically-related events, sufficient warning time must be demonstrated so that the plant may be shut down during or in advance of adverse events without causing potential damage to safety-related facilities. In cases where sufficient warning time to permit advance shutdown is considered necessary to protect safety-related components, an item in the plant Technical Specifications will be required.

SAR Section 2.4.11.1 (Low Flow in Rivers and Streams): for essential water supplies the low flow/low level design for the primary water supply source must be based on the probable minimum low flow and level resulting from the most severe drought that can reasonably be considered possible for the region. The low flow and level design bases for operation (if different than the design bases for essential water requirements) should be such that shutdowns caused by inadequate water supply will not cause frequent use of emergency systems. In cases where a common source of cooling water for operation and safety is provided, and where operation can affect minimum levels required for safety, the system will be acceptable if technical specifications are provided for shutdown before the ultimate heat sink can be adversely affected.

SAR Section 2.4.11.2 (Low Water Resulting from Surges, Seiches or Tsunamis): if the site is susceptible to such phenomena, minimum water levels resulting from setdown (sometimes called runout or rundown) from hurricane surges, seiches, and tsunamis must be higher than the intake design basis for essential water supplies. For coastal sites, the appropriate probable maximum hurricane (PMH) wind fields must be postulated to give maximum winds blowing offshore, thus creating a probable minimum surge level. Low water levels on inland ponds, lakes, and rivers due to surges must be estimated from probable maximum winds oriented away from the plant site. The same general analysis methods discussed in Standard Review Plans 2.4.3, 2.4.5 and 2.4.6 are applicable to low water estimates due to the various phenomena discussed.

SAR Section 2.4.11.3 (Historical Low Water): if historical flows and levels are used to estimate design values by inference from frequency distribution plots, the data used must be presented so that an independent determination can be made. The data and methods of the National Oceanic and Atmospheric Administration, United States Geologic Survey, Soil Conservation Service, Bureau of Reclamation and the Corps of Engineers are acceptable.

SAR Section 2.4.11.4 (Future Controls): this section is acceptable if water use and discharge limitations (both physical and legal), already in effect or under discussion by responsible federal, regional, state, or local authorities, that may affect water supply at the plant have been considered and are substantiated by reference to reports of the appropriate agencies. The most adverse possible effects of these controls must be shown and taken into account in the design basis to assure that essential water supplies are not likely to be affected adversely in the future.

SAR Section 2.4.11.5 (Plant Requirements): acceptance is based on the following required information:

1. Minimum essential cooling water flow rates and levels must be presented (or cross-referenced) and shown to be less than the probable minimum low flows and levels from the applicable sources of supply.
2. Maximum water requirements for normal operation must be presented and (if applicable) shown to be less than the water available under all likely conditions from the sources of supply.

SAR Section 2.4.11.6 (Heat Sink Dependability Requirements): the required data and information are those necessary to determine that the facility meets the criteria of Regulatory Guide 1.27. The analyses will be considered complete and acceptable if the following are adequately addressed:

1. The initial water inventory must be sufficient for shutdown and cooldown of the plant.
2. Water losses (such as seepage, drift, and evaporation) must be conservatively estimated, as suggested in Revision 1 to Regulatory Guide 1.27.
3. The design basis hydrometeorology (temperature, dewpoint, etc.) must be as conservative as the criteria of the guide (see Standard Review Plans 2.3).
4. The limit on the heat sink return water temperature must be less than the maximum allowable cooling water inlet design temperature.

III. REVIEW PROCEDURES

Minimum plant requirements (water level and flow) that are identified in SAR Sections 2.4.11.5 or 9.2.5 are compared to the estimated minimum water levels and flows given in Section 2.4.11.1. If normal operation is not assured at the minimum water supply conditions, and loss of normal operation capability can adversely affect safety-related components, estimates of warning time are reviewed to assure that shutdown or conversion to alternate water sources can be accomplished prior to the trip. For such cases, emergency operating procedures are required, and are reviewed to assure that they are consistent with the postulated conditions. The analysis of the dependability of the ultimate heat sink is reviewed and the conclusions are provided to the Auxiliary and Power Conversion Systems Branch (APCSB). Determination of the dependability of the ultimate heat sink is accomplished by using Revision 1 of Regulatory Guide 1.27 as a standard of comparison.

Each source of water for normal or emergency shutdown and cooldown, and the natural phenomena and site-related accident design criteria for each should be identified. A systems analysis is first undertaken of all water supply sources to determine the likelihood that at least one source would survive (1) the most severe of each of the natural phenomena; (2) site-related accident phenomena; and (3) reasonable combinations of less severe natural and accident phenomena. Second, arbitrarily assumed mechanistic failures of water supply structures and conveyance systems are postulated and the systems analysis repeated, to assure that the failure of one component will not cause failure of the entire system. These analyses are coordinated with the APCSB review of the ultimate heat sink and related cooling systems, to avoid duplication. Operating rules for each portion of the system are ascertained to determine the amount of water that can be assumed available in the event of normal or accidental shutdown. Consultations with the Meteorology and the Seismology, Geology, and Foundation Engineering Sections of SAB, and with Accident Analysis Branch, Structural Engineering Branch, and APCSB are undertaken where design criteria are not firmly established.

Estimates of water loss due to drift, evaporation, and blowdown are evaluated based on observed severe hydrometeorological measurements at similar locations (coordinated with the Meteorology Section of SAB). If independent analyses are deemed necessary, computer programs such as HEC-2 (Water Surface Profiles), HEC-3 (Reservoir System Analysis) HEC-4 (Monthly Streamflow Simulation), etc. are utilized.

The potential for surges in intake sumps that could cause adverse effects are reviewed to assure that the effects have been properly incorporated for the intake design. The potential for adverse hydrodynamic effects of a trip of the intake pumps is evaluated based on potential surges in intake sumps.

For multiple purpose (normal operation, normal shutdown, and emergency shutdown) water supply systems, the primary portion of the system is first reviewed to determine that the water supply will be maintained at minimum volume requirements at all times. The secondary portion of the system is then reviewed to determine whether an adequate emergency water supply can be expected to be available during operating conditions such as the regional drought of record (flows must be adjusted for historical and potential future effects). If not, the applicant is requested to provide a technical specification requiring plant shutdown at the point where an adequate shutdown water supply is still assured.

Institutional restraints on water use, such as limitations in water use and discharge permits, are reviewed to assure the plant will have an adequate supply and not exceed limitations imposed upon operation. If a conflict is foreseen, the applicant is requested to either obtain a variance or make a design change to accommodate the limitation.

For plants using rivers, minimum design service water levels are compared with asymtotic extrapolations of low flow frequency curves which have been corrected for historical and potential future effects. For ocean or estuary plants, design low water levels are compared with probable maximum hurricane and tsunami-induced low water levels. For Great Lakes plants, design low water levels are compared with minimum historical levels coincident with probable maximum surge or seiche-induced low water levels.

If the ultimate heat sink system is not capable of continued long-term water supply under the criteria in Revision 1 to Regulatory Guide 1.27, or the above considerations, the system will be reviewed in two parts; short-term capability and long-term capability. For short-term capability, the APCSB and the Licensing Project Manager (LPM) will be informed if the independently-estimated supply appears to be less than 30 days. The applicant will be asked to determine whether sufficient personnel and equipment can safely be made available to switch water supply sources in the event of an accident. If emergency procedures are required to obtain the use of alternate water supplies, the applicant's water supply sources and procedures will be reviewed with APCSB and the LPM to determine that there is continuity of water supply. The time period for which a highly dependable water supply would be available is compared with the time required to obtain water from an alternative supply, and the natural or accident environmental conditions which could prevail.

For long-term water supply capability, different sources and means of obtaining water may be required because of the limited capability of a "short-term" supply. In those cases where different sources are necessary to assure the long-term plant heat removal capability, the alternative sources and the means of supplying water from the sources to the plant should be identified. Any plant design provisions necessary for such situations should also be described or a reference provided to other SAR sections for the descriptions.

Emergency means for obtaining long-term water supplies will be judged on the basis of the time required to obtain such supplies, natural or accident phenomena likely to prevail or to have caused the need for such supplies, and the dependability of the supply itself.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews the findings will consist of a statement of the applicant and staff estimates of the design basis minimum water flows and levels. If the estimates are similar, staff concurrence in the applicant's estimate will be stated. If the staff estimates substantially lower water levels or flows, and if the proposed plant may be adversely affected, a statement of the staff position (bases) will be made. A similar finding on the design bases for the ultimate heat sink will be made. If technical specification requirements are needed to assure an adequate supply, they will be indicated in the CP statement and required for operation.

For operating license (OL) reviews of plants for which detailed low water reviews were done at the CP stage, the CP conclusions will be referenced. In addition, the results of a review to reaffirm the low water design bases will be noted. If no changes have been made to the ultimate heat sink design since the CP review, the conclusions of the CP will be referenced. However, for both the low water considerations and the ultimate heat sink, an evaluation will be made during the OL review to assure that the design bases have been properly implemented. The availability of long-term water supply will be noted. If no low water and ultimate heat sink review was undertaken at the CP stage (of the scope described), this fact will be noted also.

A sample CP-stage statement follows:

"The applicant proposes two sources of water supply; groundwater and the adjacent A River.

"Groundwater would be used for make-up to the essential service water cooling towers, for potable water supply, and for demineralizer water. The applicant estimates the demineralizer would require about 825 gallons per minute (gpm) for the first several months and an average rate of 425 gpm thereafter. Potable water requirements are estimated at about 10 gpm.

"The A River is to provide the principal source of cooling water. The applicant estimates the maximum water requirement for the plant will be 107 cfs. Of this, 61 cfs would be consumptively used and 46 cfs would be returned to the Rock River. The historical recorded low flow in the A River in the site region was about 500 cfs at the B gage on September 14, 1958 and about 440 cfs at the C gage on August 20, 1934. The applicant estimates the comparable low flow at the site to be 400 cfs. Assuming breaching of D Dam five miles downstream, the low flow would result in an estimated water surface elevation of 664 ft MSL.

"Emergency cooling sources and associated principal facilities comprise the A River, groundwater, the river screenhouse, the essential service cooling towers, groundwater well(s) and attendant distribution systems. The river screenhouse is to be a seismic Category I facility and was initially proposed to be protected from flooding up to the Standard Project Flood (SMF). Groundwater wells, located at the plant site, are above estimated PMF water levels. The applicant proposes to use groundwater for make-up to the essential service towers whenever the A River, screenhouse, or piping is unavailable. Estimated groundwater use would be 1600 gpm. At the staff's request the applicant reconsidered the flood design basis for the river screenhouse for relatively long periods of time when the A River could be higher than a SPF and an earthquake could prevent water from being available from wells. The applicant subsequently upgraded the flood design basis for the screenhouse to a Probable Maximum Flood, and concludes the proposed facilities meet the suggested criteria of Regulatory Guide 1.27 - Ultimate Heat Sink. We concur.

V. REFERENCES

1. L. R. Beard, "Methods for Determination of Safe Yield and Compensation Water from Storage", Seventh International Water Supply Congress, Barcelona, Spain (1966).

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3. B. R. Bodine, "Storm Surge on the Open Coast: Fundamentals and Simplified Prediction", Technical Memorandum No. 35, Corps of Engineers Coastal Engineering Research Center, May 1971.

4. D. K. Brady, et al., "Surface Heat Exchange at Power Plant Cooling Lakes", EEI Publication 69-901, Edison Electric Institute, New York, Nov. 1969.

5. V. T. Chow (ed), "Handbook of Applied Hydrology", McGraw-Hill Book Company, New York (1964).

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6A. J. E. Edinger, et al., "Generic Emergency Cooling Pond Analysis", prepared for U.S. Atomic Energy Commission under Contract No. AT(11-1)-2224 (1972).

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12. A. Maas, et al., "Design of Water-Resources Systems," Harvard University Press, Cambridge, Mass. (1962).

13. G. W. Platzman, "The Dynamical Prediction of Wind Tides on Lake Erie", Technical Report No. 7, Department of Geophysical Sciences, University of Chicago (1962).

14. R. O. Reid and B. R. Bodine, "Numerical Model for Storm Surges in Galveston Bay", Jour. Waterways and Harbors Division, Am. Soc. Civil Engineers, Vol. 94, No. WW1, pp. 33-57 (1968).

15. "Hydrologic Engineering Methods for Water Resources Development", Vol. 1-12, Corps of Engineers Hydrologic Engineering Center, Davis, California (1971).

16. "Reservoir Storage-Yield Procedures", Corps of Engineers Hydrologic Engineering Center, Davis, California (1967).

17. "Shore Protection Planning and Design", Technical Report No. 4, Third Edition, Corps of Engineers Coastal Engineering Research Center (1966); and "Shore Protection Manual," 1973.

18. Regulatory Guide 1.27, "Ultimate Heat Sink", Revision 2.

19. "Design of Small Dams", Second Edition, Bureau of Reclamation, U.S. Department of Interior (1973).

20. "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States", Report HUR 7-97 (see also HUR 7-97A), U.S. Weather Bureau (now NOAA) (1968).

21. "Water Surface Profiles", HEC-2, Corps of Engineers Hydrologic Engineering Center (continuously updated).

22. "Reservoir System Analysis", HEC-3, Corps of Engineers Hydrologic Engineering Center (updated).

23. "Monthly Streamflow Simulation", HEC-4, Corps of Engineers Hydrologic Engineering Center (updated).

24. Regulatory Guide 1.70, "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.4.12 DISPERSION, DILUTION, AND TRAVEL TIMES OF ACCIDENTAL RELEASES OF LIQUID EFFLUENTS IN SURFACE WATERS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The ability of the surface water environment to disperse, dilute, or concentrate normal or severe cases of accidental radioactive liquid effluent releases is reviewed with emphasis on relating the effects of such releases to existing and known future uses of surface water resources. (Note that effects of normal releases and of the more likely accidents are discussed in the applicant's environmental report.)

II. ACCEPTANCE CRITERIA

Dispersion characteristics and dilution capability of the surface water environment with respect to existing and known future users must be described for both normal and accident conditions. Estimates and bases for dilution factors, dispersion coefficients, flow velocities, and travel time between the site and existing or known future users must be described for both normal and accident conditions. Potential pathways of contamination to surface water users must be identified. Sources of data must be described and referenced. Acceptance is based on a comparison of applicant and staff results.

III. REVIEW PROCEDURES

Independent conservative calculations will be made for dispersion coefficients, dilution factors, flow velocities, travel times, recirculation, and potential contamination pathways. Dispersion coefficients for surface waters are estimated using methods such as those suggested by Brooks (Ref. 1) and Fisher (Refs. 2 and 3). The minimum historic low flow rate of a receiving stream (where applicable), adjusted for diversions or other phenomena that may have affected or likely will affect that rate, is assumed coincident with the spill. Conservatism should be used in the selection of coefficients and parameters for use in any of these methods to determine accident effects. The applicant's design basis is compared to the staff's calculations to determine whether it is adequate, and is reviewed to see that it reflects any potential future changes that might result from variations in precipitation or by the construction of known future wells, reservoirs, and intakes.

Any missing data, information, or analyses necessary to conduct the above reviews and evaluations will be requested in first-round questions. Applicant responses will be evaluated using the above referenced, or similar methods, and staff positions will be developed and supplied to the applicant. If responses to staff positions are unacceptable, resolution will be attempted with the applicant prior to preparing evaluation findings, or differences will be noted therein.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will consist of a statement of the applicant and staff estimates of dilution factors, dispersion coefficients, flow velocities, travel times, and potential contamination pathways between the site and the nearest water user. If the estimates are similar, or if no potential problem exists, staff concurrence with the applicant's estimates will be stated. If the staff predicts substantially more conservative conditions, a statement of the staff basis will be made.

For operating license (OL) reviews of plant designs that have had detailed reviews of severe accidental effluent releases at the CP stage, the CP conclusions will be referenced. If no CP review of effluent releases was undertaken of the scope indicated herein, this will be indicated. Any new potential pathways or changes in water usage that can be identified in the OL review will also be analyzed and reported.

Sample statements for CP reviews follow:

> "At the staff's request, the applicant provided analyses of the effects (travel times, dispersion coefficients, dilution factors, etc.) of an accidental spill of liquid radioactive wastes into the surface water. A postulated failure of the condensate storage tank, releasing 500,000 gallons of water containing low-level activity was evaluated. The applicant assumed that this volume of water would travel overland to the adjacent stream before any dilution would occur. The applicant concluded, and the staff concurs, that adequate dilution would occur in the surface water prior to reaching any potential users. The applicant also investigated the possibility of the spill being recirculated through the plant circulating water system. This analysis showed that it was extremely unlikely that recirculation could occur since the condensate storage tank is located downstream of the circulating water intake structure. The staff concurs in this evaluation. Accidental spills that could enter the groundwater and reach potential users before or after discharging into surface waters are discussed in Sections 2.4.13 and Section 15 of this report."

> "No accidental release of sufficient volume of liquids containing radioactivity directly into surface waters is considered reasonable at the site because storage facilities are located inside of safety-related buildings and the manner in which liquids are to be handled at the site precludes this possibility. Accidental spills of liquids into the groundwater, which could eventually reach surface waters, are discussed in Sections 2.4.13 and 15 of this report."

V. REFERENCES

In addition to the following references describing methods and techniques of evaluation, published data by federal, state, and other agencies and organizations will be used as available.

1. N. H. Brooks, "Diffusion of Sewage Effluent in an Ocean Current," in "Waste Disposal in the Marine Environment," Pergamon Press, New York (1960).

2. H. B. Fisher, "The Mechanics of Dispersion in Natural Streams," Jour. Sanitary Engineering Division, Proc. Am. Soc. Civil Engineers, Vol. 93, No. HY6, pp. 187-216 (1968).

3. H. B. Fisher, "Dispersion Predictions in Natural Streams," Jour. Sanitary Engineering Division, Proc. Am. Soc. Civil Engineers, Vol. 94, No. SA5, pp. 927-943 (1968).

4. E. Gaspar and M. Oncescu, "Radioactive Tracers in Hydrology," Elsevier Publishing Co., New York (1972).

5. S. N. Davis and R. J. M. DeWiest, "Hydrogeology," John Wiley & Sons, Inc., New York (1966).

6. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 2.4.13 GROUNDWATER

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Data presented in the applicant's safety analysis report (SAR) on local and regional groundwater reservoirs are reviewed to establish the effects of groundwater on plant foundations. Other areas reviewed under this plan include identification of the aquifers and the type of onsite groundwater use, the sources of recharge, present and future withdrawals, an evaluation of accident effects, monitoring and protection requirements, and design bases for groundwater levels and hydrodynamic effects of groundwater on safety-related structures and components. Flow rates, travel time, gradients, and groundwater levels beneath the site are reviewed, as are seasonal and climatic fluctuations, or those caused by man, that have the potential for long-term changes in the local groundwater regime.

II. ACCEPTANCE CRITERIA

For SAR Section 2.4.13.1: a full, documented description of regional and local groundwater aquifers, sources, and sinks is required. In addition, the type of groundwater use, wells, pump and storage facilities, and the flow requirements of the plant must be described. If groundwater is to be used as an essential source of water for safety-related equipment, the design basis for protection from natural and accident phenomena must compare with Regulatory Guide 1.27 guidelines. Bases and sources of data must be adequately described.

For SAR 2.4.13.2: a description of present and projected local and regional groundwater use must be provided. Existing uses, including amounts, water levels, location, drawdown, and source aquifers must be discussed and should be tabulated. Flow directions, gradients, velocities, water levels, and effects of potential future use on these parameters, including any possibility for reversing the direction of groundwater flow, must be indicated. Any potential groundwater recharge area within the influence of the plant and effects of construction, including dewatering, must be identified. The influence of existing and potential future wells with respect to groundwater beneath the site must also be discussed. Bases and sources of data must be described and referenced.

For SAR Section 2.4.13.3: dispersion characteristics and dilution capability of the groundwater environment with respect to existing and future users must be described for both operating and accident conditions. Estimates and bases for coefficients of dispersion and dilution, groundwater velocities, travel times, gradients, permeabilities, porosities, and groundwater or piezometric levels between the site and existing or future users must be described and be consistent with site characteristics. Potential pathways of contamination to groundwater users must also be identified. Sources of data must be described and referenced.

For SAR Section 2.4.13.4: the need for and extent of procedures and measures to protect groundwater users, including monitoring programs, must be discussed. These items are site-specific and will vary with each application.

For SAR Section 2.4.13.5: the design bases (and development thereof) for groundwater-induced hydrostatic loadings on subsurface portions of safety-related structures, systems, and components must be described. If construction dewatering is critical to the integrity of safety-related structures, the bases for subsurface hydrostatic loadings assumed during construction and the dewatering methods to be employed must be described. In addition, if wells are proposed for safety-related purposes, the hydrodynamic design bases (and development thereof) for protection against seismically-induced pressure waves must be described and be consistent with site characteristics.

III. REVIEW PROCEDURES

The review sequence is shown on Figure 2.4.13. Local and regional groundwater conditions are reviewed by comparing the applicant's description with reports by the U. S. Geological Survey (USGS), other agencies, and professional organizations. Other branches with related review responsibilities will be notified of any applicable groundwater data and analyses. If onsite groundwater use and facilities are safety-related, the criteria of Regulatory Guide 1.27 are applied.

The staff will compare the applicant's description of present and projected local and regional groundwater use, existing users, including ambient use, water levels, location, and drawdown with information and data from references. Drawdown effects of projected future groundwater use, including the possibility for reversing the groundwater flow, will be evaluated and may be checked by independent calculations. Construction effects, including dewatering, on potential recharge areas may also be evaluated.

Independent calculations will be made of the dispersion and dilution capabilities and potential contamination pathways of the groundwater environment under operating and accident conditions with respect to existing and future users. The needs and plans for procedures, measures, and monitoring programs to protect groundwater users will also be reviewed based upon the site-specific groundwater features. Design bases for groundwater-induced hydrostatic loadings on subsurface portions of safety-related structures are reviewed and compared with independent check calculations to determine whether the data base used is adequate to reflect any potential future changes which can be induced by variations in precipitation, or by the construction of future wells and reservoirs.

Any missing data, information and analyses necessary to conduct the above reviews and evaluations will be requested in first-round questions. Responses will be evaluated, and if necessary, computer programs for groundwater models (e.g., Refs. 2 and 5) may be used to determine the effects of changing groundwater conditions on site safety, and of accidents on regional and local groundwater supply. Staff positions will then be developed and supplied to the applicant.

The above reviews are performed only when applicable to the site or site region. Some items of review may be done on a generic basis.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings will consist of a statement of the applicant and staff estimates of groundwater levels associated with safety-related structures, and where applicable, groundwater flow directions, gradients, velocities, effects of potential future use on these parameters, and the effects of an accident on existing and future users. If the estimates are similar, staff concurrence in the applicant's estimates will be stated. If the staff predicts substantially more conservative groundwater conditions for which the proposed plant may be adversely affected, a statement of the staff bases will be made. If groundwater conditions do not constitute design bases, the findings will so indicate.

For operating license (OL) reviews of plants that have had detailed groundwater reviews at the CP stage, the CP conclusions will be referenced. In addition, a review of groundwater history since the CP review will be indicated and note of any changes in groundwater conditions or usage will be made. If no CP groundwater review was undertaken, of the scope indicated herein, this will be indicated.

A sample CP statement follows:

> "Groundwater is available at the site in low to moderate yields from the following four aquifers listed by increasing depth below the surface: (1) the unconfined water-table aquifer consisting of the A and B formations, (2) the confined C-Upper D aquifer, (3) the confined upper D aquifer, and (4) the confined middle D aquifer. Groundwater in the A-B town aquifer generally moves toward the local streams; whereas, in the deeper confined aquifers, groundwater generally moves toward centers of pumping. At the present, saltwater intrusion into the aquifers at the site is not evident as a result of brackish water movement from the E Bay, the F Canal, or G Bay.
>
> "The applicant plans to use groundwater during plant operation at a continuous rate of 140 gpm, of which 100 gpm will be used for demineralized water requirements, and 40 gpm will be service water for drinking, washing, and filling the fire protection storage tanks. The source of this supply will probably be the A-B aquifer, for which the applicant has conducted pumping tests at two locations. The applicant has indicated he may utilize another deeper aquifer for this supply, and has agreed to supply additional pumping test data to the staff for evaluation if another aquifer is chosen. This is acceptable to the staff.

"Precipitation is the source for groundwater recharge to the A-B aquifer. The recharge area for this aquifer lies to the southwest of the plant site and extends beyond the City of H. No major recharge areas for the lower confined aquifers are believed to exist in the vicinity of the site.

"A water-table design level of 65 feet MSL (15 feet below plant grade) was selected by the applicant to determine hydrostatic loadings on safety-related structures. The staff concurs that this level is conservative since the highest measured water table elevation at the plant site following an extremely rainy season was 63.4 feet MSL."

V. REFERENCES

In addition to the following, references on methods and techniques of analysis, published data by federal and state agencies, such as USGS water supply papers, will be used as available.

1. J. D. Bredehoeft and G. F. Pinder, "Digital Analysis of Areal Flow in Multiaquifer Groundwater Systems: A Quasi Three-Dimensional Model," Water Resources Research, Vol. 6, No. 3, pp. 883-888 (1970).

2. "Finite Element Solution of Steady State Potential Flow Problems," HEC 723-G2-L2440, Corps of Engineers (1970).

3. T. A. Prickett and C. G. Lonnquist, "Selected Digital Computer Techniques for Groundwater Resource Evaluation," Bulletin 55, Illinois State Water Survey, Urbana, Illinois (1970).

4. D. B. Cearlock and A. E. Reisenauer, "Sitewide Groundwater Flow Studies for Brookhaven National Laboratory, Upton, Long Island, New York," Battelle Pacific Northwest Laboratories, Richland, Washington (1971).

5. K. L. Kipp, D. B. Cearlock, A. E. Reisenauer, and C. A. Bryan, "Variable Thickness Transient Groundwater Flow Model--Theory and Numerical Implementation," BNWL-1703, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

6. D. R. Friedrichs, "*Information Storage and Retrieval System for Well Hydrograph* Data--User's Manual," BNWL-1705, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

7. K. L. Kipp and D. B. Cearlock, "The Transmissivity Iterative Calculation Routine--Theory and Numerical Implementation," BNWL-1706, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

8. S. W. Ahlstrom, R. J. Serne, R. C. Routson, and D. B. Cearlock, "Methods for Estimating Transport Model Parameters for Regional Groundwater Systems," BNWL-1717, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

9. R. C. Routson and R. J. Serne, "One-Dimensional Model of the Movement of Trace Radioactive Solutes Through Soil Columns: The PERCOL Model," BNWL-1718, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

10. R. C. Routson and R. J. Serne, "Experimental Support Studies for the PERCOL and Transport Models," BNWL-1719, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

11. K. L. Kipp, D. B. Cearlock, and A. E. Reisenauer, "Mathematical Modeling of a Large, Transient, Unconfined Aquifer with a Heterogeneous Permeability Distribution," Paper presented at the 54th Annual Meeting of the American Geophysical Union, Washington, D.C., April 1973.

12. D. L. Schreiber, A. E. Reisenauer, K. L. Kipp, and R. T. Jaske, "Anticipated Effects of an Unlined Brackish-Water Canal on a Confined Multiple-Aquifer System," BNWL-1800, Battelle Pacific Northwest Laboratories, Richland, Washington (1973).

13. Regulatory Guide 1.27, "Ultimate Heat Sink," Revision 2.

14. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

15. W. H. Li and F. H. Lai, "Experiments on Lateral Dispersion in Porous Media," Jour. Hydraulics Division, Proc. Am. Soc. Civil Engineers, Vol. 92, No. HY6 (1966).

16. W. H. Li and G. T. Yeh, "Dispersion of Miscible Liquids in a Soil," Water Resources Research, Vol. 4, pp. 369-377 (1968).

17. D. R. F. Harleman, P. F. Mehlhorn, and R. R. Rumer, "Dispersion-Permeability Correlation in Porous Media," Jour. Hydraulics Division, Proc. Am. Soc. Civil Engineers, Vol. 89, No. HY2, pp. 67-85 (1963).

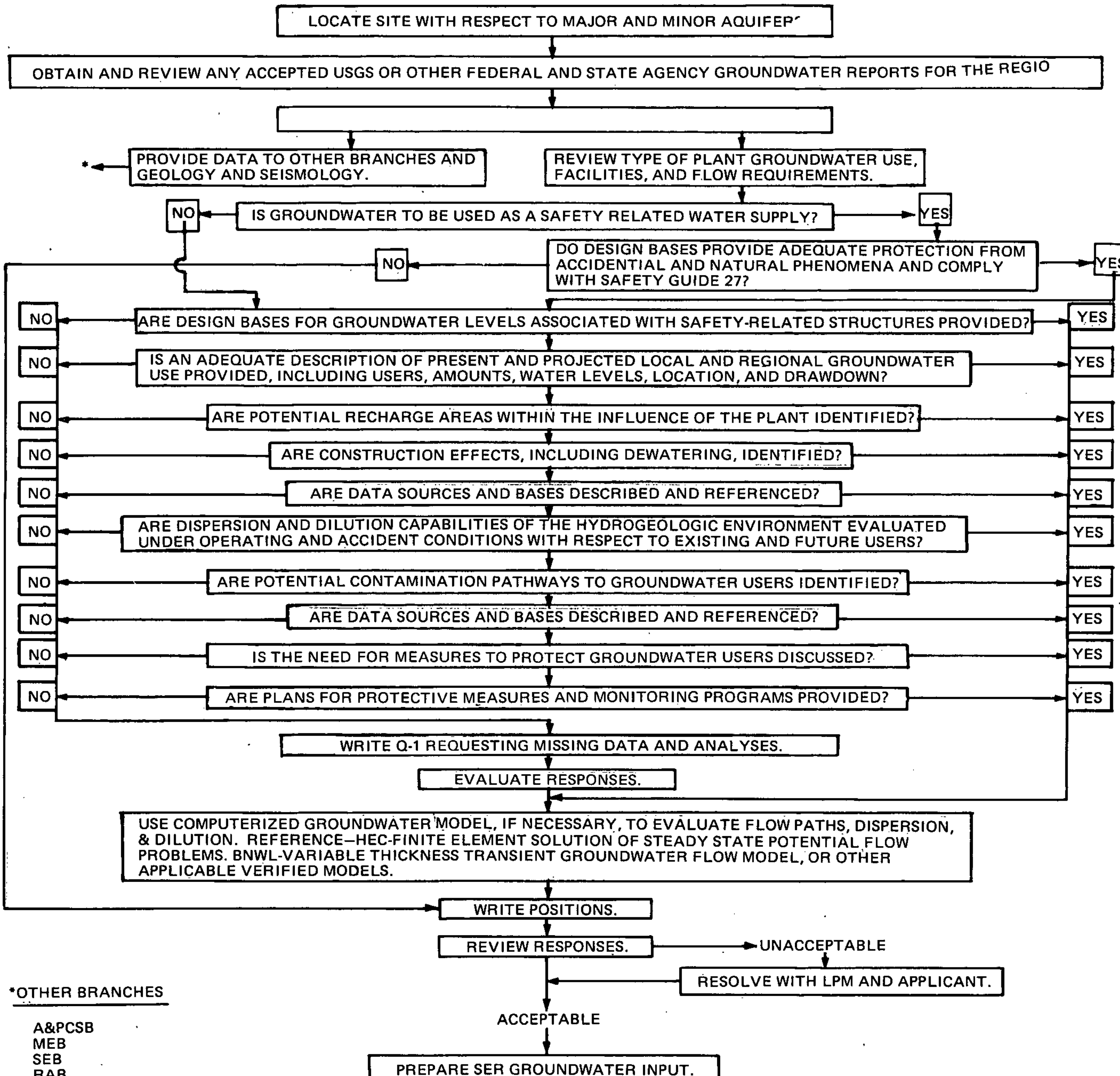
18. L. E. Addison, D. R. Friedrichs, and K. L. Kipp, "The Transmissivity Iterative Programs on the PDP-9 Computer--A Man-Machine Interactive System," BNWL-1707, Battelle Pacific Northwest Laboratories, Richland, Washington (1972).

19. "Fundamentals of Transport Phenomena in Porous Media," International Association for Hydraulic Research, Elsevier Publishing Company, New York (1972).

20. D. K. Todd, "Ground Water Hydrology," John Wiley & Sons, Inc., New York (1959).

FIGURE 2.4.13

STANDARD REVIEW PLAN 2.4.13
GROUNDWATER



*OTHER BRANCHES

A&PCSB
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SEB
RAB
AAB
ETSB

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SECTION 2.4.14 TECHNICAL SPECIFICATIONS AND EMERGENCY OPERATION REQUIREMENTS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The purpose of this section of the applicant's safety analysis report (SAR) is to identify the technical specifications and emergency procedures required to implement flood protection for safety-related facilities and to assure an adequate water supply for shutdown and cooldown purposes.

II. ACCEPTANCE CRITERIA

If the hydrologic design bases developed in preceding sections do not necessitate technical specifications or emergency procedures to ensure safety-related plant functions, this section should so state. The balance of this review plan assumes requirements for technical specifications or emergency procedures.

This section will be acceptable if the following are identified:

1. The controlling hydrologic events, as developed in the preceding sections of SAR Chapter 2.
2. The actions to be taken, and the effect of such actions on the protection of safety-related facilities.
3. The appropriate water levels and conditions at which action is to be initiated.
4. The appropriate emergency procedures, and the amount of time required to implement each procedure.

III. REVIEW PROCEDURES

The review procedures consist of comparing the proposed specifications and procedures with the flood protection and water supply design bases derived in the preceding sections, or considered necessary by the staff. Data in, or derived from, the preceding sections are used to estimate the time available to complete any required emergency action (e.g., sandbagging, shutdown, installing flood gates and stop logs). This information will also serve to substantiate the water levels and other conditions used to initiate the action. Specific questions on the structural adequacy of protective measures are referred to Structural Engineering Branch, and the general experience of the Corps of Engineers in such situations, as reflected in reports and manuals, is the principal basis for comparison.

IV. EVALUATION FINDINGS

For both construction permit and operating license reviews the findings will consist of a brief statement of technical specifications and emergency procedures and time required to implement flood protection of safety-related facilities and assure an adequate water supply for safety-related equipment. The flood or water levels and other conditions at which action is to be initiated will also be stated. If none are required, the findings will so state.

A sample Operating License statement follows:

> "The staff has taken a position that it would be prudent to shut the plant down before water could reach plant grade during severe hurricanes. The applicant has maintained that design of the safety-related facilities includes provision for protection. The staff believes the implementation of emergency procedures, required in the event of severe hurricanes to assure the watertightness of exterior doors and to minimize the possible equipment failure which could occur during such an event (should the applicant's single water barrier design provisions not be adequate), would be extremely difficult from a practical standpoint. The staff, therefore, will require a provision in the plant's Technical Specifications requiring a flood alert, referring to emergency procedures, when water levels exceed elevation 15 feet MSL. In the case of PMH, this would allow a minimum of about 4 hours before water would cross plant grade (some six hours before maximum water levels would be reached) to implement emergency action. Examples of required action are: assuring all exterior accesses are closed and sealed, adequate diesel fuel oil supplies are protected, sandbagging of vulnerable areas may be undertaken, and any necessary emergency equipment is available and operational. The weather conditions during such a situation would be severe (high winds, rain, the likelihood of tornadoes in the area, etc.), but implementation of outdoor emergency procedures are considered reasonable if accomplished before maximum storm conditions occur.
>
> "The applicant has installed a control room water level alarm that is activated when the water level in the intake canal reaches elevation 17.5 feet MSL. The staff will require the same technical specification to necessitate an orderly plant shutdown upon activation of the alarm. The requirement is prudent in view of the single line of defense inherent in the water barriers installed by the applicant. Failure of such barriers with the reactor at or near operating levels would allow a very limited time, during extreme weather conditions, for plant operating personnel to prevent a major accident. No other technical specification provisions are considered necessary for hydrologically-related events."

V. REFERENCES

Data and information presented in, or derived from, previous Standard Review Plans in the 2.4 series provide the basic reference material for this section.

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SECTION 2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

SAB reviews the geologic and seismic information submitted in the applicant's safety analysis report (SAR) in accordance with Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." SAB judges the adequacy of the geologic and seismic information cited in support of the applicant's conclusions concerning the suitability of the plant site. The geologic and seismic information which must be provided in order for the site review to proceed is divided into the following three categories:

1. Geologic features: mass-wasting, differential subsidence, faulting, soil and foundation instability, chemical weathering, cavernous or karst terrains, and volcanism.

2. Seismic features: ground failure under dynamic loading, liquefaction, vibratory ground motion, site amplification, tsunami, and residual stresses.

3. Man-made conditions: subsidence or collapse caused by withdrawal of fluids or mineral extraction, and fault movement caused by fluid injection or withdrawal.

Information relating to the above conditions should be presented in SAR Sections 2.5.1.1 (Regional Geology) and 2.5.1.2 (Site Geology). This information should be discussed in terms of the regional and site physiography, geomorphology, stratigraphy, lithology, and tectonics. In addition, with specific reference to site geology, the following subjects should be discussed as they relate to the above-mentioned conditions: topography, slope stability, fluid injection or withdrawal, mineral extraction, faulting, shearing, jointing and fracturing.

The above information should be documented by appropriate references to all relevant published and unpublished materials. Illustration should include but should not be limited to physiographic, topographic, geologic, tectonic, gravity, and magnetic maps, structure and stratigraphic sections, boring logs, and aerial photographs. Certain sites will require illustrations of specialized character such as maps of subsidence, irregular weathering conditions, landslide potential, hydrocarbon extraction (oil or gas wells), and karst features. Some site characteristics must be documented by reference to seismic reflection or refraction profiles or to maps produced by various remote sensing techniques.

As appropriate, maps should include a superimposed plot plan of the plant facilities. Other documentation should show the relationship of all seismic Category I facilities (clearly identified) to subsurface geology. Core boring logs, logs and maps of trenches, aerial and Environmental Resources Technology Satellite (ERTS) photographs, and geophysical data should be presented for evaluation. In addition, a plot plan showing the locations of all structures, borings, trenches, profiles, etc. should be included.

The review can be brought to an earlier conclusion if the following suggestions are followed by the applicant. The SAR should contain sufficient data to allow the reviewer to make an independent assessment of the applicant's conclusions. That is, the reviewer should be led in a logical manner from the data and premises given to the conclusions that are drawn without having to make an extensive independent literature search. Controversial information should not be ignored so as to enhance a particular position. The geologic terminology used should conform to standard reference works (Refs. 3, 6). Finally, the objective of Section 2.5 of the SAR is to describe geologic and seismic features as they affect the site under review, and all data, information, discussions, interpretations, and conclusions should be directed to this objective. Aimless presentation of data, although it may appear to satisfy the investigative requirements, will result in a disjointed SAR and cause needless delays in completing the safety review.

II. ACCEPTANCE CRITERIA

The "Seismic and Geologic Siting Criteria for Nuclear Power Plants" (Ref. 1) and the Standard Format (Ref. 2) are the basis for the staff review of all cases. The information presented in the SAR must be complete and thoroughly documented, and must be consistent with the requirements of References 1 and 2. United States Geological Survey (USGS) and other federal or state agency published and open file papers, maps, aerial photographs, geophysical data, etc., covering the region in which the site is located, are used to establish the staff's conclusions as to the completeness and acceptability of the SAR.

Subsection 2.5.1.1, "Regional Geology," will be considered acceptable if a complete and documented discussion is presented of all geologic, seismic, and man-made features. This section should contain a review of the regional physiography, geomorphology, stratigraphy, structure, and geologic history to provide a framework within which the geologic, seismic and man-made features of safety significance to the site can be evaluated.

Subsection 2.5.1.2, "Site Geology," will be judged acceptable if it contains a description and evaluation of site-related geologic features, seismic conditions, and man-made conditions which are a potential hazard to the site. This section should also contain the following general site information:

1. The site stratigraphy, including relationship to and correlation with the regional stratigraphy.

2. The structural geology of the site and the relationship of site structure to regional tectonics.

3. The geologic history of the site as it relates to the regional geologic history.

4. The engineering significance of geologic features underlying the site as they relate to:

   a. Dynamic behavior during prior earthquakes.

   b. Zones of alteration, irregular weathering, or zones of structural weakness.

   c. Unrelieved residual stresses in bedrock.

   d. Materials that could be unstable because of their mineralogy or unstable physical properties.

   e. Effects of man's activities in the area.

5. The site groundwater conditions.

III. REVIEW PROCEDURES

The staff review is conducted in three phases. The first phase is the acceptance review, a brief review of the SAR to evaluate its completeness and to identify obvious safety issues that could result in delays at subsequent stages of the review. After an SAR is docketed, the staff conducts a thorough review of the material. In this second phase of the review an effort is made to identify all safety issues. The reviewer should carefully examine the SAR to see that all interpretations are founded on sound geological and seismological practice and do not exceed the limits of validity of the applicant's data or of other data published in the literature. The questions and comments transmitted to the applicant will identify issues that have not been addressed, areas where staff interpretations differ from those given in the SAR, and issues that have not been sufficiently documented to permit the staff to concur in the conclusions reached by the applicant. When possible, the staff should take positions on safety-related issues at this point. The third review phase is the staff evaluation of the applicant's responses to questions raised in the second phase. At the end of the third phase, the staff takes positions on all safety-related issues, either concurring with the applicant's positions or taking more conservative positions as may be necessary in the staff's view to assure the required degree of safety.

Pertinent references, such as published geological reports, professional papers, open file material, university theses, physiographic and geological maps, and aeromagnetic and gravity maps, are ordered from the appropriate sources and reviewed. The general references used extensively by the staff are References 3 and 4. The GEO-Reference File (Ref. 5) is used to identify specific references.

The judgments on acceptance or rejection of the SAR are governed by two criteria: (1) adherence to the Standard Format in identifying and describing the geologic, seismic and man-made features that affect safety of the site; and (2) provision of adequate information and documentation to allow for an independent review of the conclusions made therein.

During the acceptance review the staff decides to what extent consultants such as the USGS, the Corps of Engineers, state geological survey organizations, or other specialists should be involved. The necessary information is then made available to these consultants. Consultants are asked to handle such varied tasks as reviewing the foundation engineering aspects of plants located at sites with complex foundation conditions, verifying an applicant's mineral identifications, or evaluating the adequacy of foundation and slope stability conditions for safety-related dams and dikes.

After docketing, a detailed review of the SAR and relevant references is conducted by the staff and its advisors. Questions and comments are developed from items that have not been adequately addressed by the applicant, those which become apparent during the detailed review, or those which develop from the additional information provided as a result of the acceptance review. These questions (Q-1) usually require the applicant to conduct additional investigations or to supply clarifying information. Many questions result from the reviewer's discovery of references not cited by the applicant that contain conclusions which are in conflict with those made by the applicant. When the applicant provides insufficient data to support his interpretations and conclusions, and there are alternative interpretations in the literature, the staff will request additional investigations. This phase of the review will usually involve meetings with the applicant to clarify questions and allow him to present new data. In addition, during the Q-1 phase, the staff visits the site.

The applicant's responses to Q-1 are reviewed and any remaining issues are settled either by additional questions or by staff positions. A staff position is usually in the form of a requirement to design for a specific condition in a way which the staff considers to be conservative and consistent with the requisites of Reference 1. When all safety issues have been resolved, the staff provides its input to the safety evaluation report (SER).

IV. EVALUATION FINDINGS

The staff's findings for construction permit (CP) reviews will consist of a report summarizing the geology at the site and the pertinent design aspects of the plant. All geologic features that may potentially affect the safety of the plant will be identified, described, and measures taken to deal with them will be given. The seismic design basis will be described.

Operating license (OL) applications are reviewed for any new information developed subsequent to the CP. The review will also determine whether the CP recommendations have been implemented.

A typical CP-stage finding for this section of the SER follows:

"Based on our review of the PSAR materials and our independent review of the relevant published literature, we have concluded that the site is located in the Piedmont tectonic province. The last recognizable regional tectonic event occurred here in

Triassic to Jurassic time (225 - 136 mybp). No Holocene faulting of tectonic origin is known in the province and no capable faults within the meaning of Appendix A to 10 CFR Part 100 have been recognized."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power PLants," Revision 2.

3. M. Gray, R. McAfee, Jr., and C. L. Wolf, eds., "Glossary of Geology," American Geological Institute, Washington (1972).

4. G. V. Cohee (chairman) et al., "Tectonic Map of the United States," U. S. Geological Survey and American Association of Petroleum Geologists (1962).

5. "Geo-Reference: Computerized File of Earth Science Titles," American Geological Institute, Washington.

6. M. W. Higgins, "Cataclastic Rocks," Professional Paper 687, U. S. Geological Survey (1971). (Includes extensive discussion of the terminology of cataclastic rocks.)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.2 VIBRATORY GROUND MOTION

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

The SAB review covers the seismological and geological investigations carried out to establish the acceleration for seismic design of the plant, the procedures and analyses used by the applicant to determine the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) for the site, and the seismic design bases for foundations.

Specific areas of review include; seismicity, relationship of earthquake occurrence to geologic or tectonic characteristics of the region, determination of the earthquake-generating potential of the geologic structures and tectonic provinces in the region, characteristics of seismic wave transmission at the site, and determination of the level and properties of the vibratory ground motion at the site resulting from potential earthquakes in the region.

II. ACCEPTANCE CRITERIA

1. The required investigations are described in 10 CFR Part 100, Section IV(a) of Appendix A. The acceptable procedures for determining the seismic design bases are given in Section V(a) of the same appendix. The seismic design bases are predicated on a reasonable, conservative determination of the safe shutdown earthquake and the operating basis earthquake. As defined in Section III of 10 CFR Part 100, Appendix A, the SSE and OBE are based on consideration of the regional and local geology and seismology and on the characteristics of the subsurface materials at the site and are described in terms of the vibratory ground motion which they would produce at the site. No comprehensive definitive rules can be promulgated regarding the investigations needed to establish the seismic design bases; the requirements vary from site to site.

2. Subsection 2.5.2.1 (Seismicity): The applicant's presentation is accepted when the complete historical record of earthquakes in the region is listed and when all available parameters are given for each earthquake in the historical record. The listing should include all earthquakes MM intensity greater than IV or magnitude greater than 3 which have been reported in all tectonic provinces any parts of which are within 200 miles of the site. A regional-scale map should be presented showing all listed earthquake

epicenters and, in areas of high seismicity, should be supplemented by a larger-scale map showing earthquake epicenters within 50 miles of the site. The following information concerning each earthquake is required whenever it is available: epicenter coordinates, depth of focus, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, source rise time, rupture velocity, total dislocation, fractional stress drop, and any strong-motion recordings; references from which the specified information was obtained should be identified. In addition, any reported earthquake-induced geologic failure, such as liquefaction, landsliding, landspreading, and lurching should be described completely, including the level of strong motion which induced failure and the material properties of the materials. The completeness of the earthquake history of the region is determined by comparison to the historical earthquake data (HED) file (Ref. 4) and other published sources of information (e.g., Refs. 5, 6, 7). When conflicting descriptions of individual earthquakes are found in the published references, a reasonable description which results in the more conservative interpretation of the seismicity is accepted.

3. Subsection 2.5.2.2 (Geologic and Tectonic Characteristics of Site and Region): The applicant's presentation is accepted when all regional geologic structures and tectonic activity which are significant in determining the earthquake potential of the region are identified. Information presented in Section 2.5.1 of the applicant's safety analysis report (SAR) and information from other literature sources (e.g., Refs. 8, 9, 10, 11, 12) dealing with regional tectonics should be developed into a coherent, well-documented discussion to be used as the basis for determining tectonic provinces and the earthquake-generating potential of the identified geologic structures. Specifically, each tectonic province, any part of which is within 200 miles of the site, must be identified. Those characteristics of geologic structure, tectonic history, present and past stress regimes, and seismicity which distinguish the various tectonic provinces and the particular areas within those provinces where historical earthquakes have occurred should be described. Alternative regional tectonic models from available literature sources should be discussed. When several of the alternative models conform equally well with the observed phenomena, the model which results in the more conservative assessment of the earthquake potential at the site is accepted. In addition, in those areas where there are capable faults, the results of the additional investigative requirements described in 10 CFR Part 100, Appendix A, Section IV(a)(8), must be presented. The discussion should be augmented by a regional-scale map showing the tectonic provinces, earthquake epicenters, locations of geologic structures and other features which characterize the provinces, and the locations of any capable faults.

4. Subsection 2.5.2.3 (Correlation of Earthquake Activity with Geologic Structure or Tectonic Provinces): Acceptance is based on the development of the relationship between the relatively short history of earthquake activity and the geologic structures or tectonic provinces of a region. The applicant's presentation is accepted when the earthquakes discussed in Subsection 2.5.2.1 of the SAR are shown to be associated with either geologic structure or a tectonic province. Whenever an earthquake epicenter or concentration of earthquake epicenters can be reasonably correlated with geologic

structure, the rationale for the association should be developed considering the properties of the geologic structure and the regional tectonic model. The discussion should include identification of the methods used to locate the earthquake epicenters, an estimate of their accuracy, and a detailed account which compares and contrasts the geologic structure involved in the earthquake activity with other areas within the tectonic province. Particular attention should be given to determining the capability of faults with which instrumentally-located earthquake epicenters are associated.

The applicant may choose to define tectonic provinces to correspond to subdivisions generally accepted in the literature. A subdivision of a tectonic province is accepted if it can be corroborated on the basis of detailed seismicity studies, tectonic flux measurements, contrasting structural fabric, different geologic history, differences in stress regime, etc. If detailed investigations reveal no significant differences between areas within a tectonic province, the areas should be considered to compose a single tectonic province. The presentation should be augmented by a regional-scale map showing the tectonic provinces, the earthquake epicenters, and the locations of geologic structures and measurements used to define provinces. Acceptance of the proposed tectonic provinces is based on the staff's independent review of the seismicity, tectonic flux (Ref. 31), geologic structure, and stress regime in the region of the site.

5. Subsection 2.5.2.4 (Maximum Earthquake Potential): The applicant's presentation is accepted when the vibratory ground motion due to the maximum credible earthquake associated with each geologic structure or the maximum historic earthquake associated with each tectonic province has been assessed and when the earthquake which would produce the maximum vibratory ground motion at the site has been determined. Earthquakes associated with each geologic structure or tectonic province must be identified. Where an earthquake is associated with geologic structure, the maximum earthquake which could occur on that structure should be evaluated, taking into account such factors as the type of the faulting, fault length, fault displacement, and earthquake history, (e.g., Refs. 14, 15).

In order to determine the maximum earthquake that could occur on those faults which are shown or assumed to be capable, the staff accepts conservative values based on historic experience in the region and specific considerations of the earthquake history, sense of movement, and geologic history of movement on the faults. Where the earthquakes are associated with a tectonic province, the largest historical earthquake within the province should be identified and, whenever possible, the return period for the earthquake should be estimated. Isoseismal maps should also be presented for the most significant earthquakes. The ground motion at the site should be evaluated assuming seismic energy transmission effects are constant over the region of the site and assuming that the largest earthquake associated with each geologic structure or with each tectonic province occurs at the point of closest approach of that structure or province to the site.

The set of conditions describing the occurrence of the earthquake which would produce the largest vibratory ground motion at the site should be defined. If different potential earthquakes would produce the maximum ground motion in different frequency bands, the conditions describing all such earthquakes should be specified. The description of the potential earthquake occurrence is to include the maximum intensity or magnitude and the distance from the assumed location of the potential earthquake to the site. The staff independently evaluates the effects on site ground motion of the largest earthquake associated with each geologic structure or tectonic province. Acceptance of the description of the potential earthquake which would produce the largest ground motion at the site is based on the staff's independent analysis.

6. Subsection 2.5.2.5 (Seismic Wave Transmission Characteristics of the Site): The applicant's presentation is accepted when the seismic wave transmission characteristics (amplification or deamplification) of the materials overlying bedrock at the site are described as a function of the significant frequencies. The following material properties should be determined for each stratum under the site: seismic compressional and shear velocities, bulk densities, soil properties and classification, shear modulus and its variation with strain level, and water table elevation and its variation. In each case, methods used to determine the properties should be described or a cross-reference should be given indicating where in the SAR the description is provided. For each set of conditions describing the occurrence of the maximum potential earthquake, determined in Subsection 2.5.2.4, the type of seismic waves producing the maximum ground motion and the significant frequencies must be determined. For each set of conditions an analysis should be performed to determine the effects of transmission in the site material for the identified seismic wave types in the significant frequency bands.

Where horizontal shear waves produce the maximum ground motion, an analysis similar to that of Schnabel, et al. (Ref. 16) is appropriate. Where compressional or surface waves produce the maximum ground motion, other methods of analysis (Refs. 17, 18) may be more appropriate. However, since the latter techniques are still in the developmental stages and no generally agreed-on procedures can be promulgated at this time, the staff accepts the shear wave model as representative of site amplification. The site amplification determined in this way should be compared with characteristics of site amplification in the epicentral area of the historical earthquake used as the basis for each maximum potential earthquake. If detailed soils investigations have been made in the epicentral area, the amplification analysis should be based on these. Because detailed geologic investigations are generally not available for the epicentral areas of historical earthquakes, several factors should be considered in assessing amplification effects there, including: regional geology and soil conditions, earthquake isoseismal maps, and descriptions of earthquake effects.

7. Subsection 2.5.2.6 (Safe Shutdown Earthquake): The applicant's presentation is accepted when the vibratory ground motion specified for the safe shutdown earthquake is described in terms of the level of acceleration for seismic design and its time history and is as conservative as that which would result at the site from the maximum potential earthquake (determined in Subsection 2.5.2.4) and considering the variations in site transmission effects (determined in Subsection 2.5.2.5). If several different maximum potential earthquakes produce the largest ground motions in different frequency bands (as noted in Subsection 2.5.2.4), the vibratory ground motion specified for the SSE must be as conservative in each frequency band as that for each earthquake, including site transmission effects (as noted in Subsection 2.5.2.5).

The amplitude of acceleration at the ground surface, the effective frequency range, and the duration corresponding to each maximum potential earthquake must be identified. The acceleration is to be expressed as a fraction of the acceleration of gravity (g). Where the earthquake has been associated with a specific geologic structure, the acceleration should be determined using a relation between acceleration, magnitude or fault length and distance from the fault (cf. Refs. 13, 15). Where the earthquake has been associated with a tectonic province, the acceleration should be determined using appropriate relations between acceleration, intensity, epicentral intensity, and distance (e.g., Refs. 19, 20, 21, 24).

Numerous correlations between intensity and acceleration are given in the literature (Refs. 19, 20, 21, 22, 23); several of them are considered acceptable by the staff. The correlation used is accepted if it is conservative when compared to the actual observational data. Acceptance is based on an analysis of the site's seismic energy transmission properties (Ref. 16, or equivalent). Conservatism should be assessed based on consideration of the amplification analysis and in comparison with the actual published data. The staff will generally accept an acceleration for seismic design as being conservative if, when applied at the ground surface, it results in a value at the foundation free field level as large as would be obtained from the empirical relation of the mean of the intensity acceleration values in Reference 23.

Available ground motion time histories for earthquakes of comparable values of magnitude, epicentral distance, and acceleration level should be presented. The spectral content for each potential maximum earthquake should be described; it should be based on consideration of the available ground motion time histories and regional characteristics of seismic wave transmission. The dominant frequency associated with the peak acceleration should be determined either from analysis of ground motion time histories or by inference from descriptions of earthquake phenomenology, damage reports, and regional characteristics of seismic wave transmission.

In some cases, the peak acceleration may not be as significant for engineering design purposes as a sustained acceleration at a lower level. One situation where the sustained acceleration level may differ from the peak acceleration is in proximity to the causative fault of the earthquake. It is appropriate in such cases to define the "reference acceleration for seismic design" as representative of the level of sustained acceleration. The "reference acceleration for seismic design" determined in this section of the applicant's SAR is taken to be the high frequency asymptote to the design response spectrum defined in Reference 2. At this time, the staff is not aware of any published relations between earthquake intensity or magnitude and sustained acceleration. Such relations could be developed from analyses of the response spectra of accelerograph time histories in those areas where magnitude and intensity measurements are also available. In lieu of such studies, the peak accelerations are considered to represent conservative reference accelerations for seismic design. Lower levels of reference acceleration may be justified on a site-specific basis.

The staff's review of proposed reference accelerations for seismic design considers: the proximity of the site to the geologic structure or province with which the potential earthquake is associated, characteristics of acceleration time histories at epicentral distances similar to that of the potential SSE, results of time-dependent spectral analyses of such time histories (cf. Refs. 25, 26), the level and dominant frequency of the peak acceleration, and seismic wave amplitude attenuation as a result of transmission from the source to the site and in the material underlying the site.

The design response spectrum is reviewed under Standard Review Plan (SRP) 3.7.1; however, as noted above there are certain seismological conditions which may require special modifications of the response spectrum. In general, the design response spectrum is acceptable if it is as conservative as the response spectrum from each of the potential earthquakes as described above.

The time duration of strong ground motion is required for analysis of site foundation liquefaction potential and for design of many plant components. The adequacy of the time history for structural analysis is reviewed under SRP 3.7.1. The time history is reviewed in this standard review plan to confirm that it is compatible with the seismological and geological conditions in the site vicinity and with the accepted SSE model. At present, there is no truly adequate model for deterministically computing the time history of strong ground motion from a given source-site configuration. It is, therefore, acceptable to generate the time history record from the design response spectrum for the SSE using the method of Tsai (Ref. 27) or an equivalent method. Total duration of the motion is acceptable when (1) it is as conservative as values determined using the procedure described by Bolt (Ref. 28) for hard rock sites or for analyses where nonstationarity of strong motion time functions is unimportant* and (2) the spectrum of the derived accelerogram is found acceptable in the review under SRP 3.7.1.

8. Subsection 2.5.2.7 (Operating Basis Earthquake): The vibratory ground motion for the OBE should be described with the SSE and the acceleration level at the site specified. The minimum value of the acceleration level for the OBE is currently one-half the reference acceleration for seismic design corresponding to the SSE. For sites in highly seismic regions, mainly in the western United States, the complete description of the OBE, as given in 10 CFR Part 100, Appendix A, Section III(d),

*For sites on sediments or for analyses where nonstationarity is important, more conservative values may be required. See, e.g., Refs. 24 and 30.

is required. In some cases, probability calculations, like those described by Algermissen (Ref. 29), would be helpful in estimating the acceleration level reasonably expected to affect the plant site during the operating life of the plant. Acceptable source regions that can be used as input to these calculations are those geologic structures or tectonic provinces with which historical earthquake activity has been associated. Such descriptions should include the acceleration level of the OBE and a determination of the probability of exceeding that level during the 40-year operating life of the plant.

III. REVIEW PROCEDURES

1. Upon receiving the applicant's SAR, an acceptance review is conducted to determine: compliance with the investigative requirements of 10 CFR Part 100, Appendix A, and conformance with the Standard Format (Regulatory Guide 1.70). The reviewer also identifies any site-specific problems, the resolution of which could result in extended delays in completing the review.

2. After SAR acceptance and docketing, those areas are identified where additional information is required to determine the earthquake hazard and to establish the design acceleration. These are transmitted to the applicant in requests for additional information (Q-1).

3. A site visit is conducted during which the reviewer inspects the foundation conditions, local faulting, and other geologic conditions. During the site visit the reviewer also discusses and clarifies the Q-1 questions with the applicant and his consultants so that it is clearly understood what additional information is required by the staff to continue the review.

4. Following the site visit a revised set of requests for additional information (Q-2), including any additional questions which may have been developed during the site visit, is formally transmitted to the applicant. At the Q-2 stage the review procedure consists mainly of an evaluation of the applicant's response to the Q-1 questions. The reviewer prepares requests for additional clarifying information and formulates positions which may agree or disagree with those of the applicant. These are formally transmitted to the applicant.

5. The safety analysis report and supplements responding to the requests for additional information (Q-1, Q-2) are reviewed to determine that the information presented by the applicant is acceptable according to the critieria described in Section II above. Based on information supplied by the applicant, obtained from site visits, or from staff consultants or literature sources, the reviewer independently identifies the relevant seismotectonic provinces, evaluates the capability of faults in the region, and determines the earthquake potential for each province and each capable fault using procedures noted in Section II above. The reviewer evaluates the vibratory ground motion which the potential earthquakes could produce at the site and defines the safe shutdown earthquake and operating basis earthquake.

IV. EVALUATION FINDINGS

For construction permit (CP) reviews, the findings are included in the staff's safety evaluation report and consist of statements (including or referencing diagrams, maps, etc.) describing the applicant's and the staff's (1) definitions of seismotectonic provinces; (2) evaluations of the capability of geologic structures in the region; (3) determinations of the SSE acceleration at ground surface, reference acceleration for seismic design, time duration of strong ground motion, and any alterations in the design response spectrum based on evaluation of the potential earthquakes; and (4) determinations of the OBE acceleration at ground surface. If the staff's findings are consistent with those of the applicant, staff concurrence is stated; otherwise, a statement requiring use of the staff's findings is made.

For operating license (OL) reviews, the staff's positions from the CP review are referenced and a detailed review of any new data which might affect the seismic design bases is presented.

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1.

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

4. "Historical Earthquake Data File," National Geophysical and Solar-Terrestrial Data Center, National Oceanic and Atmospheric Administration.

5. "Earthquake History of the United States," Publication 41-1, National Oceanic and Atmospheric Administration, U. S. Department of Commerce (1973).

6. S. D. Townley and M. W. Allen, "Description Catalog of Earthquakes of the Pacific Coast of the United States, 1769 to 1928," Bulletin Seismological Society of America, Vol. 29 (1939).

7. W. E. T. Smith, "Earthquakes of Eastern Canada and Adjacent Areas," Publications of the Dominion Observatory (1962).

8. P. B. King, "The Tectonics of North America - A Discussion to Accompany the Tectonic Map of North America Scale 1:5,000,000," Professional Paper 628, U. S. Geological Survey (1969).

9. A. J. Eardley, "Tectonic Divisions of North America," Bulletin American Association of Petroleum Geologist, Vol. 35 (1951).

10. J. B. Hadley and J. F. Devine, "Seismotectonic Map of the Eastern United States," Publication MF-620, U. S. Geological Survey.

11. M. L. Sbar and L. R. Sykes, "Contemporary Compressive Stress and Seismicity in Eastern North America: An Example of Intra-Plate Tectonics," Bulletin Geological Society of America, Vol. 84 (1973).

12. R. B. Smith and M. L. Sbar, "Contemporary Tectonics and Seismicity of the Western United States with Emphasis on the Intermountain Seismic Belt," Bulletin Geological Society of America, Vol. 85 (1974).

13. P. B. Schnabel and H. B. Seed, "Acceleration in Rock for Earthquakes in the Western United States," Report No. EERC 72-2, Earthquake Engineering Center, University of California, Berkeley (1972).

14. J. N. Brune, "Tectonic Stress and Spectra of Seismic Shear Waves from Earthquakes," Journal of Geophysical Research, Vol. 75 (1970).

15. D. Tocher, "Earthquake Energy and Ground Breakage," Bulletin Seismological Society of America, Vol. 48 (1958).

16. P. B. Schnabel, J. Lysmer, and H. B. Seed, "SHAKE-A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," Report No. EERC 72-12, Earthquake Engineering Research Center, University of California, Berkeley (1972).

17. M. D. Trifunac and F. E. Udwadia, "Variations of Strong Earthquake Ground Shaking in the Los Angeles Area," Bulletin Seismological Society of America, Vol. 64 (1974).

18. L. A. Drake, "Love and Rayleigh Waves in Nonhorizontally Layered Media," Bulletin Seismological Society of America, Vol. 62 (1972).

19. N. N. Ambraseys, "Dynamics and Response of Foundation Materials in Epicentral Regions of Strong Earthquakes," Proceedings of the Fifth World Conference on Earthquake Engineering (1973).

20. F. Neumann, "Earthquake Intensity and Related Ground Motion," University of Washington Press (1954).

21. B. Gutenberg and C. Richter, "Earthquake Magnitude, Intensity, Energy, and Acceleration," Bulletin Seismological Society of America, Vol. 46 (1956).

22. N. N. Ambraseys, "The Correlation of Intensity with Ground Motions," Paper presented at Trieste Conference on Advancements of Engineering Seismology in Europe (1974).

23. M. D. Trifunac and A. G. Brady, "On the Correlation of Seismic Intensity Scales with Peaks of Recorded Strong Ground Motion," Bulletin Seismological Society of America, Vol. 65 (1975).

24. O. W. Nuttli, "State-of-the-Art for Assessing Earthquake Hazards in the United States, Report 1, Design Earthquakes for the Central United States," Miscellaneous Paper S-73-1, U. S. Army Engineer Waterways Experiment Station (1973).

25. V. Perez, "Peak Ground Accelerations and Their Effect on the Velocity Response Envelope Spectrum as a Function of Time, San Fernando Earthquake, February 9, 1971," Proceedings of the Fifth World Conference on Earthquake Engineering (1973).

26. V. Perez, "Velocity Response Envelope Spectrum as a Function of Time, for the Pacoima Dam, San Fernando Earthquake, February 9, 1971," Bulletin Seismological Society of America, Vol. 63 (1973).

27. N. C. Tsai, "Spectrum-Compatible Motions for Design Purposes," Journal Engineering Mechanics Division, American Society of Civil Engineers, Vol. 98 (1972).

28. B. A. Bolt, "Duration of Strong Ground Motion," Proceedings of the Fifth World Conference on Earthquake Engineering (1973).

29. S. T. Algermissen and D. M. Perkins, "Techniques for Seismic Zoning: 1. General Considerations and Parameters," Proceedings of the International Conference on Microzonation for Safer Construction Research and Application (1972).

30. L. Esteva and E. Rosenblueth, "Espectros Temblores a Distomicas Moderodas y Grandes," Proceedings of Chilean Conference on Seismology and Earthquake Engineering, Vol. 1, University of Chile (1963).

31. P. St. Amand, "Two Proposed Measures of Seismicity," Bull. Seism. Soc. Am., Vol. 46, pp. 41-45 (1956).

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.3 SURFACE FAULTING

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

SAB reviews information in the applicant's safety analysis report (SAR) related to the existence of a potential for surface faulting affecting the site. The information presented in this section results largely from detailed surface and subsurface geological and geophysical investigations performed in the site and vicinity. The following specific subjects are addressed: the structural and stratigraphic conditions of the site and vicinity (Subsection 2.5.3.1), any evidence of fault offset or evidence demonstrating the absence of faulting (Subsection 2.5.2.2), earthquakes associated with faults (Subsection 2.5.3.3), determination of age of most recent movement on faults (Subsection 2.5.3.4), determination of structural relationships of site area faults to regional faults (Subsection 2.5.3.5), identification and description of capable faults (Subsection 2.5.3.6), and zones requiring detailed fault investigations (Subsection 2.5.3.7).

II. ACCEPTANCE CRITERIA

The data and analyses presented in the SAR are judged acceptable if, as a minimum, they describe and document the information required by References 1 and 2, and other data that are necessary, depending on the complexity of the site. The GEO-Reference File (Ref. 3) is used by the staff as the principal reference guide to judge whether or not all of the pertinent references have been consulted. References 4 through 9 are also used by the staff.

Subsection 2.5.3.1 is considered acceptable if the discussions of the stratigraphy, methods of fault dating, structural geology, and geologic history of the site are complete, compare well with studies conducted by others in the same area, and are supported by detailed investigations performed by the applicant. Site and regional geologic maps and profiles constructed at scales adequate to illustrate clearly the surficial and bedrock geology, structural geology topography, and the relationship of the safety-related foundations of the nuclear power plant to these features should be included in the SAR.

Subsection 2.5.3.2 is acceptable if sufficient surface and subsurface information is provided and supported by detailed investigations, either to confirm the absence of faulting or, if faulting is present, to demonstrate its age. If faulting is present in the site vicinity, it must be defined as to fault geometry, amount and sense of movement, and age of latest movement. In addition to geologic evidence which may indicate faulting, linears interpreted from topographic maps, low-altitude aerial photographs and Environmental Resource Technology Satellite imagery should be documented and investigated. Evidence for absence of faulting is obtained by conducting site surface and subsurface investigations in such detail and areal extent to ensure that undetected offsets are not likely to exist. These investigations will vary in detail according to the geological complexity of the specific site.

Subsection 2.5.3.3 is acceptable if all historically reported earthquakes within five miles of the site or near faults which trend within five miles of the site, as discussed in Section 2.5.2, are evaluated with respect to hypocenter accuracy and source origin. In conjunction with these discussions, a plot of the earthquake epicenters superimposed on a map showing the local tectonic structures as defined in Section 2.5.1 should be provided. Estimated error regions of the earthquake epicenters should be shown.

Subsection 2.5.3.4 is acceptable when every fault, any part of which is within five miles of the site, is investigated in sufficient detail using geological and geophysical techniques of sufficient sensitivity to demonstrate the age of most recent movement. An evaluation of the sensitivity and resolution of the exploratory techniques used should be given.

Subsection 2.5.3.5 is acceptable when a discussion is given of the structural and genetic relationship between site area faulting and the regional tectonic framework. In regions of active tectonism it may be necessary to conduct detailed geological and geophysical investigations to demonstrate the structural relationships of site area faults to regional faults known to be seismically active. Both a theoretical and an observational basis for the conclusions reached should be given.

Subsection 2.5.3.6 is acceptable when it has been demonstrated that the investigative techniques used have sufficient sensitivity to identify all faults greater than 1000 feet in length within five miles of the site and when the geometry, sense of movement, and amount of offset is given for each.

Subsection 2.5.3.7 is judged acceptable if the zone designated by the applicant as requiring detailed faulting investigation is consistent with the description of such a zone in Reference 1.

Subsection 2.5.3.8 must be presented by the applicant if the aforementioned investigations reveal that surface displacement must be taken into account. No nuclear plant has ever been constructed on a capable fault and it is an open question as to whether it is possible to design for surface or near-surface displacement with confidence that the integrity of the safety-related features of the plant would remain intact should displacement occur. It is, therefore, staff policy to recommend relocation of plant sites found to be located on capable faults as determined by the detailed faulting investigation. If in the future it becomes possible to design for surface faulting, it will be necessary to present the design basis for surface faulting and supporting data in considerable detail.

III. REVIEW PROCEDURES

The staff review procedure involves an evaluation to determine that the applicant has followed the investigations outlined in Reference 1. The U.S. Geological Survey (USGS) acts as staff advisor in reviewing this section of the SAR, on a case-by-case basis. On request, the USGS provides expertise in numerous earth science disciplines and often is able to provide first-hand knowledge of the site. A literature search is conducted concerning the regional and local geology. The staff also contacts state geological surveys and universities to obtain additional data.

Generally, the steps that applicants must follow in determining the presence and extent of faulting, and whether near-surface faulting (if present) represents a hazard or not, is outlined in the seismic and geologic siting criteria (Ref. 1). Specific investigative techniques are not given in the criteria, however. The site area must be investigated by a combination of exploratory methods which may include borings, trenching, seismic profiling, geologic mapping, and geophysical investigations. The results of these explorations are cross-compared and evaluated by the staff.

It has been the policy of the staff to encourage applicants to avoid areas where there is a possibility for surface faulting. As the question of whether or not a surface faulting condition exists is so critical in determining whether a particular site is suitable, this consideration is usually addressed very early in the review. Exceptions are those cases in which a fault, the existence of which was previously unknown, is revealed in excavations during construction or is discovered during the course of other investigations in the area.

When faults are identified in the site vicinity, it must be demonstrated that the faults are not capable. This is accomplished by determining the ages of the faults by absolute age dating (radiometric), associating the faulting with regional tectonic activity of known age, stratigraphic or geomorphic evidence, etc. In such cases the staff will carry out limited site observations and investigations of its own such as examinations of excavations, and selecting and dating samples taken from shear zones. Applicants are usually required to trench in the areas where major facilities are to be located.

Subsection 2.5.3.1 is evaluated by conducting an independent literature search and cross-comparing the results with the information submitted in the SAR. The comparison should show that the conclusions presented by the applicant are based on sound data, are consistent with the published reports of experts who have worked in the area, and are consistent with the conclusions of the staff and its advisors. If the applicant's conclusions and assumptions conflict with the literature, substantive investigative results to support those conclusions must be submitted to the staff for review.

Subsection 2.5.3.2 is evaluated by first determining through a literature search that all known evidences of fault offset have been considered in the investigation. The results of the applicant's site investigations are studied and cross-compared in detail to see if there is evidence of existing or potential displacements. If such evidence is found, additional investigations such as field mapping, geophysical investigations, borings, trenching, etc., must be carried out to demonstrate that there is no offset or to define the characteristics of the fault if it does exist.

Subsection 2.5.3.3 is reviewed in conjunction with the consideration of Section 2.5.2. Historic earthquake data derived from the review of Section 2.5.2 are compared with known local tectonic features and a determination is made as to whether any of these earthquakes can reasonably be associated with the local structures. This determination includes an evaluation of the error regions of the earthquake locations. When available, the earthquake source mechanisms should be evaluated with respect to fault geometry.

Subsection 2.5.3.4 is evaluated to determine if the age dating methodology used by the applicant is based on accepted geological procedures. In some cases unusual age dating techniques may be used. When such methods are employed, the staff will require extensive documentation of the technique and may treat it as a generic review item. The resolution of all age dating techniques should be carefully documented.

Subsection 2.5.3.5 is evaluated by determining through a literature search that the applicant's evaluation of the regional tectonic framework is consistent and recognized by experts whose reports appear in the published literature. The conclusions reached by the applicant should be based on sound geologic principles and should explain the available geological and geophysical data. When special investigations are made to determine the structural relationship between faults which pass within five miles of the site and regional faults, the resolution of the investigative techniques should be given.

Subsection 2.5.3.6 is evaluated to determine if a sufficiently detailed investigation has been made by the applicant to define the specific characteristics of all capable faults located within 5 miles of the site. The fault characteristics requiring definition include: length, orientation, relationship of the fault to regional structures; the nature, amount, and geologic history of displacements along the fault; and the outer limits of the fault zone established by mapping fault traces 10 miles along trends in both directions from the point of nearest approach to the site. The staff must be satisfied that the investigation covers a large enough area in sufficient detail to demonstrate that there is little likelihood of near-surface displacement hazards associated with capable faults existing undetected near the site.

Subsection 2.5.3.7 Criteria for determining the zone requiring detailed faulting investigation are clearly outlined in Reference 1. The staff reviews the results of the applicant's faulting investigation together with the published literature. The investigative techniques employed by the applicant are evaluated to ascertain that they are consistent with the state of the art. As part of this phase, experts in specific disciplines are asked to review certain aspects of the investigative program. The results of the investigation are analyzed to determine whether the outer limits of the zone requiring faulting investigation are appropriately conservative. If there are insufficient data to substantiate the outer boundaries, more conservative assumptions are required.

Subsection 2.5.3.8 If the detailed faulting investigations reveal that there is a potential for surface displacement at the site, the staff recommends that the site be moved to an alternate location. In the future, when it may be possible to design a nuclear power plant for displacements, substantial information will be required to support the design basis for surface faulting.

IV. EVALUATION FINDINGS

After completing the review, the staff summarizes its conclusions regarding surface faulting in the SER. If after the staff completes a detailed review of the applicant's investigations and conclusions it has been effectively demonstrated that near-surface displacement cannot occur at the site, the entire section of the SER can be summarized by a statement such as: "The staff concludes that there are no surface or near-surface displacement potentialities existing at the site." If it is determined that surface displacement cannot be precluded, the staff notifies the applicant of its conclusions well in advance of publication of the SER.

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. "GEO-Reference: Computerized File of Earth Science Titles," American Geological Institute, Washington.

4. M. R. Grey, R. McAfee, Jr., and C. L. Wolf, eds., "Glossary of Geology," American Geological Institute, Washington (1972).

5. G. V. Cohee (chairman) et. al., "Tectonic Map of the United States," U. S. Geological Survey and American Association of Petroleum Geologists (1962).

6. State geological maps and accompanying texts.

7. U. S. Geological Survey 7.5- and 15-minute topographic and geologic quadrangle maps.

8. U. S. Department of Agriculture and U. S. Geological Survey aerial photographs.

9. Environmental Resources Technology Satellite photographs.

11/24/75

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Information must be presented by the applicant concerning the stability of all soils and rock supporting the nuclear power plant foundations, under both static and dynamic conditions including the vibratory ground motions associated with the safe shutdown earthquake. Stability of these materials, as they influence the safety of seismic Category I facilities, must be demonstrated. Much of the information discussed in this section may be presented in other sections, in which case it may be cross-referenced rather than repeated here.

The staff review covers the following specific areas:

1. Geologic features (Subsection 2.5.4.1) in the vicinity of the site:
   a. Areas of actual or potential surface or subsurface subsidence, uplift, or collapse.
   b. Zones of alteration or irregular weathering profiles, and zones of structural weakness.
   c. Unrelieved stresses in bedrock.
   d. Rocks or soils that might be unstable because of their minerology, lack of consolidation, water content, or potentially undesirable response to seismic or other events.
   e. History of deposition and erosion, including glacial and other preloading influence on soil deposits.

2. The static and dynamic engineering properties of soil and rock strata underlying the site (Subsection 2.5.4.2) as supported by representative field and laboratory data provided by the applicant.

3. The relationship of the foundations for safety-related facilities and the engineering properties of underlying materials as illustrated on plot plans and profiles (Subsection 2.5.4.3) provided by the applicant.

4. The results of seismic refraction and reflection surveys, including in-hole and cross-hole explorations, as presented in the safety analysis report (SAR) by discussions, plot plans, boring logs, tables, and profiles to support the assumed dynamic soil or rock characteristics (Subsection 2.5.4.4) and stratigraphy.

5. Safety-related excavation and backfill plans and engineered earthwork analyses and criteria (Subsection 2.5.4.5) as illustrated on plot plans and profiles, discussed in the text, and supported by explorations for borrow material and adequate representative laboratory test records.

6. Variable groundwater conditions (Subsection 2.5.4.6) as they affect the loading and stability of structural foundations and foundation materials. This part of the staff review also includes an evaluation of the applicant's plans for dewatering during construction as well as groundwater control throughout the life of the plant.

7. The responses of site soils or rocks to dynamic loading (Subsection 2.5.4.7), including appropriate laboratory and field test records in sufficient number and detail adequate to support conclusions derived from the analyses. Soil-structure interaction analyses are reviewed to assure foundation stability and to confirm the validity of the soil profile model used in the analyses.

8. The liquefaction potential (Subsection 2.5.4.8) and consequences of liquefaction or partial liquefaction of all subsurface soils, including the settlement of foundations. These analyses are based on soil properties obtained by state-of-the-art laboratory and field tests.

9. The earthquake design bases (Subsection 2.5.4.9), as evaluated in detail in Section 2.5.2. These are summarized and cross-referenced in this subsection. The safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) accelerograms and response spectra are evaluated in this subsection in combination with other hazards (floods, etc.) to assess the adequacy of the site materials and the foundation design of the nuclear power plant under dynamic conditions.

10. The results of investigations and analyses conducted to determine foundation stability and settlement under static conditions (Subsection 2.5.4.10).

11. Criteria, references, and design methods (Subsection 2.5.4.11) used in static and seismic analyses, including an explanation of computer programs used in the analyses.

12. Techniques and specifications to improve subsurface conditions (Subsection 2.5.4.12), which are to be used at the site to provide adequate support for foundations.

II. <u>ACCEPTANCE CRITERIA</u>

A thorough evaluation of the foundation engineering aspects of the nuclear plant site as described in the following subsections must be presented along with the basic data supporting all conclusions. Sufficient information must be provided to allow the staff and its advisors to conduct independent analyses.

Subsection 2.5.4.1. The section defining geologic features is acceptable if the discussions, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology are complete and are supported by site investigations sufficiently detailed to obtain an unambiguous representation of the geology. The information must be presented in this subsection or cross-referenced to the appropriate subsection in Section 2.5.1.

Subsection 2.5.4.2. The description of properties of underlying materials are considered acceptable if state-of-the-art methods are used to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. These methods are described, for example, in geotechnical journals published by the American Society of Civil Engineers (Ref. 3), applicable standards published by the American Society for Testing and Materials (Ref. 4), publications of the Institution of Civil Engineers (Ref. 5), and various research reports prepared by universities (Ref. 6). The properties of foundation material must be supported by field and laboratory test records.

Normally, a complete field investigation and sampling program must be performed to define the occurrence and properties of underlying materials at a given site (Ref. 7). Summary tables must be provided which catalog the important test results; test results should be plotted when appropriate. Also, a detailed discussion of laboratory sample preparation must be given when applicable. For critical laboratory tests, full details must be given, e.g., how saturation of the sample was determined and maintained during testing, how the pore pressures changed.

The applicant should provide a detailed and quantitative discussion of the criteria used to determine that the samples were properly taken and tested in sufficient number to define all the critical soil parameters for the site. For sites that are underlain by saturated soils and sensitive clays, it should be shown that all zones which could become unstable due to liquefaction or strain-softening phenomena have been adequately sampled and tested. The relative density of the soils at the site should be determined. The applicant must also show that he has adequately defined the consolidation behavior of the soils as well as their static and dynamic strength. The discussion should explain how the developed data is used in the safety analyses, how the test data is enveloped for design, and why the design envelope is conservative.

Subsection 2.5.4.3. The discussion of the relationship of foundations and underlying materials is acceptable if it includes:

1. A plot plan or plans showing the locations of all site explorations, such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities superimposed thereon.

2. Profiles illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials.

3. Logs of core borings and test pits.

4. Logs and maps of exploratory trenches in the preliminary safety analysis report (PSAR), and geologic maps and photographs of the excavations for the facilities of the nuclear power plant in the final safety analysis report (FSAR).

Subsection 2.5.4.4. The presentation of the dynamic characteristics of soil or rock is acceptable if geophysical investigations have been performed at the site and the results obtained therefrom are presented in detail. Completeness of the presentation is judged by whether or not the exploratory techniques used by the applicant yield unambiguous and useful information, whether they represent state-of-the-art exploration methods (Refs. 3, 4, 7), and whether the applicant's interpretations are supported by adequate field records in the SAR. See also Subsection 2.5.2.3.

Subsection 2.5.4.5. The presentation of the data concerning excavation, backfill, and earthwork analyses is acceptable if:

1. The sources and quantities of backfill and borrow are identified and are shown to have been adequately investigated by borings, pits, and laboratory property and strength testing (dynamic and static) and these data are included, interpreted, and summarized.

2. The extent (horizontally and vertically) of all Category I excavations, fills, and slopes are clearly shown on plot plans and profiles.

3. Compaction specifications and foundation properties are justified by tests and analyses to assure stability.

4. Quality control methods are discussed and the quality assurance program described and referenced.

5. Control of groundwater during excavation to preclude degradation of foundation materials is described and referenced.

Subsection 2.5.4.6. The analysis of groundwater conditions is acceptable if the following are included in this subsection or cross-referenced to the appropriate subsections in Section 2.4:

1. Discussion of critical cases of groundwater conditions relative to the foundation stability of the safety-related facilities of the nuclear power plant.

2. Plans for dewatering during construction.

3. Analysis and interpretation of seepage conditions during construction.

4. Records of field and laboratory permeability tests.

5. History of groundwater fluctuations as determined by periodic monitoring of local wells and piezometers. Flood conditions should also be considered.

Subsection 2.5.4.7. Descriptions of the response of soil and rock to dynamic loading are acceptable if:

1. An investigation has been conducted and discussed to determine the effects of prior earthquakes on the soils and rocks in the vicinity of the site. Evidence of liquefaction and sand cone formation should be included.

2. Field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations) have been accomplished and the data presented and interpreted.

3. Dynamic tests have been performed in the laboratory on samples of the foundation soil and rock and the results included. The section should be cross-referenced with Subsection 2.5.2.3.

The soil-structure interaction analysis should be described in and cross-referenced to this subsection. Soil-structure interaction is reviewed to ensure that:

1. The static and dynamic properties of the soil supporting the structure are properly determined and compatible with the characteristics of the analytical model used to evaluate soil-structure interaction effects.

2. The soil profile has been properly modeled when a two-dimensional finite-element analysis is used, or if a half-space analysis method is used, when foundation moduli are consistent with soil properties and soil profiles at the site.

3. The static and dynamic loads, and the stresses and strains induced in the soil surrounding and underlying the structure are adequately and realistically evaluated in the soil-structure analysis.

4. The consequences of the induced soil stresses and strains, as they influence the support capability of the soil surrounding and underlying the structure, have been conservatively assessed.

5. The integrity of soil-supported or soil-imbedded safety-related facilities (such as Category I pipelines) have been investigated and analyzed to show they are not adversely influenced by the consequences of soil-structure interaction effects on soil supporting capacity.

Subsection 2.5.4.8. If the foundation materials at the site adjacent to and under Category I structures are soils and the water table is above bed rock, then an analysis of the liquefaction potential at the site is required. The need for a detailed analysis is determined by a study of the site stratigraphy, critical soil parameters, and the location of safety-related foundations. Undisturbed samples obtained at the site and appropriate laboratory tests show if the soils are likely to liquefy.

The analysis may be based on cyclic triaxial test data obtained from undisturbed soil samples taken from the critical zones in the site area. The shear stresses induced in the soil by the postulated earthquake should be determined in a manner that is consistent with Standard Review Plan (SRP) 2.5.2. The criterion that should be used to determine when the soil samples tested "liquefied" should be taken as the onset of (initial) liquefaction (defined as the cycle when the pore pressure first equals the confining pressure). If the behavior of the pore pressure is such that strains greater than a few percent occur before initial liquefaction, then the applicant must include the effects of these strains in his assessment of the potential hazards that complete or partial liquefaction could have on the stability and settlement of any Category I structures.

Non-seismic liquefaction (such as that induced by wind and wave action) should be analyzed using state-of-the-art soil mechanics principles.

Subsection 2.5.4.9. The earthquake design basis analysis is acceptable if a brief summary of the derivation of the safe shutdown and operating basis earthquakes (SSE and OBE) is presented and references are included to Subsections 2.5.2.6 and 2.5.2.7.

Subsection 2.5.4.10. The discussions of static analyses are acceptable if the stability of all safety-related facilities has been analyzed from a static stability standpoint including rebound, settlement, and differential settlements under deadloads of fills and plant facilities. Field and laboratory test procedures and results must be included to document soil and rock properties used in the analyses. The applicant must show that the methods of analysis used are appropriate for the local soil conditions.

Subsection 2.5.4.11. The discussion of criteria and design methods is acceptable if the criteria used for the design, the design methods employed, and the factors of safety obtained in the design analyses are described and a list of references presented. An explanation and verification of the computer analyses used and source references should be included.

Subsection 2.5.4.12. The discussion of techniques to improve subsurface conditions is acceptable if plans, summaries of specifications, and methods of quality control are described for all techniques to be used to improve foundation conditions (such as grouting, vibraflotation, dental work, rock bolting, or anchors).

III. REVIEW PROCEDURES

The review process is conducted in a similar manner and concurrent with that described in SRP 2.5.1. The services of the Corps of Engineers are used on selected sites to aid the staff in evaluating the foundation engineering aspects of particular sites.

After acceptance of the SAR, the results of site investigations (such as borings, geologic maps, logs of trenches and pits, permeability test records, results of seismic investigations, laboratory test results, profiles, and plot plans) are studied and cross-checked in considerable detail to determine whether or not the assumptions used in the design are conservative. The design criteria are reviewed to ascertain that they are within the present state-of-the-art. Staff comments and questions at this phase of the review, concerning the information in the SAR are sent to the applicant as first-round questions (Q-1). For those facilities that have complex foundation conditions, where marginal factors of safety have been achieved, or where the applicant proposes to construct a seismic Category I earth or rockfill dam, an independent analysis of the design is performed by the staff or its advisors, the Corps of Engineers. The evaluations conducted by the staff and its advisors may identify additional unresolved items, or reveal that the applicant's investigations and analyses are not complete or sufficiently conservative. Additional information is then requested in a second round of questions (Q-2), or a staff position is taken requiring adoption of a more conservative approach.

The data needed to satisfy the requirements of this section are not usually complete in the early stages. Detailed design investigations are usually still in progress and final conclusions have often not been made. Because of this, the question and answer exchange may not be complete at the Q-2 stage. Most of the open items of Section 2.5 remaining at the time that the safety evaluation report (SER) input is required are in the foundation engineering area because actual site conditions may not be revealed until excavations are opened and construction has begun. Thus, a site visit, in addition to that noted in Section 2.5.1, "Basic Geologic and Seismic Information," is necessary during the post-CP period to examine the foundation materials exposed in excavations during construction. Information and final designs, including confirming tests and revised analyses, are to be submitted in the FSAR.

Generally, the staff is guided by the Seismic and Geologic Siting Criteria (Ref. 1) and the Standard Format (Ref. 2) in reviewing Section 2.5.4.

Following is a brief description of the review procedures conducted by the staff in evaluating the foundation engineering aspects of nuclear power plant sites.

Subsection 2.5.4.1. Geologic features are evaluated by conducting (the staff and the U.S. Geological Survey) an independent literature search and comparing these results with the information included in the applicant's SAR. References used in reviewing this subsection include published or unpublished reports, maps, geophysical data, construction records, etc., by the USGS, other federal agencies, state agencies, and private companies (such as oil corporations and architect-engineering firms). In conjunction with the literature search, the staff and its USGS advisors review the geological investigations conducted by the applicant. Using the references listed at the end of this section and other sources, the following questions are considered in detail:

1. Are the exploratory techniques used by the site investigator representative of the present state-of-the-art? Do the samples represent the in situ soil conditions?

2. Do the applicant's investigations provide adequate coverage of the site area and in sufficient detail to define the specific subsurface conditions with a high degree confidence?

3. Have all areas or zones of actual or potential surface or subsurface subsidence, uplift, or collapse; deformation, alternation or structural weakness; unrelieved stresses in bedrock; or rocks or soils that might be unstable because of their physical or chemical properties been identified and adequately evaluated?

Subsection 2.5.4.2. Properties of underlying materials are evaluated to determine whether or not the investigations performed (including laboratory and field testing) were sufficient to justify the soil and rock properties used in the foundation design analyses.

To determine whether sufficient investigations were performed, the staff carefully reviews the criteria developed and used by the applicant in laying out his boring, sampling and testing program and evaluates the effectiveness of the program in defining the specific foundation conditions at the site and assuring that all critical conditions have been adequately sampled and tested. If suitable criteria have not been developed and used by the applicant, the staff develops appropriate criteria, using the data given in the SAR, and determines if sufficient investigation and testing have been carried out. If criteria are given, the staff reviews them to determine if they are appropriate and have been implemented.

If it is the staff's judgment that the applicant's investigations or testing are insufficient, additional investigations will be required. The final conclusion is based on professional judgment, considering the complexity of the site subsurface conditions. As part of the review, the staff must ascertain, often with the help of the Corps of Engineers, that state-of-the-art laboratory and field techniques and equipment are employed in determining the material properties.

Subsection 2.5.4.3. Plot plans and profiles are reviewed by comparing the subsurface materials with the proposed locations (horizontal and vertical) of foundations and walls of all seismic Category I facilities. The profiles and plot plans are cross-checked in detail with the results of all subsurface investigations conducted at the site to ascertain that sufficient exploration has been carried out and to determine whether or not the interpretations made by the investigators are valid and the foundation design assumptions contain adequate margins of safety.

Subsection 2.5.4.4. Staff evaluation consists of a detailed review of all geophysical explorations conducted at the site, including seismic refraction, reflection, and in-hole surveys and magnetic and gravity surveys. Expertise within the USGS regarding specific techniques is drawn upon in this review. Logs of core borings, trenches, and test pits are reviewed and compared with data from the seismic surveys and other geophysical explorations. Results must be consistent or additional investigations are required, or the applicant must use the most conservative values. Following the PSAR review and during the FSAR review the staff compares conditions as mapped in the open excavations with interpretations and assumptions derived during the investigation program.

Subsection 2.5.4.5. Excavations, backfill, and earthwork are evaluated by the staff as follows:

1. The investigations for borrow material, including boring and test pit logs, and compaction test data are reviewed and judged as to their adequacy.

2. Laboratory dynamic and static records of tests performed on samples compacted to the design specifications are reviewed to ascertain that state-of-the-art criteria are met.

3. Analyses and interpretations are reviewed to assure that static and dynamic stability requirements are met.

4. Excavation and compaction specifications and quality control procedures are reviewed to ascertain conformance to state-of-the-art conservative standards.

Subsection 2.5.4.6. Groundwater conditions as they affect foundation stability are evaluated by studying the applicant's records of the historic fluctuations of groundwater at the site as obtained by monitoring local wells and springs and by analysis of piezometer and permeability data from tests conducted at the site. The applicant's dewatering plans during and following construction are also reviewed. Adequacy of these plans is evaluated by comparing with the results of the groundwater investigations and by professional judgment of groundwater and soil conditions at the site.

Subsection 2.5.4.7. Response of soil and rock to dynamic loading and soil-structure interaction is evaluated by a detailed study of the results of the investigations and analyses performed. Specifically, the effects of past earthquakes on site soils or rocks (a requirement in SRP 2.5.2) are determined. The data from core borings, from geophysical investigations, and from dynamic laboratory tests such as sonic and cyclic triaxial tests on undisturbed samples are evaluated. The object of the staff review is to ascertain that reasonably conservative dynamic soil and rock characteristics are used in the design and analyses and that all the significant soil and rock strata have been considered in the analyses. In some cases, independent analyses and interpretations are carried out as outlined in SRP 2.5.2, or as required to verify the liquefaction analysis discussed in Subsection 2.5.4.8.

Subsection 2.5.4.8. Liquefaction potential is reviewed by a study of the boring logs and profiles to determine if any of the site soils could be susceptible to liquefaction. The results of standard penetration tests and undisturbed sampling performed in exploration borings are examined and, when appropriate, related to the liquefaction potential of in situ soils.

If it is determined that there are liquefaction-susceptible soils beneath the site, the applicant's site exploration methods, laboratory test program, and analyses are reviewed for adequacy and reasonableness of results. The analysis submitted by the applicant is reviewed in detail and compared to an independent study performed by the staff. As a minimum, the staff study consists of:

1. A careful review of the cyclic triaxial test data to insure that appropriate samples were obtained from critical, liquefiable zones.

2. Confirmation that an adequate number of samples were properly tested and that the test results account for the natural variation in different samples as well as define the cyclic resistance to liquefaction of the soils.

3. An assessment of the liquefaction potential using a conservative envelope of the test data submitted.

4. A calculation of the stress induced by the earthquake that has been arrived at by an envelope of critical conditions calculated for the site based on possible variations in the properties of the soil strata.

5. Assurance that conservative ranges of relative density of the soils are estimated.

The applicant's estimates of the "safety factor" obtained from his analysis is compared to that estimated by the staff. (The applicant's plans to "fix" the liquefaction condition, usually by excavation and backfill, vibraflotation, or chemical grouting is evaluated as discussed in Subsections 2.5.4.5 and 2.5.4.12.)

6. An assessment of past earthquake settlements due to partial liquefaction using state-of-the-art techniques.

7. An assessment of non-seismic liquefaction based on state-of-the-art techniques.

Subsection 2.5.4.9. The in-depth staff evaluation of the safe shutdown and operating basis earthquakes is contained in SRP 2.5.2. The staff's evaluation of the amplification characteristics of specific soils and rocks beneath the site as determined by procedures discussed in that section and in Subsections 2.5.4.2, 2.5.4.4, and 2.5.4.7 are summarized and cross-referenced herein.

The review of Subsection 2.5.4.9 concentrates on determining its consistency or inconsistency with other subsections. Cross-referencing with other sections is expected.

Subsection 2.5.4.10. Static analyses of the bearing capacity and settlement of the supporting soils under the loads of fills, embankments, and foundations are evaluated by conventional, state-of-the-art methods (Ref. 8). In general, the evaluation procedure includes:

1. Determining whether or not the soil and rock properties used in the analyses represent the actual site conditions beneath the plant facilities. The site investigation, sampling, and laboratory test programs must be adequate for this evaluation.

2. Determining whether or not the methods of analysis are appropriate for the earthworks, foundations, and soil conditions at the site.

3. Determining whether or not the bearing capacity, settlement, differential settlement, and tilt estimates indicate conservative and tolerable behavior of the plant foundations when these values are compared to design criteria and quality assurance specifications.

4. Evaluation of particularly complex cases on the basis of accepted principles and techniques as supplemented by case histories and confirmatory measurement and analysis programs (Ref. 8).

Subsection 2.5.4.11. Criteria and design methods, including construction control and monitoring systems, are evaluated on the basis of conservative accepted practice for similar facilities. Site exploration, sampling, testing, and interpretation are judged with respect to completeness, care and technique, meaningful documentation, performance records for similar projects, published guidelines, and state-of-the-art practice. However, unconventional or research-oriented tests and interpretations are encouraged whenever such work aids or supplements conventional practices. Design criteria and methods are compared to similar standards published or utilized by public agencies such as the U. S. Navy Department, U. S. Army Engineers, and U. S. Department of the Interior. Design safety features, the applicant's proposed confirmatory tests and measurements, and monitoring of performance for safety-related foundations and earthworks are reviewed and evaluated on a case-by-case basis.

Subsection 2.5.4.12. Techniques to improve subsurface conditions are evaluated by reviewing the applicant's specifications and techniques for performance and quality control for such activities as grouting, excavation and backfill, vibraflotation, rock bolting, and anchoring. Confirmatory data should be contained in the FSAR.

IV. EVALUATION FINDINGS

If the evaluation by the staff, on completion of the review of the foundation engineering aspects of the plant site, confirms that of the applicant, the conclusion in the SER states that the investigations performed at the site are adequate to justify the soil and rock characteristics used in the design, and that the design analyses contain adequate margins of safety for construction and operation of the subject nuclear power plant. Staff reservations about any portion of the applicant's analyses are stated, in sufficient detail to make clear the precise nature of the staff concern.

A typical staff SER finding follows:

> "The site is located in the Piedmont at an elevation of +395 feet mean sea level (msl). Exploratory borings have been made and refraction and reflection seismic surveys conducted to establish the stratigraphy of the site. Additionally, undisturbed samples of representative soils and core borings have been obtained to evaluate the characteristics of the foundation materials; close-centered cross-hole seismic tests have been conducted to determine the elastic properties of these materials. Groundwater at the site varies from +375 to +380 feet msl.

"The area has been exposed to subaerial weathering and erosion since middle Mesozoic time, and a deep weathering profile has developed. The depth of weathering depends on the location and degree of jointing, orientation of schistocity, and composition of the parent rock.

"The applicant has categorized the foundation material into three zones according to the degree of weathering:

(a) Zone 1 contains residual soil derived from severely weathered slate. The soil is a sandy, silty clay containing slate and quartz fragments. Decomposed to severely weathered slate is also present. The slate still retains the original rock structure, although it is soft and partly friable. Quartz veins within the slate are extremely fractured. Seismic compression (P) and shear (S) wave velocities exceed 4000 ft/sec and 1800 ft/sec, respectively. Zone 1 ranges in thickness from less than 20 feet to more than 50 feet.

(b) Zone 2 consists of moderately weathered slate and varies from 15 to 60 feet thick. P and S wave velocities generally exceed 6500 ft/sec and 2500 ft/sec, respectively.

(c) Zone 3 contains slightly weathered to unweathered slate and is encountered at depths of 60 to 90 ft below ground surface.

"The site area will be leveled to about elevation +390 feet msl, and containments will be founded on a thick, reinforced concrete mat on slightly weathered slate. The outer perimeter will also be on a reinforced concrete mat. The reactor service building between the reactors and the control building will be on mats at elevation +385 feet msl on slightly to moderately weathered rock. The turbine generators will be founded on moderately weathered rock at elevation +380 feet msl. The diesel generator building, reactor plant component air-cooled heat exchanger enclosures, and the CACS air-cooled heat exchanger will be founded on either concrete footings or continuous footings (grade beams) at +385 feet msl, on moderately weathered slate. All piping will be entrenched and bedded in moderately to severely weathered slate. Allowable bearing capacities from laboratory tests and field plate tests for Zone 1, Zone 2, and Zone 3 materials are 4, 10, and 25 tons per square foot, respectively.

"Settlement and differential settlement of safety-related facilities will be less than one inch.

"The applicant states that severely weathered or soft zones will be excavated and replaced with lean concrete. This procedure will also be followed wherever severe weathering extends along joints, schistocity, etc., below the base of the foundations; this material will be excavated to a depth 1-1/2 times the width of the zone and backfilled with concrete.

"All backfill under structures will be concrete. Category I backfill around structures will either be concrete or compacted granular backfill. If granular soil is used, the applicant will place the backfill at 95 percent of maximum as determined by Modified Proctor. These backfill criteria are acceptable.

"Suitable borrow material for dikes, dams and impervious linings are available for the ultimate heat sink ponds. The applicant's tests on these materials and the construction criteria to be followed ensure that leakage, piping and cracking hazards of these vital earthworks are minimal. Filters, blanket drains, relief wells, piezometers and settlement monuments will assure the reliable performance of the ultimate heat sink water-retention facilities.

"The applicant has estimated that the appropriate acceleration to use as input to Regulatory Guide 1.60 spectrum at foundation level is 0.12g for the safe shutdown earthquake (SSE). The operating bases earthquake (OBE) value is taken as 0.06g. The applicant has performed a site-dependent analysis to estimate the site amplification effects and found that the weathered rock would amplify the motion. An acceleration level of 0.17g for the SSE will be used for those structures founded on weathered rock. The synthetic time history used for seismic design of Category I earth dams and for liquefaction assessment envelops the response spectra for the site and has a conservative duration.

"The seismic design of Category I buried piping is adequate to safely resist static soil pressures and displacements, dynamic soil pressures, strains induced by ground and structure movements, and pump shutdown pressures.

"Soil-structure interaction will be evaluated based on the Reissner solutions for a rigid foundation on an elastic half-space. Appropriate foundation moduli and damping values were determined by laboratory tests and field seismic investigations. This approach for interaction effects has been shown to be realistic and has staff concurrence. Peak foundation pressures during the SSE will be less than 20 percent of the allowable pressures on the weathered slate.

"Based on the results of the applicant's investigations, laboratory and field tests, analyses, and criteria for design and construction, we and our consultants conclude that the site and the plant foundations will be adequate to safely support the planned nuclear power plant and that safety-related earthworks will perform their functions reliably."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. Journal of the Geotechnical Engineering Division, Proceedings of the American Society of Civil Engineers.

4. Book of ASTM Standards and Special Technical Publications, American Society for Testing and Materials.

5. Geotechnique, The Institution of Civil Engineers, London.

6. Earthquake Engineering Research Center, University of California, Berkeley.

7. M. Juul Hvorslev, "Subsurface Exploration and Sampling of Soils for Civil Engineering Purposes," Waterways Experiment Station, U. S. Army Corps of Engineers, November 1949.

8. GEODEX INTERNATIONAL, Soil Mechanics Information Service, Sonoma, California.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 2.5.5 STABILITY OF SLOPES

REVIEW RESPONSIBILITIES

Primary - Site Analysis Branch (SAB)

Secondary - None

I. AREAS OF REVIEW

Information, including analyses and substantiation, must be presented in the applicant's safety analysis report (SAR) and reviewed by the staff concerning the stability of all earth and rock slopes both natural and man-made (cuts, fills, embankments, dams, etc.) whose failure, under any of the conditions to which they could be exposed during the life of the plant, could adversely affect the safety of the plant. The following subjects must be evaluated using the applicant's data in the SAR and information available from other sources: slope characteristics (Subsection 2.5.5.1); design criteria and design analyses (Subsection 2.5.5.2); results of the investigations including borings, shafts, pits, trenches, and laboratory tests (Subsection 2.5.5.3); properties of borrow material, compaction and excavation specifications (Subsection 2.5.5.4).

II. ACCEPTANCE CRITERIA

The information in the SAR must be in compliance with the Standard Format (Ref. 2) and the Seismic and Geologic Siting Criteria (Ref. 1). This section of the SAR is judged acceptable if the information presented is sufficient to demonstrate the dynamic and static stability of all slopes whose failure could adversely affect, directly or indirectly, safety-related structures of the nuclear plant or pose a hazard to the public. The emergency cooling water source is of particular interest with regard to slope stability. The secondary source of emergency cooling water should survive the operating basis earthquake (OBE) and design basis flood. Completeness is determined by the ability to make an independent evaluation on the basis of information provided by the applicant.

Subsection 2.5.5.1. The discussion of slope characteristics is acceptable if the subsection includes:

a. Cross sections and profiles of the slope in sufficient quantity and detail to represent the slope and foundation conditions.

b. A summary and description of static and dynamic properties of the soil and rock comprising seismic Category I embankment dams and their foundations, natural and cut slopes, and all soil or rock slopes whose stability would directly or indirectly affect safety-related and Category I facilities. The text should include a complete discussion of procedures used to estimate, from the available field and laboratory data, conservative soil properties and profiles to be used in the analysis.

c. A summary and description of groundwater, seepage, and high and low groundwater conditions.

Subsection 2.5.5.2. The discussion of design criteria and analyses is acceptable if the criteria for the stability and design of all seismic Category I slopes are described and valid static and dynamic analyses have been presented to demonstrate that there is an adequate margin of safety. A number of different methods of analysis are available in the literature. Computer analyses should be verified by manual methods.

To be acceptable the static analyses should include calculations with different assumptions and methods of analysis to assess the following factors:

1. The uncertainties with regard to the shape of the slope, boundaries of the several types of soil within the slope and their properties, the forces acting on the slope, and pore pressures acting within the slope.

2. Failure surfaces corresponding to the lowest factor of safety.

3. The effect of the assumptions inherent in the method of analysis used.

4. Adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. In general, safety factors related to the slope hazard are needed; however, actual values depend somewhat on the method of analysis, on the assumptions concerning the soil properties, on construction techniques, and on the range of material parameters.

To be acceptable, the dynamic analyses must account for the effect of cyclic motion of the earthquake on soil strength properties. Actual test data are needed for both the in situ soils as well as for any materials used in the construction of dams or embankments. As discussed above, the various parameters, such as geometry, soil strength, modeling method (location and number of elements (mesh) if a finite-element analysis is used), and hydrodynamic and pore pressure forces, should be varied to show that there is an adequate margin of safety. Where liquefaction is possible, major dam foundation slopes and embankments should be analyzed by state-of-the-art finite-element or finite-difference methods of analysis. Where there are liquefiable soils, changes in pore pressure due to cyclic loading must be considered in the analysis to assess not only the potential for liquefaction but also the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake stability of the slopes.

Subsection 2.5.5.3 In discussing the soil investigations, the applicant should describe the borings and soil testing that was carried out for slope stability studies and dam and dike analyses. The test data, which must meet the criteria set forth in Sections 2.5.1 and 2.5.4, could be presented in those sections and referenced in this subsection. Because dams, dikes, and natural or cut slopes are often remote from the main plant area, additional exploration, tests, and analyses for these areas should be presented in this subsection.

Subsection 2.5.5.4 Compaction specifications should be discussed in this section. The applicant should describe the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Planned construction procedures and control of earthworks should be described. To be acceptable, the information must be given as discussed in Subsection 2.5.4.5. Some of this information could be presented in Subsection 2.5.4.5. Because dams, dikes, and other earthworks are often remote from the main seismic Category I structures, it is necessary to complete this information in this subsection. Quality control techniques and requirements during and following construction must also be discussed and referenced to quality assurance sections of the SAR.

III. REVIEW PROCEDURES

The review process is conducted in a similar manner and concurrent with that described in Standard Review Plans (SRP) 2.5.1, 2.5.2, and 2.5.4. The Corps of Engineers is the principal advisor to the staff regarding foundation engineering and slope stability analyses, particularly in the evaluation of safety-related and seismic Category I earthworks, earth and rock-fill dams, dikes, and reservoirs. Standard references used by the staff are listed in Section V of this SRP.

An acceptance review is conducted to determine if the Standard Format (Ref. 2) has been adhered to and to judge whether or not the information presented is sufficient to permit an in-depth review of the safety of the proposed facility. After acceptance of the SAR, the results of site investigations such as borings, maps, logs of trenches, permeability test records, results of seismic investigations, laboratory test results, profiles, plot plans, and stability analyses are studied and cross-checked in considerable detail to determine whether or not the assumptions and analyses used in the design are conservative. The degree of conservatism required depends upon the type of analysis used, the reliability of parameters considered in the slope stability analysis, the number of borings, the sampling program, the extent of the laboratory test program, and the resultant safety factor. In general, the applicable soil strength data should be conservatively selected for the various possible soil profiles and slope conditions. For lower safety factors, several soil profiles should be analyzed to insure that reasonable ranges of soil properties have been considered. Other factors such as flood conditions, pore pressure effects, possible erosion of soils, and possible seismic amplification effects should be conservatively assessed.

The design criteria and analyses are reviewed to ascertain that the techniques employed are appropriate and represent the present state-of-the-art. Staff comments and questions at this phase of the review, concerning the information in the SAR, are sent to the applicant as first-round questions (Q-1). An independent analysis of the design of safety-related earth or rock-fill embankments is performed by the staff's advisors, the Corps of Engineers, or by the staff as deemed necessary. The Corps also evaluates natural or cut slopes, as required, on a case-by-case basis. The evaluations conducted by the staff and its advisors may identify additional unresolved items or reveal that the applicant's analyses are not conservative. Additional information is then requested in a second round of questions (Q-2), or a staff position is taken requiring conformance to a more conservative approach.

After completing the review, if the staff's conclusions are consistent with those reached by the applicant, these conclusions are summarized in the safety evaluation report (SER) or in a supplement to the SER. In the event that the applicant's investigation and design are not judged to be sufficiently conservative, a staff position is stated and the applicant is asked to further substantiate his position by additional investigations or monitoring, to demonstrate that a failure of the slopes in question will not harm the safety functions of the plant, or to concur in the staff position.

The data needed to satisfy the requirements of this section are often incomplete in the early stages. However, sufficient field and laboratory data should be presented and conservatively interpreted to allow a realistic assessment of the safety of proposed slopes and supporting foundations. Detailed design investigations are usually still in progress and final design conclusions have often not been made. Because of this, the question and answer exchange is not generally complete at the Q-2 stage. Most of the open items of Section 2.5 remaining at the time that the safety evaluation report (SER) input is required are in the foundation engineering and slope stability areas because actual conditions may not be revealed until excavations are opened; site visits conducted after construction permit (CP) issuance are therefore necessary.

All natural safety-related slopes are examined during at least one of the two site visits required of the staff. Because excavated slopes or embankments are not usually constructed until after a construction permit has been granted, detailed as-built documentation of these slopes and embankments, as well as complete stability and safety analyses, are necessary in the FSAR.

Following is a brief description of the review procedures conducted by the staff in evaluating the slope stability aspects of nuclear power plant sites.

Subsection 2.5.5.1. Plot plans, cross sections, and profiles of all safety-related slopes in relation to the topography and physical properties of the underlying materials are reviewed and compared with exploratory records to ascertain that the most critical conditions have been addressed and that the characteristics of all slopes have been defined. The soil and rock test data are reviewed to insure that there is sufficient relevant test data to verify the soil strength characteristics assumed for the slopes, dikes, and dams under analysis. The evaluation is to some extent a matter of engineering judgment; however, if the safety factors resulting from the analysis are not appropriate to the hazards posed by a slope failure and other than clearly conservative soil properties and profiles were used, the applicant is required to obtain additional data to verify his assumptions, or to show that, even if the worse possible conditions are assumed, there is an adequate margin of safety. With respect to seismic analysis this subsection and Subsection 2.5.5.2 are reviewed concurrently because different methods of analysis may involve different approximations, assumptions, and soil properties.

In addition to generic state-of-the-art literature, other potential sources of information are those containing design, construction, and performance records of natural slopes, excavation slopes, and dams that may have been constructed in the general vicinity of the nuclear power plant. Examples of such documents are design memoranda and construction reports regarding nearby projects of public agencies such as the Corps of Engineers, the Tennessee Valley Authority, the Bureau of Reclamation, and private construction contractors or architect-engineers.

Subsection 2.5.5.2. The criteria, design techniques, and analyses are evaluated by the staff to ascertain that:

1. Appropriate state-of-the-art methods have been employed.

2. Conservative assumptions regarding soil and rock properties have been used in the design and analysis of slopes and embankments as discussed above in Subsection 2.5.5.1.

3. Appropriately conservative margins of safety have been incorporated in the design.

The criteria and design methods used by the applicant are reviewed to ascertain that state-of-the-art techniques are being employed. The design analyses are reviewed to be sure that the most conservative failure approach has been used and that all adverse conditions to which the slope might be subjected have been considered. Such conditions include ground motions from the safe shutdown earthquake, settlement, cracking, flood or low-water steady-state seepage, sudden drawdown of an adjacent reservoir, or a reasonable assumption of the possible simultaneous occurrence of two natural events such as an earthquake and flood. The review is also concerned with determining whether or not the soil and rock characteristics derived from the investigations described in Subsection 2.5.5.3 have been completely and conservatively incorporated into the design. When marginal factors of safety are indicated by the independent analyses performed by the staff and its consultants, additional substantiation and refinement is required or the applicant must use more conservative assumptions.

No single method of analysis is entirely acceptable for all stability assessments; thus, no single method of analysis can be recommended. Relevant manuals issued by public agencies (such as the U.S. Navy Department, U.S. Army Corps of Engineers, and U.S. Bureau of Reclamation) are often used in reviews to ascertain whether the analyses performed by the applicant are reasonable. Many of the important interaction effects cannot be included in current analyses and must be treated in some approximate fashion. Engineering judgment is an important factor in the staff's review of the analyses and in assessing the adequacy of the resulting safety factors.

If the staff review indicates that questionable assumptions have been made by the applicant or some non-standard or inappropriate method of analysis has been used, then the staff or its consultant may model the dam or slope in a manner which it feels is more consistent with the data and perform an independent analysis.

During the operating license review, all open items requiring resolution, including construction data and as-built analyses, settlement records, piezometer records, and absence of seepage, that support the adequacy and safety of the design is reviewed by the staff.

Subsection 2.5.5.3. A comprehensive program of site investigations including borings, sampling, geophysical surveys, test pits, trenches, and laboratory and field testing must be carried out by the applicant to define the physical characteristics of all soil and rock beneath safety-related and seismic Category I slopes, and borrow material that is to be used to construct safety-related dams, fills, and embankments. The staff reviews these investigations to ascertain that the program has been adequate to define the in situ and earthwork soil and rock characteristics. The decision as to the adequacy of the investigation program is based on the methods discussed in Section 2.5.4.

Subsection 2.5.5.4. The preliminary specifications and quality control techniques to be used during construction are reviewed by the staff to ascertain that all design conditions are likely to be met. During this part of the review the following are among those subjects reviewed for adequacy:

1. Proposed construction dewatering plan to ensure that it will not result in damage either to the natural or engineered foundation materials or to the structural foundation.

2. The excavation plan to remove all unsuitable materials from beneath the foundations and the quality control procedures which establish suitable materials.

3. The techniques and equipment to be used in compacting foundation and embankment materials.

4. The quality control and testing program to provide a high level of assurance that:

    a. The selected borrow material is as good and as relatively homogeneous as anticipated from the investigation program.

    b. The compacted foundation soil meets design specifications.

5. The techniques for improving the stability of natural slopes such as drainage, grouting, rock bolting, and applying gunite.

6. The plans for monitoring during and after construction to detect occurrences that could detrimentally affect the facility. Such monitoring includes periodic examination of slopes, survey of settlement monuments, and measurements of local wells and piezometers.

IV. EVALUATION FINDINGS

The staff's conclusions regarding the stability of slopes are summarized in the safety evaluation report (SER) or in a supplement to the SER. The following is an example:

"Both natural and man-made slopes exist at the site. At the plant site, which is located several hundred feet from the Green Valley and about 280 feet above the level of Jones Pond, the slope is relatively gentle for about 250 feet west of the westernmost Category I structures, then steepens, attaining an angle of more than 45° near the bottom of the valley wall. Major structural trends, schistocity, and one of the predominant joint trends are nearly perpendicular to the slope. A second predominant joint set is nearly parallel to the river and dips to the southwest, but no slope movements have apparently affected the valley walls in the vicinity of the site. Seven other joint trends were detected by the applicant. These joint sets are reported to be moderately spaced and discontinuous. The applicant has drilled several exploratory holes and cored others to assess the natural slope characteristics and groundwater regime. Even though the natural slopes are some distance from safety-related plant facilities and slope failures are not obvious safety hazards, the applicant has performed stability analyses of these slopes under safe shutdown earthquake (SSE) conditions. The minimum computed safety factor was 1.6 using conservative slope and material parameters.

"Man-made earth slopes related to the safety of the plant include excavation cuts for the ultimate heat sink canal and dams and dikes for the ultimate heat sink storage pond. An extensive investigation and test program has determined all the significant characteristics and properties of cut slopes and fill embankments. Earthwork compaction criteria, construction control, and select fill materials are consistent with high-quality water-retention facilities. Conservative stability analyses of these slopes under SSE conditions indicated minimum safety factors of 1.5.

"Based on the results of the applicant's investigations, laboratory and field tests, analyses, and criteria for design and construction, we and our consultants conclude that natural and man-made slopes will remain stable under SSE conditions and that safety-related earthworks will function reliably."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

3. "ASCE Soil Mechanics and Foundation Division Conference on Stability and Performance of Slopes and Embankments, August 22-26, 1966." Published in J. Soil Mech. and Found., ASCE, Vol. 93 (1967).

4. P. Chakrabarti and A. K. Chopra, "A Computer Program for Earthquake Analysis of Gravity Dams Including Hydrodynamic Interaction," Report No. EERC 73-7, Earthquake Engineering Research Center, Univ. of California, Berkeley (1973).

5. I. M. Idress, J. Lysmer, R. Hwang, and H. B. Seed, "Quad-4 a Computer Program for Evaluating the Seismic Response of Soil Structures by Variable Damping Finite Element Procedures," Report No. EERC 73-16, Earthquake Engineering Research Center, Univ. of California, Berkeley (1973).

6. Bureau of Reclamation, "Earth Manual," First Edition, U. S. Dept. of Interior (1968).

7. K. Stagg and O. Zienkiewicz, "Rock Mechanics in Engineering Practice," John Wiley & Sons (1968).

8. Shannon & Wilson, Inc. and Agbabian-Jacobsen Associates, "Soil Behavior Under Earthquake Loading Conditions - State-of-the-Art Evaluation of Soil Characteristics for Seismic Response Analyses," U. S. Atomic Energy Commission Contract W-7405-eng-26, January 1972.

9. F. H. Kulhawy, J. M. Duncan, and H. B. Seed, "Finite Element Analysis of Stresses and Movements in Embankments During Construction," Report No. TE-69-4, U. S. Army Engineers Waterways Experiment Station, Vicksburg (1969).

10. K. Terzaghi and R. B. Peck, "Soil Mechanics in Engineering Practice," 2nd ed., John Wiley & Sons (1967).

11. Corps of Engineers, "Engineering and Design Stability of Earth and Rock-Fill Dams," Manual N. EM 1110-2-1902, Office of the Chief of Engineers, Dept. of the Army (1970).

12. J. W. Snyder, "Pore Pressures in Embankment Foundations," Report S-28-2, U. S. Army Engineers Waterways Experiment Station, Vicksburg (1968).

13. Corps of Engineers, "Procedures for Foundation Design of Buildings and Other Structures (Except Hydraulic Structures)," Tech. Report TM 5-818-1 (formerly EM 1110-345-147), Office of the Chief of Engineers, Dept. of the Army (1965).

14. GEODEX, INTERNATIONAL, Soil Mechanics Information Service, Sonoma, California.

15. Department of the Navy, "Soil Mechanics, Foundations, and Earth Structures," NAVFAC DM-7, March 1971.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Nuclear power plant systems and components important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

The RSB reviews the applicant's classification system for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety, and the assignment by the applicant of quality groups to those sections of systems required to perform safety functions. Where required, specific information or assistance may be required from the EICSB to review electrical and instrumentation systems needed for functioning of plant features important to safety. This review is done for both construction permit (CP) and operating license (OL) applications. Excluded from this review are: structures; parts such as pump motors, shafts, seals, impellers, packing, and gaskets; containment; fuel and reactor core internals; mechanical, electrical, and instrumentation systems and valve actuation devices; vessel and piping supports; and snubbing devices.

The applicant presents data in his safety analysis report (SAR) in the form of a table which identifies the fluid systems important to safety; the system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves; the associated quality group classification, ASME Code and code class; and the quality assurance requirements. In addition, the applicant presents on suitable piping and instrumentation diagrams the system quality group classifications.

The CSB reviews, in SAR Section 6.2.5, the detailed system design of fluid systems designated AEC Quality Group B, which are provided for the control of combustible gas concentrations in containment following a loss of coolant accident.

The APCSB reviews, in SAR Sections 9 and 10, the detailed system design of auxiliary fluid systems important to safety that are designated AEC Quality Groups B and C.

The ETSB reviews, in SAR Sections 11.2 and 11.3, the detailed design of liquid, gaseous, and solid radioactive waste systems designated AEC Quality Groups C and D.

The RSB will review the detailed system design of engineered safeguards systems that are designated AEC Quality Group B.

The branches that have secondary review responsibility will confirm that the quality group classifications of systems and components within their review scopes are acceptable. If there are systems or components other than those identified by the RSB that are deemed to be important to safety this information should be transmitted to the RSB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records." This criterion requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants important to safety.

3. Regulatory Guide 1.26, "Quality Group Classification and Standards." This Regulatory Guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants. The applicant may use the AEC Group Classification system identified in the Regulatory Guide 1.26 or, alternately, the corresponding ANS classification system of Safety Classes which can be cross-referenced with the classification groups in Regulatory Guide 1.26. Clarification of Regulatory Guide 1.26 provisions with respect to boiling water reactor plant main steam and feedwater systems, and acceptable alternate provisions for these systems are given in branch technical positions attached to this plan.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgement on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

Section 50.55a of 10 CFR Part 50 identifies those ASME Section III, Code Class 1 components of light-water-cooled reactors important to safety which are part of the reactor coolant pressure boundary. These components are designated in Regulatory Guide 1.26 as Quality Group A. In addition, Regulatory Guide 1.26 identifies, on a functional basis, water- and steam-containing components of those systems important to safety which are

Quality Groups B and C. Quality Group D applies to water- and steam-containing components of systems that are less important to safety.

There are also systems of light-water-cooled reactors important to safety that are not identified in Regulatory Guide 1.26 and which the staff considers should be classified Quality Group C. Examples of these systems are: diesel fuel oil system; diesel generator cooling, lubricating oil, and air startup systems; instrument and service air systems required to perform a safety function; and certain ventilation systems. Gas treatment systems which are considered as engineered safeguards systems should be classified Quality Group B.

The information supplied in the application identifying fluid systems important to safety is reviewed for completeness, and the quality group classification, ASME Code and code class, and quality assurance requirements of each individual major component are checked for compliance with the above criteria. The various modes of system operation are checked to assure that the assigned AEC quality groups are acceptable.

The piping and instrumentation diagrams are reviewed to assure that the applicant has delineated in detail the system quality group classification boundaries for systems important to safety. Each individual line on a diagram is checked to assure the accuracy of the assigned quality group classification, including branch lines such as vents, drains, and sample lines. Changes in quality group classification are permitted normally only at valve locations, with the valve assigned the higher classification. A change in quality group classification with no valve present is permitted only when it can be demonstrated that the safety function of the system is not impaired by a failure on the lower-classification side of the boundary.

The following fluid systems important to safety for pressurized water reactor (PWR) and boiling water reactor (BWR) plants are reviewed by the RSB with regard to quality group classification.

FLUID SYSTEMS IMPORTANT TO SAFETY FOR PWR PLANTS

Reactor Coolant System
Emergency Core Cooling System
Containment Spray System
Chemical and Volume Control System
Boron Thermal Regeneration System
Boron Recycle System
Residual Heat Removal System
Component Cooling Water System[2/]
Spent Fuel Pool Cooling and Cleanup System[2/]
Sampling System[3/]
Service Water System[2/]
Compressed Air System[2/]
Diesel Fuel Oil System

Diesel Generator Auxiliary Systems
Main Steam System[3/]
Feedwater System[3/]
Auxiliary Feedwater System
Liquid Waste Processing System[1/]
Gaseous Waste Processing System[1/]
Containment Cooling System
Containment Purge System
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms
Fire Protection System[1/][2/]
Combustible Gas Control System
Condensate Storage System[1/]

FLUID SYSTEMS IMPORTANT TO SAFETY FOR BWR PLANTS

Reactor Recirculation System
Main Steam System (up to but not including the turbine)
Feedwater System[3/]
Relief Valve Discharge Piping
Control Rod Drive Hydraulic System
Standby Liquid Control System
Reactor Water Cleanup System
Liquid Radwaste System[1/]
Gaseous Radwaste System (Off-gas)[1/]
Fuel Pool Cooling and Cleanup System[2/]
Sampling System[3/]
Residual Heat Removal System
High Pressure Core Spray System
Low Pressure Core Spray System
Reactor Core Isolation Cooling System
RHR Service Water System
Emergency Equipment Service Water System
Compressed Air System[2/]
Diesel Generator Auxiliary Systems
Standby Gas Treatment System
Combustible Gas Control System
Containment Cooling System
Main Steam Line Isolation Valve Sealing System
Condensate and Refueling Water Storage System[2/]
Ventilation Systems for Areas such as Control Room and Engineered Safety Features Rooms
Fire Protection System[1/][2/]

[1/]On some plants this system may be non-safety-related, providing it complies with the requirements of Regulatory Guide 1.26.

[2/]Portions of the system that perform a safety-related function.

[3/]Portions of the system to outermost containment isolation valve.

Provisions applicable to BWR main steam and feedwater system quality group and seismic classifications, for those portions of the system on the turbine side of the containment isolation valves, are given in Branch Technical Positions RSB No. 3-1 and 3-2, attached to this plan.

In the event an applicant intends to take exception to Regulatory Guide 1.26 and has not provided adequate justification for his proposed quality group classification, questions are prepared by the staff which may require additional documentation or an analysis to establish an acceptable basis for his proposed quality group classification. Staff comments may also be prepared requesting clarification, in order to assure a clear understanding of the quality group classifications assigned to a system by the applicant.

Exceptions and alternatives to the specified quality group classifications of Regulatory Guide 1.26 are unacceptable unless "equivalent quality level" is justified. In such cases, justification can be demonstrated if: the component is classified to meet the requirements of a higher group classification than specified in Regulatory Guide 1.26 or alternative design rules are based on the use of a more conservative design; the extent of component nondestructive examination is equal to or greater than required by the specified code; and the quality assurance requirements of Appendix B, 10 CFR Part 50 are met.

If the staff's questions are not resolved in a satisfactory manner, a staff position is taken requiring conformance to Regulatory Guide 1.26.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at a conclusion of the following type, which is to be included in the staff's Safety Evaluation report:

"Fluid system pressure-retaining components important to safety will be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Water-and steam-containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (3) to contain radioactive material, have been classified in an acceptable manner in Tables 3.X.X and 3.X.X and on system piping and instrumentation diagrams.

"The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with the Commission's regulations as set forth in General Design Criterion 1, with the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, with Regulatory Guide 1.26, and with staff technical positions and industry standards."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design CRiterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

4. ANSI N18.2a-1975, Revision and Addenda to ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1973).

5. ANS N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," Draft No. 4, Rev. 2, April 1974, ANS Standard Issued for Trial Use and Comment, American Nuclear Society (1974).

6. ANS N213, "Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants," Draft No. 9, Rev. 2, January 1974, ANS Standard Issued for Comment, American Nuclear Society (1974).

7. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1974).

8. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers (1974).

9. ANSI B31.1-1973, "power Piping," American National Standards Institute (1973).

10. API Standard 620, Fifth Edition, "Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks," Americal Petroleum Institute (1973).

11. API Standard 650, Fifth Edition, "Welded Steel Tanks for Oil Storage," American Petroleum Institute (1973).

12. AWWA D100-73, "AWWA Standard for Steel Tanks-Standpipes, Reservoirs, and Elevated Tanks for Water Storage," American Water Works Association (1973).

13. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field-Erected Storage Tanks," American National Standards Institute (1973).

14. Branch Technical Position - RSB No. 3-1, "Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants."

15. Branch Technical Position - RSB No. 3-2, "Classification of BWR/6 Main Steam and Feedwater Components Other Than the Reactor Coolant Pressure Boundary."

CLASSIFICATION OF MAIN STEAM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY FOR BWR PLANTS

A. BACKGROUND

A pipe classification of "D + QA" for main steam line components of BWR plants was proposed by the General Electric Company in 1971 as an alternative to Quality Group B and has been accepted by the staff in a number of licensing case reviews.

However, we have recently identified a number of potential problems which are applicable to main steam lines of BWR plants. These problems relate to postulated breaks in high-energy fluid-containing lines outside the containment. The criteria pertaining to protection required for structures, systems, and components outside containment from the effects of postulated pipe breaks, as contained in the Director of Licensing's letter to utilities dated July 12, 1973, reference ASME Section III, Class 2, which corresponds to AEC Quality Group B.

The recent ASME Code Section XI revision contains in-service inspection requirements for Class 2 components. Steam lines classified as "D + QA" could be intrepreted to be exempt from these inspection requirements. Such interpretations would be contrary to the intent of the code and inconsistent with requirements of the AEC Codes and Standards rule, Section 50.55a of 10 CFR Part 50.

Furthermore, the applicability of the following AEC Regulatory Guides and Regulations, as they relate to ASME Class 2 components is not always clearly identified or implemented in case applications wherever "D + QA" classification is adopted:

1. Regulatory Guide 1.51, "In-service Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components."

2. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

4. 10 CFR § 50.55a, "Codes and Standards for Nuclear Power Plants."

5. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

In view of the foregoing, we find it necessary to clarify the quality group classification criteria for main steam components for BWR plants.

B. BRANCH TECHNICAL POSITION

The main steam line components of BWR plants should conform to the criteria listed in the attached Table 3-1.1.

C. REFERENCES

1. Letter of March 22, 1973, J.A. Hinds to J.M. Hendrie.

2. Letters of August 13, 1973 and November 26, 1973, J.M. Hendrie to J.A. Hinds.

Table 3-1.1

CLASSIFICATION REQUIREMENTS FOR MAIN STEAM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

Item	System or Component	Classification Quality Group	
1.	Main Steam Line from 2nd Isolation Valve to Turbine Stop Valve.	B	
2.	Main Steam Line Branch Lines to First Valve.	B	
3.	Main Turbine Bypass Line to Bypass Valve.	B	
4.	First Valve in Branch Lines Connected to Either Main Steam Lines or Turbine Bypass Lines.	B	
5.	a. Turbine Stop Valves, Turbine Control Valves, and Turbine Bypass Valves.	D + QA or Certification	1/ 2/
	b. Main Steam Leads from Turbine Control Valves to Turbine Casing.	D + QA or Certification	1/3/ 2/

1/ The following requirements shall be met in addition to the Quality Group D requirements:

1. All cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective shall be examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods.

2. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examination in ANSI B31.1-1973, Par. 136.4.

2/ The following qualification shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valves to the turbine casing shall utilize quality control procedures equivalent to those defined in General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine - Generator Quality Control Program."

2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

3/ The following requirements shall be met in addition to the Quality Group D requirements:

1. All longitudinal and circumferential butt weld joint shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may

be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1-1973.

2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrant methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1-1973.

3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

BRANCH TECHNICAL POSITION - RSB NO. 3-2

CLASSIFICATION OF BWR/6 MAIN STEAM AND FEEDWATER COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

A. BACKGROUND

At various times the AEC staff has discussed with the General Electric Company the subject of appropriate classification requirements in boiling water reactor (BWR) plants for main steam system components. These discussions have included consideration of components that are (a) not classified as safety-related items but are located downstream of the isolation valves, (b) not specifically designed to seismic Category I standards, and (c) not housed in seismic Category I structures.

To date, BWR plant reviews have resulted in various approaches for different individual applications. While these different approaches have resulted in acceptable levels of safety in each case, they have required time-consuming case-by-case reviews. The GESSAR BWR/6 application, under review as part of our standardization program, includes this portion of the BWR plant.

In the course of the GESSAR review, we have identified a systematic basis for classification of such components that will result in an acceptable and uniform design basis for the main steam lines (MSL) and main feedwater lines (MFL) in BWR/6 plants.

B. BRANCH TECHNICAL POSITION

The main steam and feedwater system components of BWR/6 plants should be classified in accordance with BTP-RSB No. 3-1, or alternately, in accordance with the attached Table 3-2.1. The classifications indicated are acceptable alternates to the guidelines currently specified in Regulatory Guide 1.26 and Regulatory Guide 1.29.

As an additional requirement, a suitable interface restraint should be provided at the point of departure from the Class I structure where the interface exists between the safety and nonsafety-related portions of the MSL and MFL.

A sketch is attached (Figure 3-2.1) to clarify the specified alternate classification system.

C. REFERENCES

1. Letter of April 19, 1974, J.M. Hendrie to J.A. Hinds.

Table 3-2.1

CLASSIFICATION REQUIREMENTS FOR BWR/6 MAIN STEAM AND FEEDWATER SYSTEM COMPONENTS OTHER THAN THE REACTOR COOLANT PRESSURE BOUNDARY

ITEM	SYSTEM OR COMPONENT	QUALITY GROUP CLASSIFICATION
1.	Main Steam Line (MSL) from second isolation valve to and including shutoff valve.	B
2.	Branch lines of MSL between the second isolation valve and the MSL shutoff valve, from branch point at MSL to and including the first valve in the branch line.	B
3.	Main feedwater line (MFL) from second isolation valve and including shutoff valve.	B
4.	Branch lines of MFL between the second isolation valve and the MFL shutoff valve, from the branch point at MFL to and including the first valve in the branch line.	B
5.	Main steam line piping between the MSL shutoff valve and the turbine main stop valve.	D (1)
6.	Turbine bypass piping..	D
7.	Branch lines of the MSL between the MSL shutoff valve and the turbine main stop valve.	D
8.	Turbine valves, turbine control valves, turbine bypass valves, and main steam leads from the turbine control valves to the turbine casing.	D (1,2) or Certification (3)
9.	Feedwater system components beyond the MFL shutoff valve.	D

(1) All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

(2) All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective shall be examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards may be used as an alternate to radiographic methods. Examination procedures and acceptance standards shall be at least equivalent to those defined in Paragraph 136.4, "Examination Methods of Welds - Non-Boiler External Piping," ANSI B31.1-1973.

(3) The following qualifications shall be met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valves to the turbine casing shall utilize quality control procedures equivalent to those defined in General Electric Publication GEZ-4982A, "General Electric Large Steam Turbine-Generator Quality Control Program."
2. A certification shall be obtained from the manufacturer of these valves and steam leads that the quality control program so defined has been accomplished.

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SECTION 3.2.1 SEISMIC CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

Nuclear power plant structures, systems, and components important to safety should be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions. Information presented by the applicant identifying those structures, systems, and components (including their foundations and supports) which are important to safety and are designed to withstand, without loss of function, the effect of a safe shutdown earthquake (SSE) is reviewed. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those structures, systems, and components that are designed to remain functional if the SSE occurs are designated seismic Category I.

The RSB reviews the classification of those plant features (excluding electrical features) specified as seismic Category I by the applicant in his safety analysis report (SAR). Where required, specific information or assistance may be obtained from the EICSB to review classification of electrical and instrumentation systems. This review is done for both construction permit (CP) and operating license (OL) applications.

The applicant's proposed classifications may be presented in the form of a table which identifies structures and fluid systems that are seismic Category I. Where portions of structures and fluid systems are seismic Category I they also must be clearly identified. For fluid systems important to safety, the classification tables in the application should identify system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, have suitable footnotes defining interfaces, and be in sufficient detail so that there is a clear understanding of the extent of those portions of the system that are classified as seismic Category I.

Alternately, such information may be presented on suitable piping and instrumentation diagrams, or may be combined with the information presented in SAR Section 3.2.2, in which case it may be cross-referenced rather than repeated here.

The CSB reviews, in SAR Section 6.2.5, the detailed system design of seismic Category I fluid systems that are provided for the control of combustible gas concentrations in containment following a loss-of-coolant accident.

The APCSB reviews, in SAR Sections 9 and 10, the detailed system design of auxiliary fluid systems important to safety that are designated seismic Category I.

The ETSB reviews, in SAR Sections 11.2 and 11.3, the detailed system design of seismic Category I liquid, gaseous, and solid radioactive waste systems that are provided to reduce the radioactivity to levels which will not be in excess of the appropriate limits.

In the event a branch that has secondary review responsibility identifies other plant features important to safety that have not been previously identified by the RSB, this information should be transmitted to the RSB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 2. This criterion requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform necessary safety functions.

2. Regulatory Guide 1.29, "Seismic Design Classification." This Regulatory Guide describes an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgement on the areas to be given attention during the review is to be based on an inspection of the material presented, the similarity of the material to that recently reviewed on other plants, and whether items of special safety significance are involved.

Regulatory Guide 1.29, which identifies structures, systems, and components of light-water-cooled reactors on a functional basis, is the principal document used for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements.

The staff review should establish whether the applicant has indicated compliance with Regulatory Guide 1.29 in the SAR. Where there are differences with respect to the Guide, these differences should be identified.

The information supplied by the applicant identifying seismic Category I structures, systems, and components is reviewed for completeness and to assure there is sufficient detail to permit identification of specific equipment. Where portions of a system are classified seismic Category I, the boundary limits of that portion of the system designed to Category I requirements should be identified on the piping and instrumentation diagrams. In addition, where portions of a structure are classified seismic Category I, those portions of the building foundations and supports designed to Category I requirements should be identified on the plant arrangement drawings. The interfaces between components and associated support structures designed to seismic Category I requirements are then checked to assure compatability.

For systems which are partially seismic Category I, the Category I portion of the system should extend to the first seismic restraint beyond the isolation valves which isolate that part which is Category I from the non-seismic portion of the system.

In the event an applicant intends to take exception to Regulatory Guide 1.29 and has not provided an adequate justification for his proposed seismic classification, questions are prepared by the staff which may require additional documentation or analysis to establish an acceptable basis for his proposed seismic classification. Staff comments may also be prepared requesting clarification in order to assure a clear understanding of the seismic classification assigned to a system by the applicant.

If the staff's questions are not resolved in a satisfactory manner, a staff position is taken requiring conformance to Regulatory Guide 1.29.

IV. EVALUATION FINDINGS

The staff's review should verify that adequate and sufficient information is contained in the SAR and amendments to arrive at conclusions of the following type, which are to be included in the staff's safety evaluation report:

"Structures, systems, and components important to safety that are required to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as seismic Category I items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

"All other structures, systems, and components that may be required for operation of the facility are designed to other than seismic Category I requirements. Included in this classification are those portions of Category I systems which are not required to perform a safety function. Structures, systems, and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner in Tables 3.X.X and 3.X.X, and on system piping and instrumentations diagrams.

"The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria and design bases for structures, systems and components important to safety with the Commission's regulations as set forth in General Design Criterion 2, and to Regulatory Guide 1.29, staff technical positions, and industry standards."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

3. Regulatory Guide 1.29, "Seismic Design Classification."

4. ANSI N18.2a-1975, Revision and Addenda to ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1973).

5. ANS N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," Draft No. 4, Rev. 2, April 1974, ANS Standard Issued for Trial Use and Comment, American Nuclear Society (1974).

6. ANS N213, "Nuclear Safety Criteria for the Design of Stationary Gas Cooled Reactor Plants," Draft No. 9, Rev. 2, January 1974, ANS Standard Issued for Comment, American Nuclear Society (1974).

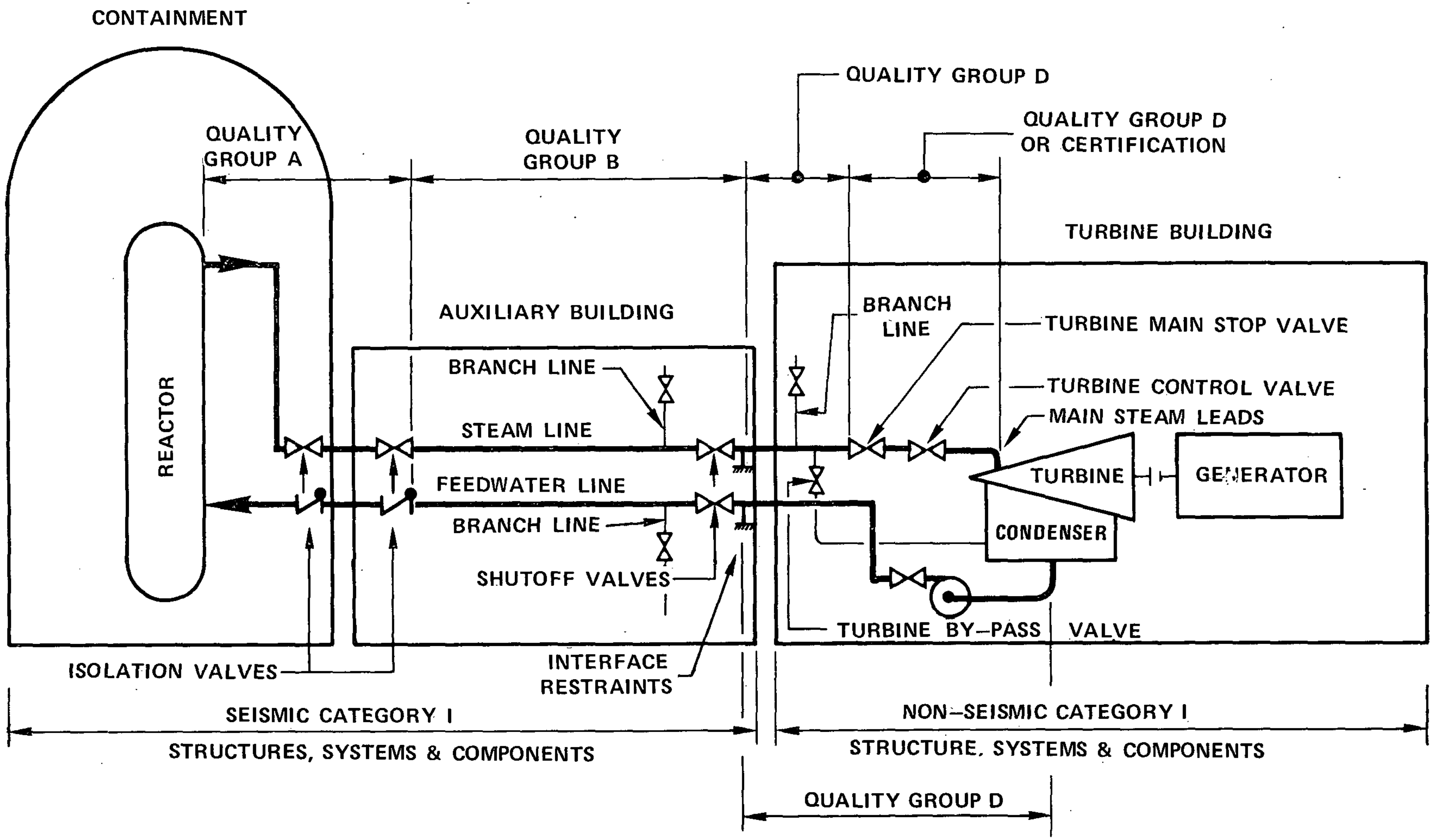

Figure 3-2.1 AEC Quality Group and Seismic Category Classifications Applicable to Power Conversion System Components in BWR/6 Plants.

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SECTION 3.3.1 WIND LOADINGS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The following areas relating to the design of seismic Category I structures to withstand the effects of the design wind specified for the plant are reviewed.

1. The design wind velocity and its recurrence interval, the velocity vertical profiles, and the applicable gust factors are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for wind loadings. The bases for the selection and the values of these parameters are within the review responsibility of the Site Analysis Branch (SAB) as stated in Standard Review Plans 2.3.1 and 2.3.2.

2. The procedures that are utilized to transform the design wind velocity into an effective pressure applied to exposed surfaces of seismic Category I structures are reviewed with particular emphasis on the shape coefficients and distribution of the wind pressure on rectangular flat surfaces and on circular structures such as containments.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The acceptance criteria for the design wind velocity and its recurrence interval, the velocity vertical profiles, the applicable gust factors, and the bases for determining these site-related parameters, are established by the Site Analysis Branch (SAB) and are contained in Standard Review Plans 2.3.1 and 2.3.2. The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.

2. For the procedures utilized to transform the wind velocity into an effective pressure applied to exposed surfaces of structures, the procedures delineated in either the American Society of Civil Engineers (ASCE) Paper No. 3269, "Wind Forces on Structures"

(Ref. 1), or in ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (Ref. 2), are acceptable. In particular, the procedures utilized are acceptable if found in accordance with the following:

a. If the ASCE Paper No. 3269 is selected:
For a design wind velocity of v mph, the dynamic pressure q is given by:

$$q = 0.00256\ v^2\ \text{psf}$$

To arrive at the equivalent uniform pressure acting on a particular structure, the dynamic pressure q should be modified by a shape or drag coefficient which is primarily dependent on the geometry and physical configuration of the structure. Shape or drag coefficients for a variety of structures are given in Table 4 of ASCE Paper No. 3269 (Ref. 1). Geometrical shapes that are not specifically covered in the ASCE Paper No. 3269 case reviewers on a case-by-case basis.

b. If the ANSI A58.1-1972 document is selected:
For a design wind velocity of V_{30} mph specified at a height of 30 ft above the ground, the velocity pressure, q_{30}, is given by:

$$q_{30} = 0.00256\ V_{30}^2\ \text{psf}$$

The effective pressure for structures, q_F, and for portions thereof, q_P, at various heights above the ground should be in accordance with Table 5 and Table 6 of ANSI A58.1-1972, respectively. Since most nuclear power plants are located in relatively open country, Exposure C, as defined in ANSI A58.1-1972, should be selected for both tables.

Depending on the structure geometry and physical configuration, pressure coefficients may be selected in accordance with Section 6.4 of ANSI A58.1-1972. Geometrical shapes that are not covered in this document are reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. The site-related parameters described in Section I. 1. of this plan are reviewed by the Site Analysis Branch (SAB) and are covered by Standard Review Plans 2.3.1 and 2.3.2. The structural reviewer examines the approved values of these parameters to assure himself that the procedures utilized in designing the structures to withstand the specified wind loadings are appropriate and applicable.

2. After the applicability of the site-related parameters is established, the reviewer proceeds with his review of the structural aspects of wind design. The procedures utilized by the applicant to transform wind velocities into applied pressures are reviewed and compared with those procedures delineated in either ASCE Paper No. 3269 or in ANSI A58.1-1972 document, whichever has been selected. In particular, the pressures and shape coefficients utilized for rectangular buildings and circular structures are reviewed and compared with those referenced in Section II.2 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's Safety Evaluation report:

"The procedures utilized to determine the loadings on seismic Category I structures induced by the design wind specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

2. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," Committee A58.1, American National Standards Institute (1972).

3. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

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SECTION 3.3.2 TORNADO LOADINGS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)
Accident Analysis Branch (AAB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The following areas relating to the design of structures that have to withstand the effects of the design basis tornado specified for the plant are reviewed.

1. The design parameters applicable to the tornado, including the tornado wind translational and tangential velocities, the tornado-generated pressure differential and its associated time interval, and the spectrum of tornado-generated missiles including their characteristics, are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for tornado loadings. The bases for the selection and the values of these parameters are within the review responsibility of the Site Analysis Branch (SAB) and the Accident Analysis Branch (AAB), as stated in Standard Review Plans 2.3.1, 2.3.2,and 3.5.1.4.

2. The procedures that are utilized to transform the tornado parameters into effective loads on structures are reviewed, including the following.
   a. The transformation of the tornado wind into an effective pressure applied to exposed surfaces of structures, with particular emphasis on shape coefficients and the pressure distribution on flat surfaces and circular structures such as containments.
   b. If venting of a structure is utilized, the procedures for transforming the tornado-generated differential pressure into an effective reduced pressure are reviewed, upon request, by the Auxiliary and Power Conversion Systems Branch (APCSB).
   c. The transformation of tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads.
   d. The combination of the above individual loadings in a manner that will produce the most adverse total tornado effect on structures.

3. The information provided to demonstrate that failure of any structure or component not to be designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The acceptance criteria for the tornado wind velocity, the differential pressure and its associated time interval, the spectrum of tornado-generated missiles and their characteristics, and the bases for determining these parameters, are established by the Site Analysis Branch (SAB) and the Accident Analysis Branch (AAB) and are contained in Standard Review Plans 2.3.1, 2.3.2 and 3.5.1.4. The approved values of these parameters should serve as basic input to the review and evaluation of the structural design procedures.

2. The acceptance criteria for the procedures utilized to transform the tornado parameters into effective loadings on structures are as follows:
   a. For transforming the tornado wind velocity into an effective pressure applied to exposed surfaces of structures, the criteria delineated in either the American Society of Civil Engineers (ASCE) Paper No. 3269, "Wind Forces on Structures" (Ref. 1), or in ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (Ref. 2), are, in general, acceptable. In particular, the following shall apply:
      (i) The maximum velocity pressure, p, should be based on the maximum tornado velocity, V, using the following formula:
      $$p = 0.00256\ V^2 \text{ psf, in which V is in mph.}$$
      (ii) The velocity pressure should be assumed constant with height.
      (iii) The maximum velocity pressure, p, applies at the radius of the tornado funnel at which the maximum velocity occurs. It can therefore be modified or reduced at points away from this radius. The manner of such a reduction will be reviewed on a case-by-case basis.
      (iv) For calculating velocity pressures on external surfaces of structures, on external portions thereof, and on internal surfaces, where there are openings in the structure, appropriate shape coefficients shall be used in accordance with ASCE Paper No. 3264 (Ref. 1). Gust factors may be taken as unity.
   b. If venting of a structure is adopted as a design measure to permit transforming the tornado-generated differential pressure into an effective reduced pressure, the acceptance criteria are established on a case-by-case basis, upon request, by the Auxiliary and Power Conversion Systems Branch (APCSB).
   c. The acceptance criteria for transforming the tornado-generated missile impact into an effective or equivalent static load on structures are delineated in Section II of Standard Review Plan 3.5.3.
   d. Having established the effective loads for each of the above three individual tornado-generated effects, the combination thereof should then be determined in a conservative manner for each particular structure, as applicable. An acceptable method of combining these effects is as follows:

(i) $W_t = W_w$
(ii) $W_t = W_p$
(iii) $W_t = W_m$
(iv) $W_t = W_w + .5\ W_p$
(v) $W_t = W_w + W_m$
(vi) $W_t = W_w + .5\ W_p + W_m$

where: W_t ..... total tornado load,
W_w ..... tornado wind load,
W_p ..... tornado differential pressure load, and
W_m ..... tornado missile load.

For each particular structure or portion thereof, the most adverse of the above combinations should be used, as appropriate.

These combined effects constitute the total tornado load which should then be combined with other loads as specified in Standard Review Plans 3.8.1, 3.8.4, and 3.8.5.

3. The information provided to demonstrate that failure of any structure or component not to be designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions, is acceptable if found in accordance with either of the following:
   a. The postulated collapse or structural failure of structures and components not to be designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures or components.
   b. Safety-related structures are designed to resist the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. The site-related parameters described in Section I.1 of this plan are reviewed by the Site Analysis Branch (SAB) and the Accident Analysis Branch (AAB) in accordance with Standard Review Plans 2.3.1, 2.3.2, and 3.5.1.4. The structural reviewer examines the approved values of these parameters to assure himself that the design basis and procedures utilized in designing structures to withstand tornado loads are appropriate and applicable.

2. After the applicability of the site-related parameters is established, the reviewer proceeds with his review of the structural aspects of tornado design in the following manner:
   a. The procedures utilized by the applicant to transform tornado wind velocities into effective pressures are reviewed and compared with those procedures delineated in

either ASCE paper No. 3269 or in ANSI A58.1-1972, whichever is selected, and, in particular, with the acceptance criteria delineated in Section II.2.a of this review plan.

b. Where venting is utilized, procedures for transforming the tornado-generated differential pressure into an effective reduced pressure are reviewed, upon request, by the Auxiliary and Power Conversion Systems Branch (APCSB).

c. The treatment of tornado-generated missiles is covered in Standard Review Plan 3.5.1.4 and the review procedures for design of missile barriers are described in Standard Review Plan 3.5.3.

d. After procedures for determining the individual tornado effects are reviewed, the manner in which these effects are then combined to arrive at the most adverse total tornado effect is reviewed and compared with the acceptance criteria delineated in Section II.2.d of this plan. Other proposed methods which may depend on the geometry and configuration of a particular structure are reviewed on a case-by-case basis.

3. The information provided to demonstrate that failure of any structure or component not to be designed for tornado loads will not affect the capability of other structures or components to perform necessary safety functions is reviewed to assure that one of the acceptance criteria of Section II.3 of this plan is satisfied.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of a design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

2. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," Committee A58.1, American National Standards Institute (1972).

3. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

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SECTION 3.4.1 FLOOD PROTECTION

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Site Analysis Branch (SAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The APCSB review of the plant flood protection includes all systems and components whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. The facility design and equipment arrangements presented in the applicant's safety analysis report (SAR) are reviewed with respect to the following considerations: to identify the safety-related systems and components that must be protected against flooding; to determine the capabilities of structures housing safety-related systems or equipment to withstand a flood, i.e., the relationship between structure elevation and flood elevation as determined by the Section 2.4 Standard Review Plans (SRP); to determine adequacy of the isolation of redundant safety-related systems or equipment subject to flooding; to identify possible inleakage sources, such as cracks in structures not designed to withstand seismic events and exterior or access openings or penetrations in structures located at a lower elevation than the flood level. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this review plan.

The review of flood protection involves secondary evaluations performed by other branches. The conclusions of their evaluations will be used by the APCSB to complete the overall evaluation of the subject area. The Site Analysis Branch verifies the elevations determined for the various conditions of site flooding, including the probable maximum flood and the adequacy of the type of flood protection utilized (SRP for Section 2.4). The Structural Engineering Branch determines the acceptability of the design analyses, procedures, and criteria used for seismic Category I structures that must withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the design basis flood, and tornado missiles. The Electrical, Instrumentation and Control Systems Branch will, upon request, verify the adequacy of instrumentation needed for flood protection, including adequacy of detectors and alarms necessary to detect rising water levels within structures, and will evaluate the consequences of flooding on other safety-related instrumentation and electrical equipment in affected areas (SRP 7.6).

II. ACCEPTANCE CRITERIA

Acceptability of the flood protection measures described in the SAR, including related portions of Chapter 3 of the SAR, is based on specific general design criteria and regulatory guides and on the reviewer's independent evaluation and calculations with respect to area or component flooding. Listed below are specific criteria as they relate to flooding.

The facility design and equipment locations are acceptable if they are in accordance with General Design Criterion 2, "Design Bases for Protection against Natural Phenomena," as related to systems and components withstanding flood conditions, and Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants." An additional basis for determining the acceptability of the facility will be the degree of similarity to previously approved plants with respect to means of providing flood protection.

III. REVIEW PROCEDURE

The review procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan. For the review of operating license (OL) applications the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer will select and emphasize material from the paragraphs below as may be appropriate for a particular case.

The general review procedures for OL's include a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements developed as a result of the staff's review. Where necessary, the review will include requirements for system testing, minimum performance, and surveillance.

The review procedure consists of:

1. A determination from the SAR as to which systems and components are safety-related and should be protected against floods or flooded conditions.

2. An evaluation using the plant arrangement and layout drawings as to the various means to prevent flooding of safety-related systems or components, such as pumping systems, stoplogs, and watertight doors. The measures utilized are reviewed as to their ability to cope with the design basis flood, as established in the SRP for Section 2.4 of the SAR.

3. An assessment of leakage, a determination if liquid-carrying systems could produce flooding, and an evaluation of the measures taken to protect safety-related

equipment. A failure modes and effects analysis may be performed to determine that the flooding consequences resulting from failures of such liquid-carrying systems close to essential equipment will not preclude required functions of safety systems.

4. A review of the SAR to ascertain if safety-related systems or components are capable of normal function while completely or partially flooded.

5. A review of plant arrangement and layout drawings to determine if any safety-related equipment or components are located within individual compartments or cubicles which act as positive barriers against possible means of flooding.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The flood protection review included all systems and components whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. The scope of the flood protection review for the ______________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for all systems and components that are essential to the safe operation and shutdown of the plant. [The review has included the applicant's proposed design criteria and design bases for safety-related systems, structures and components, the adequacy of those criteria and bases, and the requirements to maintain the capability for a safe plant shutdown during the design basis flood (CP).] [The review has included the applicant's analysis of the manner in which the design of structures, systems and components conforms to the applicable design criteria and bases, and demonstrates the ability to perform a safe plant shutdown during the design basis flood (OL).]

"The staff concludes that the design of the facility for flood protection conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

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SECTION 3.4.2 ANALYSIS PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The following areas relating to the design of seismic Category I structures to withstand the effects of the flood or highest ground water specified for the plant are reviewed.

1. The design parameters of the flood or highest groundwater are reviewed from the standpoint of use in defining the input parameters for the structural design criteria appropriate to account for flood and groundwater loadings. Further, for plants where the flood level is higher than the proposed grade around the plant structures, the dynamic phenomena associated with such a flooding such as currents, wind waves, and their hydrodynamic effects, are similarly reviewed. The bases for these parameters are within the review responsibility of the Site Analysis Branch (SAB) as stated in Standard Review Plan 2.4.2.

2. The procedures that are utilized to transform the static and dynamic effects of the flood and highest ground water into effective loads applied to seismic Category I structures are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The acceptance criteria for the flood or highest groundwater level, for establishing the dynamic effects of the flood where it is above the plant grade, and for the bases for determining these site-related and hydrodynamic parameters, are established by the Site Analysis Branch (SAB) as stated in Standard Review Plan 2.4.2.

2. In most situations, the flood level is below the proposed plant grade and only its hydrostatic effects need be considered. Unless the hydrostatic head associated with the flood or with the highest groundwater level is relieved by utilizing a drainage

and pumping system around the foundations of structures, it has to be considered as a structural load on the basement walls and foundation slab of the building. Another consideration in such a situation is to prevent any uplift or floating of the structure. The total bouyancy force may be based on the flood or highest groundwater head excluding wave action, if applicable. However, the lateral, overturning, and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, which should be considered in the structural design of these elements, should be based on the total head including wave action, if any.

Where the flood level is above the proposed plant grade, the dynamic loads of wave action should be considered. Procedures for determining such dynamic loads are acceptable if they are in accordance with or similar to those delineated in the U.S. Army Coastal Engineering Research Center, Technical Report No. 4 (Ref. 1), as applicable. Other methods are reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. The site-related and hydrodynamic parameters described in Section II.1 of this plan are reviewed by the Site Analysis Branch (SAB) and are covered in Standard Review Plan 2.4.2. The structural reviewer examines the approved values of these parameters to assure that the procedures utilized in designing the structures to withstand the specified flood loadings are appropriate and applicable.

2. After the applicability of the site-related and hydrodynamic parameters is established, the reviewer proceeds with his review of the structural aspects of the design for flood or groundwater. The procedures utilized by the applicant to determine effective flood loads are reviewed and compared with those procedures delineated in Section II.2 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures will withstand such environmental forces.

"The use of these procedures provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required. Conformance with these design procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2."

V. REFERENCES

1. U.S. Army Coastal Engineering Research Center, Technical Report No. 4, "Shore Protection, Planning and Design," 3rd Edition, 1966.

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

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SECTION 3.5.1.1 INTERNAL GENERATED MISSILES (OUTSIDE CONTAINMENT)

REVIEW RESPONSIBILITIES

Primary - Auxiliary And Power Conversion Systems Branch (APCSB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The APCSB review of the structures, systems and components (SSC) to be protected from internally generated missiles (outside containment) includes all other balance of plant SSC SSC on the site that have been provided to support the reactor facility. The review includes missile sources and internally generated missiles associated with component overspeed failures and missiles that could originate from high-pressure system ruptures.

The APCSB reviews the functional operations and performance requirements for all structures, systems, and components outside containment and identifies the SSC that are necessary for the safe shutdown of the reactor facility in the event of a postulated accident or other circumstances that might result in internally generated missiles. Safety-related SSC will be reviewed with respect to their capability to perform functions required for attaining and maintaining a safe shutdown condition during such accident conditions.

The review of internally generated missile protection includes the following: structures, systems or portion of systems, and components that require protection from internally generated missiles are identified; pressurized components and systems are reviewed to determine their potential for generating missiles; such as valve bonnets and hardware retaining bolts, relief valve parts, and instrument wells; high speed rotating machinery are reviewed to determine their potential for generating missiles from component overspeed or failure, such as failure of the pump itself (resulting from seizure), pump or component parts, and rotating segments (e.g., impellers and fan blades).

The Structural Engineering Branch determine the acceptability of the analysis and criteria used for the design of structures or barriers that protect essential systems and components from internally generated missiles (Standard Review Plan 3.5.3). Their results are used by the APCSB to complete the overall evaluation of protection against internally generated missiles.

II. ACCEPTANCE CRITERIA

Acceptability of the design information on protection of essential systems and components from internally generated missiles presented in the applicant's safety analysis report (SAR) is based on meeting specific general design criteria and regulatory guides. An additional basis for determining acceptability is the degree of similarity of the design to previously approved plants.

The design of structures, systems, and components is acceptable if the integrated design affords missile protection in accordance with the following criteria: General Design Criterion 4, "Environmental and Missile Design Basis" as it relates to structures housing essential systems and to the systems being capable of withstanding the effects of internally generated missiles; Regulatory Guide No. 1.13, "Fuel Storage Facility Design Basis," as it relates to the design of essential spent fuel pool systems to withstand the effects of internally generated missiles and to provisions to prevent missiles from contracting spent fuel assemblies; and Regulatory Guide No. 127, "Ultimate Heat Sink," as it relates to the design of heat sinks and connecting piping to withstand the effects of internally generated missiles.

A statement in the SAR that essential structures, systems, and components will be protected by locating the systems or components in individual missile-proof structures, physically separating redundant systems or components of the system, or providing special localized protective shields or barriers, is acceptable for the construction permit stage for providing protection from internally generated missiles (outside containment).

III. REVIEW PROCEDURES

The review procedures set forth below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design in applicant's preliminary safety analysis report meet the acceptance criteria given in Section II of this review plan. For the review of the operating license (OL) application, the review procedures and acceptance criteria are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer selects and emphasizes areas within the scope of this plan as may be appropriate in a particular case.

The first objective in the review of the reactor facilities, structures, systems and components, with regard to protection requirements for internally generated missiles, is to determine whether they are needed to perform a safety function. Some structures and systems are designed as safety-related in their entirety, others have portions that are safety-related, and others are classified as not needed for safety. In order to determine their safety category, the APCSB evaluates the SSC with regard to their function in achieving safe reactor shutdown conditions or in preventing accidents or mitigating the consequences of such accidents. The single failure criterion is used in the analysis. The safety functions to be performed by the SSC in the various plant designs are essentially the same. However, the location of the SSC and the methods used vary from plant to plant depending upon the individual design. This review identifies variations in the various designs that must be evaluated or an individual case basis. Structures, systems, or components that perform a safety function, or by virtue of their failure could have an adverse effect on a safety function shall be protected from the effects of internally generated missiles.

The information provided in the SAR pertaining to SSC design bases and criteria, system descriptions and safety evaluations, piping and instrumentation diagrams, station layout drawings, and system and component characteristic and classification tables are reviewed to identify potential sources of missiles and to determine that protective measures are provided to maintain their safety-related functions. The reviewer may use failure mode and effect analyses and the results of reviews by other branches in evaluating specific SSC and the orgin of possible missiles, in identifying the structures, systems, and components that require protection from internally generated missiles and the adequacy of the protection provided.

Additional guidance can be found in the branch technical positions attached to Standard Review Plans 3.6.1 and 3.6.2 with regard to high and moderate energy breaks in piping systems outside containment.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this plan and that his evaluation is complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The review of possible effects of internally generated missiles (outside containment) included structures, systems, and components whose failure could prevent safe shutdown of the plant or result in signifcant uncontrolled release of radioactivity. The scope of review in this area for the ABC nuclear power plant included layout drawings, piping and instrumentation diagrams, and descriptive information for systems and components essential to the safe operation and shutdown of the plant. [The review has included the applicant's proposed design criteria and bases for essential structures, systems, and components, the adequacy of those criteria and bases, and the equipment necessary to maintain the capability for a safe plant shutdown in the event of an internally generated missile (outside containment). (CP)]. [The review has included the applicant's analysis of the manner in which the design of essential structures, systems, and components conforms to the previously approved design criteria and bases and demonstrates the ability to perform a safe plant shutdown after any internally generated missile accident (outside containment). (OL)].

"The staff concludes that the facility design with regard to protection from internally generated missiles (outside containment) conforms to the Commission's Regulations and to applicable Regulatory Guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside of Containment," attached to SRP 3.6.1.

3. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP 3.6.2.

4. Branch Technical Position SEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP 3.6.2.

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SECTION 3.5.1.2 INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Structural Engineering Branch (SEB)
Containment Systems Branch (CSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The RSB review of the structures, systems, and components (SSC) to be protected from internally generated missiles (inside containment) includes all SSC within the containment and the containment itself. The review includes internally generated missiles associated with component overspeed failures and missiles that could originate from high energy fluid system failures.

The RSB with the assistance of the CSB reviews the functional operations and performance requirements for structures, systems, and components inside containment and identifies which of the operations are necessary for the safe shutdown of the reactor facility in the event of an accident or other circumstances that might result in an internally generated missile, or for the mitigation of the effects of loss-of-coolant or other accidents. Safety-related SSC are reviewed with respect to their capability to perform functions required for attaining and maintaining a safe shutdown condition during such accident conditions.

The review of internally generated missile protection includes the following:

1. Structures, systems or portion of systems, and components are identified as requiring protection from internally generated missiles.

2. Pressurized components and systems are reviewed to determine the potential for generating missiles such as valve bonnets and hardware, retaining bolts, relief valves parts, and instrument wells.

3. High speed rotating machinery is reviewed to determine the potential for generating missiles from component overspeed or failure, such as failure of the pump itself (resulting from seizure), pump or component parts, and rotating segments (e.g., flywheels, impellers and fan blades).

The SEB determines the acceptability of the analytical procedures and criteria used for structures or barriers that protect the containment structure and liner, essential systems, and safety-related components from internally generated missiles (Standard Review Plan 3.5.3). Their results are used by the RSB and CSB to complete the overall evaluation of protection against internally generated missiles. The RSB identifies those systems which are designed to withstand the effects of postulated high energy piping failures in accordance with the criteria stated in Regulatory Guide 1.46 (Ref. 4). These systems provide substantial protection from potential missiles and are reviewed by MEB for missile consequences only in those situations for which the protection provided for piping failures is not considered completely adequate by RSB or CSB.

II. ACCEPTANCE CRITERIA

Acceptability of the design information on protection of structures and essential systems and components from internally generated missiles, as presented in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for determining acceptability is the degree of similarity of the design to that of previously approved plants.

The design of structures, systems, and components is acceptable if the integrated design affords missile protection in accordance with the following criteria:

1. General Design Criterion 4, as it relates to structures housing essential systems and to the systems being capable of withstanding the effects of internally generated missiles.

2. ASME Code Section III, as it relates to the design of steel or concrete containment, whichever is appropriate.

A statement in the SAR that essential structures, systems, and components will be afforded protection by locating the systems or components in individual missile-proof structures, physically separating redundant systems or components of the system, or providing special localized protective shields or barriers, is an acceptable design basis at the construction permit stage for providing protection from internally generated missiles (inside containment).

III. REVIEW PROCEDURES

The review procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this review plan. For the review of operating license (OL) applications, the review procedures and acceptance criteria are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The reviewer selects and emphasizes areas within the scope of this plan as may be appropriate in a particular case.

The first objective in the review of the reactor facilities, structures, systems, and components, with regard to protection requirements for internally generated missiles, is to determine whether the equipment is needed to perform a safety function. Some structures and systems are designed as safety-related in their entirety, others have portions that are safety-related, and others are classified as not needed for safety. In order to determine the safety category of the SSC, the RSB and CSB evaluate the SSC with regard to their function in achieving safe reactor shutdown conditions or in preventing accidents or mitigating the consequences of accidents. The location of the SSC and the protection provided varies from plant to plant depending upon the individual design. The reviewer identifies variations in the design that must be evaluated on an individual case basis. Structures, systems, or components that perform a safety function, or by virtue of their failure could have an adverse effect on a safety function should be protected from the effects of internally generated missiles.

The information provided in the SAR pertaining to SSC design bases and criteria, system descriptions and safety evaluations, piping and instrumentation diagrams, station layout drawings, and system and component characteristic and classification tables is reviewed to identify potential sources of missiles and to determine any protective measures afforded the system or component if safety functions can be affected. The reviewer may use failure mode and effect analyses and the results of other parts of the facility review in evaluating specific SSC and the origin of possible missiles, and in determining which structures, systems, and components require protection from internally generated missiles and whether the degree of protection provided is adequate.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The review of possible effects of internally generated missiles (inside containment) included structures, systems, and components whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity. The scope of review in this area for the __________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for structures, systems, and components essential to the safe operation and shutdown of the plant. [The review has included the applicant's proposed design criteria and bases for essential structures, systems, and components, the adequacy of those criteria and bases, and the equipment necessary to maintain the capability for a safe plant shutdown in the event of an internally generated missile (inside containment)(CP).] [The review has included the applicant's analysis of the manner in which the design of essential structures, systems, and components conforms to the previously approved design criteria and bases and demonstrates the ability to perform a safe plant shutdown after any internally generated missile accident (inside containment)(OL).]
>
> "The staff concludes that the facility design with regard to protection from internally generated missiles (inside containment) conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

3. Regulatory Guide 1.27, "Ultimate Heat Sink."

4. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," and Division 2 (ACI-359), "Standard Code for Concrete Reactor Vessels and Containments," American Society of Mechanical Engineers.

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SECTION 3.5.1.3 TURBINE MISSILES

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCS)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The turbine missile analysis is reviewed with the objective of establishing whether safety-related plant structures, systems, and components have adequate protection against potential turbine missiles. The primary areas of review are the high trajectory turbine missile strike probabilities and the turbine-generator orientation and placement relative to the safety-related plant structures, systems, and components. Additional review areas include the following:

1. Turbine missile barrier design procedure adequacy (SEB).

2. Turbine disk failure analysis (MTEB).

3. Turbine disk fracture toughness properties and startup procedures which assure adequately high disk temperatures (MTEB).

4. Turbine overspeed protection system reliability (EICSB and APCSB).

5. Target redundancy and independence (APCSB).

6. Inservice inspection (MTEB and APCSB).

II. ACCEPTANCE CRITERIA

Plant design and layout must satisfy General Design Criterion 4 (Ref. 1), which states that structures, systems, and components important to safety should be protected against the effects of missiles that might result from equipment failures. Specifically, in the areas reviewed by the AAB, acceptability will be based on the following considerations:

1. Plant design and layout in relation to plant vital systems or structures exposed to potential low trajectory turbine missiles that may be ejected in the event of a destructive overspeed failure of any turbine-generator unit in the vicinity of the plant.

2. Protection against high trajectory turbine missiles including: the total plant area associated with a reactor unit's vital systems which are vulnerable to high trajectory turbine missiles, the overall high trajectory turbine missile strike and damage probability of leading to consequences greater than the 10 CFR Part 100 guidelines, the units within reach of potential high trajectory turbine missiles from more than one turbine-generator, redundant overspeed protection systems, and the exclusion of vulnerable vital systems from high trajectory turbine missile target areas on the basis of redundancy if the systems are sufficiently separated and isolated from each other so that a single missile could not damage both systems.

3. The turbine overspeed protection system should be designed to limit turbine speed to less than 130% of normal speed. There should be sufficient redundancy so that any single failure in the overspeed sensing and trip actuation portions of the system, as well as in the turbine steam valves, would not prevent the overspeed protection system from operating.

4. The overspeed protection system should be tested frequently to confirm that all overspeed detection and turbine trip actuation functions are operable. All turbine steam valves (i.e., stop valves, dump valves, etc.) which are used to reduce, divert, or otherwise limit the steam flow that is available for driving the turbine into an overspeed condition should be tested frequently. Where turbine design does not permit frequent stop valve testing an equivalent means of assuring comparable valve reliability should be provided.

5. Low pressure turbine disk materials, manufacturing processes and operating conditions should conform to the recommendations of Reference 3.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved. The review procedure involves the following:

1. A review of turbine orientation and placement with respect to low trajectory turbine missiles.

2. A review of the plant vital systems with respect to high trajectory turbine missiles in terms of target plan areas, horizontal barriers, target turbine orientations, and distances. If necessary, a structural damage assessment will be made on the basis of information provided by the MTEB regarding turbine missile characteristics and from the

SEB regarding barrier penetration and spalling damage methodology using such techniques as described in Appendix A (Ref. 4).

The reviewer should be aware of the following parallel work which may affect the turbine missile evaluation:

1. The adequacy of structural turbine missile barrier design procedures are verified by the SEB.

2. The fracture toughness properties of the low pressure turbine wheels are reviewed by the MTEB.

3. The turbine overspeed protection system and its testing (including the turbine steam valves) are evaluated by the EICSB and the APCSB.

4. The identification of plant essential systems to be protected against turbine missiles is reviewed by the APCSB.

5. The description and analysis associated with the physical and kinematic properties of postulated turbine missiles are evaluated by the MTEB.

References 6 through 8 provide general background on the turbine missile problem.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, one (or a combination) of which should be included in the staff's safety evaluation report:

1. The overall probability that turbine missiles could damage the plant and lead to consequences in excess of the 10 CFR Part 100 exposure guidelines is acceptably low, so that the plant essential systems are protected adequately against potential turbine missile damage.

2. The overall high trajectory turbine missile strike and damage probability for the plant is too high, and leads to potential consequences greater then the 10 CFR Part 100 guidelines. Additional protection against design overspeed high trajectory turbine missiles is required to reduce the essential system target area so that the overall turbine missile damage probability is acceptable.

3. The indicated turbine orientation and placement exposes the (<u>plant systems</u>) to potential low trajectory or direct strike turbine missiles. Reorientation of the turbine unit(s) or repositioning of the (<u>plant systems</u>) are required to reduce the probability of destructive overspeed turbine missile damage to an acceptable level.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Standard Review Plan 10.2.3, "Turbine Disk Integrity."

4. Appendix A, "High Trajectory Turbine Missile Analyses," appended.

5. S. H. Bush, "Probability of Damage to Nuclear Components," Nuclear Safety, Vol. 14, No. 3, May-June 1973.

6. ANSI N177, "Plant Design Against Missiles," draft standard of ANS 20.1 Working Group (1973).

7. H. G. Mangelsdorf, letter from the Advisory Committee on Reactor Safeguards on turbine missiles, April 18, 1973.

APPENDIX A

STANDARD REVIEW PLAN 3.5.1.3

HIGH TRAJECTORY TURBINE MISSILE ANALYSES

I. STRIKE PROBABILITY ANALYSIS FOR HIGH TRAJECTORY TURBINE MISSILES

If various turbine internals (such as stator blade rings) did not offer any resistance to turbine missiles, the missile trajectories would tend to stay within the plane of the original wheel. In practice, failed wheel fragments can interact with various parts of the turbine, and thus can be deflected away from the plane of the wheel. The limit of angular deviation, Δ, from the wheel plane usually is less for inner wheels than for the end wheels. In this analysis, it is assumed that all turbine missiles are limited to inner wheel deflections. This is a conservative assumption when analyzing high trajectory strike probabilities because a greater departure from the wheel plane would spread the missiles over a larger target area, thus lowering the strike probability density. It should be noted that there are significantly more inner wheels than end wheels.

Denoting the solid angle described by the deflection angles Δ as Ω^*, we can formulate the directional probability density as follows. Assuming a uniform distribution of initial missile directions within the solid angle Ω^*, the directional probability density per unit solid angle, ρ_Ω, can be written as

$$\rho_\Omega \, d\Omega \equiv \frac{d\Omega}{\Omega^*}. \tag{1}$$

The incremental solid angle $d\Omega$ can be expressed in terms of the missile elevation angle ϕ as (see Figure 1)

$$d\Omega = \frac{(R d\phi)(R \cos\phi \, d\theta)}{R^2} = \cos\phi \, d\phi \, d\theta \tag{2}$$

where R is an arbitrary radius of a sphere. The total solid angle is given by

$$\Omega^* = \frac{\frac{1}{2}\displaystyle\int_{\phi=\frac{\pi}{2}-\Delta}^{\phi=\frac{\pi}{2}+\Delta} (R d\phi)(2\pi R \sin\phi)}{R^2} = 2\pi \sin\Delta \tag{3}$$

where the 1/2 in front of the integral denotes that the locus of all eligible missiles is confined to a surface above the horizontal plane.

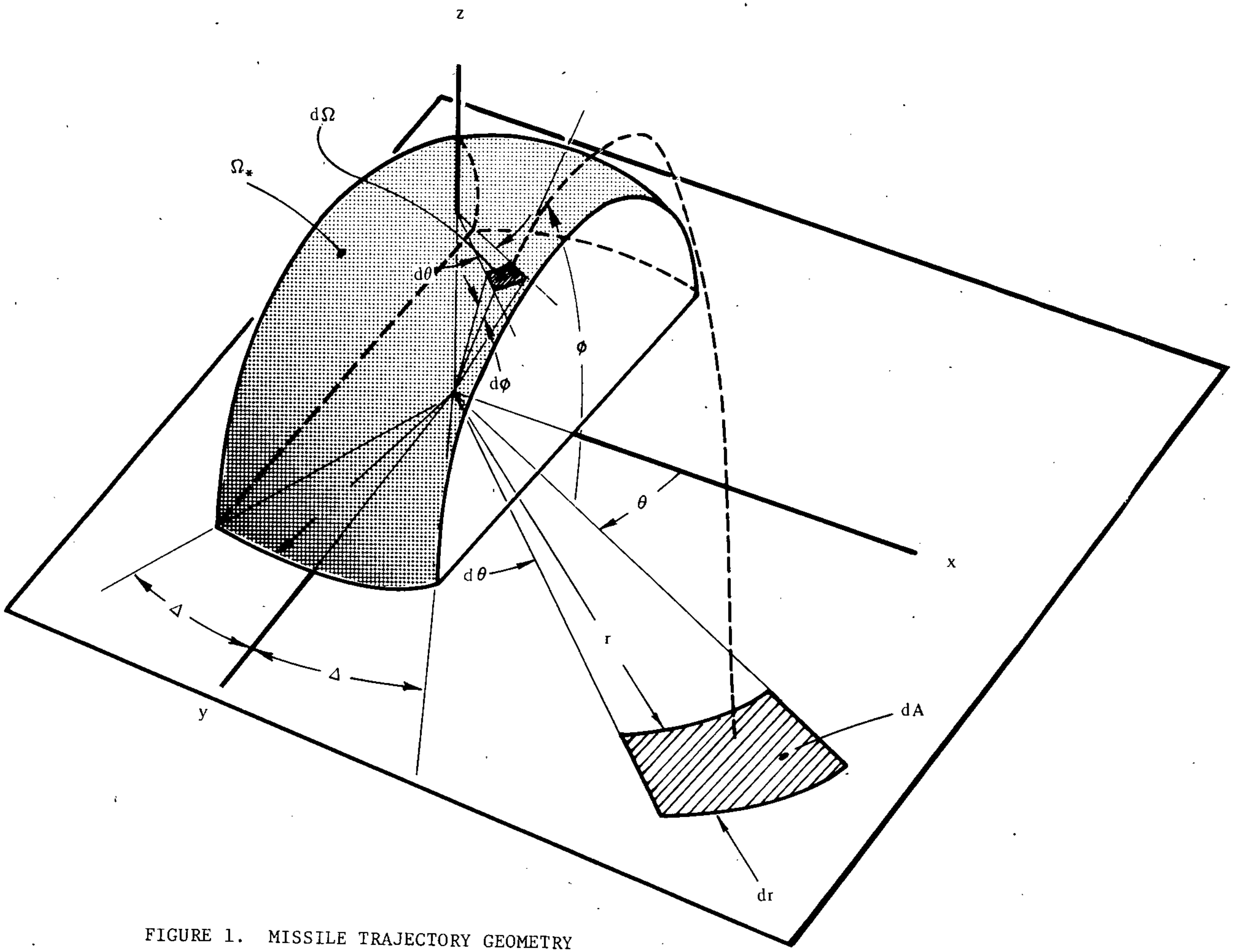

FIGURE 1. MISSILE TRAJECTORY GEOMETRY

In order to define a directional probability density per unit elevation angle, we may note the following. The probability of finding a missile direction within the incremental solid angle $d\Omega$ should be the same as the probability of finding a missile direction within the angular increments $d\theta$ and $d\phi$ which bound $d\Omega$. That is,

$$\rho_\Omega \, d\Omega = \rho_\phi(\theta,\phi) \, d\phi \tag{4}$$

Applying Equations (2) and (3) to (4), we obtain

$$\rho_\phi(\theta,\phi) \, d\phi = \frac{\cos\phi}{2\pi \sin\Delta} \, d\theta d\phi \tag{5}$$

Assuming a uniform distribution of initial missile speeds in the range V_1 to V_2, the speed probability density per unit speed, $\rho_V(V)$, is defined by

$$\rho_V(V) \equiv \frac{dV}{V_2 - V_1} \tag{6}$$

where

$$V_1 \leqslant V \leqslant V_2$$

The compound probability that a missile will have an initial speed within V and V + dV, and an initial direction within ϕ and $\phi+d\phi$, , θ and $\theta+d\theta$, is given by

$$\rho_V(V) \, \rho_\phi(\theta,\phi) \, dV d\phi = \frac{d\theta}{V_2 - V_1} \; \frac{\cos\phi}{2\pi \sin\Delta} \; dV d\phi \tag{7}$$

with the ballistic constraint that the corresponding missile strike range is given by

$$r = \frac{V^2}{g} \sin 2\phi \tag{8}$$

Using the variable transformation

$$\chi = V \sin 2\phi \tag{9}$$

we have from Equation (8) that

$$\phi = \frac{1}{2} \sin^{-1}\left(\frac{\chi^2}{rg}\right) \tag{10}$$

and

$$V = \frac{rg}{\chi} \tag{11}$$

for which the Jacobian is given by

$$|J| = \frac{1}{2r\sqrt{1 - \left(\frac{\chi^2}{rg}\right)^2}} \tag{12}$$

Using the Jacobian, the strike probability density per unit horizontal strike area, ρ_A can be written as

$$\rho_A \, dA = \iint_{\substack{\phi \; V \\ \phi, V \in dA}} \rho_V(V)\, \rho_\phi(\phi)\, dV d\phi = \int_\chi \rho_V(V(r,\chi))\, \rho_\phi(\phi(r,\chi))\, |J|\, dr d\chi \qquad (13)$$

where the incremental strike area dA is given by

$$dA = r d\theta dr \qquad (14)$$

Applying Equations (7), (10), (11), (12), and (14) to (13) we have

$$\rho_A dA = \int_\chi \left(\frac{1}{V_2 - V_1}\right)\left(\frac{\cos\phi}{2\pi \sin\Delta}\, d\theta\right)\left(\frac{1}{2r\sqrt{1-\left(\frac{\chi^2}{rg}\right)^2}}\right)\left(\frac{r}{r}\right) dr d\chi \qquad (15)$$

which yields

$$\rho_A = \left(\frac{1}{V_2 - V_1}\right)\left(\frac{1}{2\pi \sin\Delta}\right)\left(\frac{1}{2r^2}\right)\int_{\chi_{min}}^{\chi_{max}} \cos\left[\frac{1}{2}\sin^{-1}\frac{\chi^2}{rg}\right]\frac{d\chi}{\sqrt{1-\left(\frac{\chi^2}{rg}\right)^2}} \qquad (16)$$

The values χ_{min} and χ_{max} represent the limits on χ such that the target area dA at r is struck. These limits are subject to change as the azimuth angle θ and distance r to the target change due to the constraints imposed by the speed range V_1, V_2 and the deflection angles Δ. The variation of χ_{max} and χ_{min} can be illustrated as follows (Figure 2). Consider a qualitative graph of V versus ϕ as constrained by Equation (8) for some value of r and θ. The graph segment AB represents the locus of all combinations of $V_1 \leqslant V \leqslant V_2$ and $\frac{\pi}{4} \leqslant \phi \leqslant \frac{\pi}{2}$ which permit a missile to reach the target at r, θ as indicated in Figure 2. The variable χ can be expressed as

$$\chi = \sqrt{rg \sin 2\phi} \qquad (17)$$

Its graph versus ϕ is indicated by the dashed curve in Figure 2. The graph segment CD of χ versus ϕ represents the range of corresponding values of χ, such that in going from V_1, ϕ_1 to V_2, ϕ_2, the variable χ ranges from χ_{max} to χ_{min}. In this illustration, the limits on χ are dictated by the dynamic constraint given in Equation (8). The limits can be expressed by

$$\chi_{min} = \frac{rg}{V_2}, \quad \chi_{max} = \frac{rg}{V_1} \qquad (18)$$

As mentioned earlier, the deflection angles Δ represent an additional constraint which is illustrated in Figure 3 by the vertical line EF for a given azimuthal direction θ. In this case, missiles with speeds between V_1 and V_i cannot reach a target at r, θ since the necessary elevation angles below ϕ_i are not permitted by the constraint

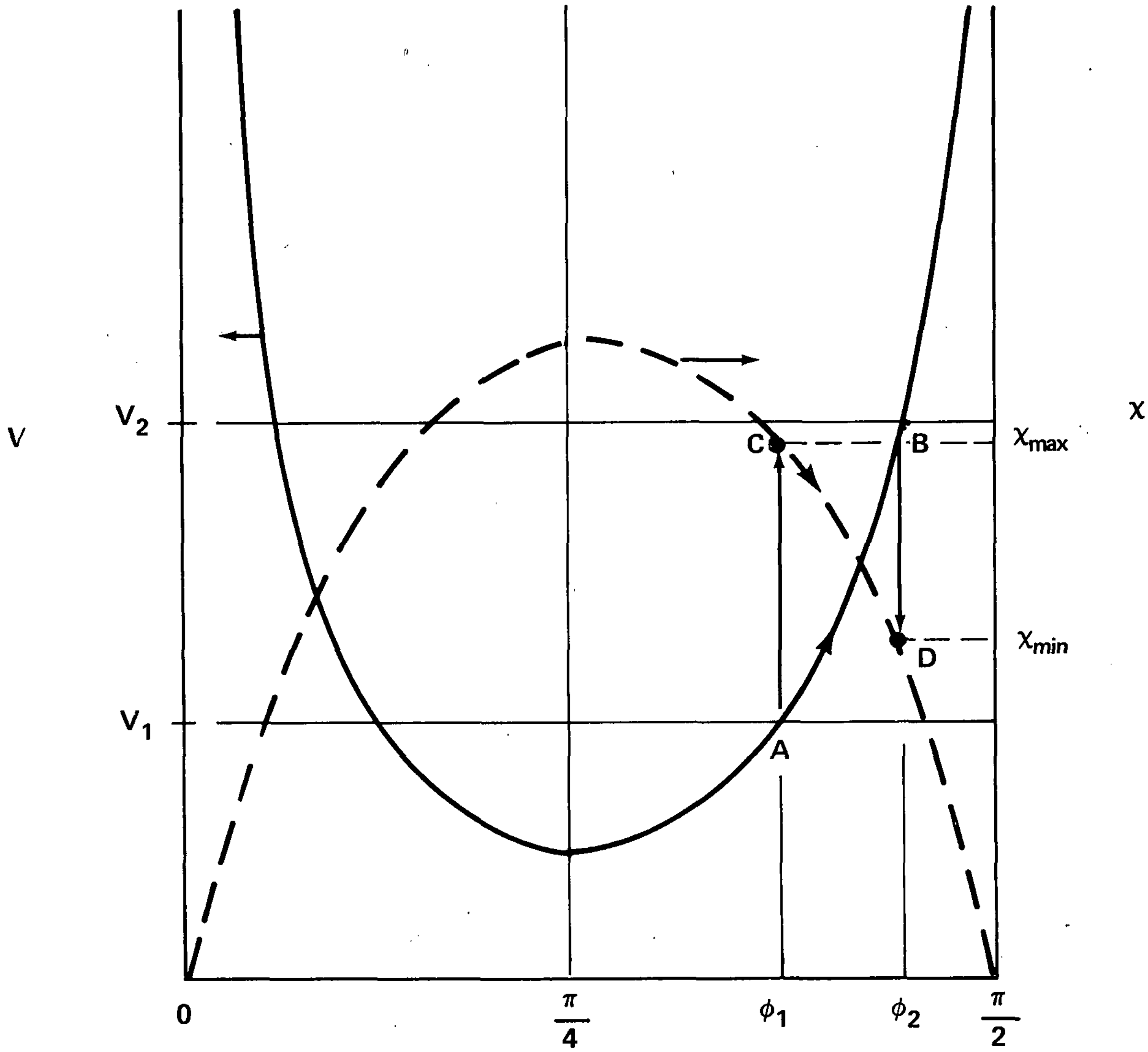

Figure 2. Dynamic Constraints on ϕ.

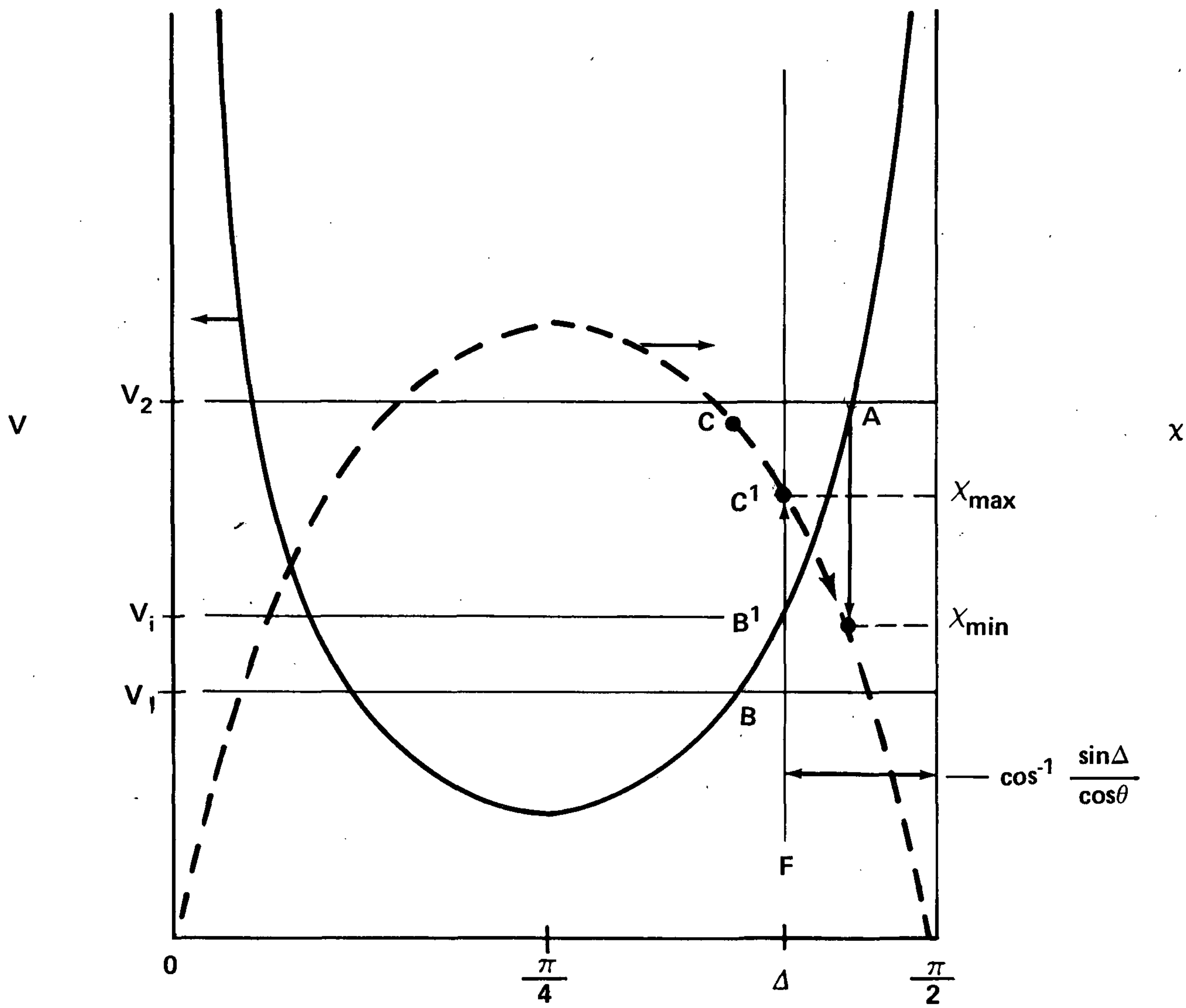

Figure 3. Deflection Angle Δ Constraint on ϕ.

$$\phi \geqslant \cos^{-1} \frac{\sin\Delta}{\cos\theta} \qquad (19)$$

In this case, the limits on X are of the form

$$X_{min} = \frac{rg}{V_2}\ , \quad X_{max} = \sqrt{rg \sin 2\left[\frac{\pi}{2} - \cos^{-1}\left(\frac{\sin\Delta}{\cos\theta}\right)\right]} \qquad (20)$$

Considering the typical values of V_1, V_2, r, and Δ for turbine units on nuclear power plant sites, it can be shown that the integrand of Equation (16) is a slowly varying function near unity. Thus, an approximate solution of Equation (16) is

$$\rho_A \cong \frac{X_{max} - X_{min}}{(V_2 - V_1)\, 4\pi r^2 \sin\Delta} \qquad (21)$$

Applying the limits in (18) and (20), we have

$$\rho_A \cong \frac{g}{V_1 V_2\, 4\pi r \sin\Delta} \qquad \text{for } \cos^{-1}\frac{\sin\Delta}{\cos\theta} > \frac{\pi}{2} - \frac{1}{2}\sin^{-1}\frac{rg}{V_1^2} \qquad (22)$$

$$\rho_A \cong \frac{\sqrt{rg \sin 2\left[\frac{\pi}{2} - \cos^{-1}\left(\frac{\sin\Delta}{\cos\theta}\right)\right]} - \frac{rg}{V_2}}{(V_2 - V_1)\, 4\pi r^2 \sin\Delta} \qquad \text{for } \cos^{-1}\frac{\sin\Delta}{\cos\theta} \leqslant \frac{\pi}{2} - \frac{1}{2}\sin^{-1}\frac{rg}{V_1^2} \qquad (23)$$

Figure 4 shows a plot of Equations (22) and (23) versus target distance for a speed range between 200 and 600 feet per second and several values of θ, where $\Delta = 5°$.

II. ESTIMATES OF THE PROBABILITY OF PENETRATION OF STRUCTURES BY TURBINE MISSILES

Estimates of the minimum reinforced concrete thickness required for preventing turbine missile penetration can be obtained using the Petry equation described in Reference 1. This equation is limited to estimating penetration depths in concrete. It does not take into account the possibility of concrete spalling. Suitable safety factors should be applied to the equation to account for spalling unless design features preclude spalling. Figure 5 illustrates the thickness T required to prevent penetration at various speeds, V, for various missile sizes and shapes (as characterized by the parameter A_p), where:

T = Minimum concrete thickness,

V = Missile strike speed,

A_p = Sectional Pressure = $\dfrac{\text{Missile Weight}}{\text{Cross Sectional Missile Area}}$

The curves in Figure 5 correspond to 5500 psi concrete.

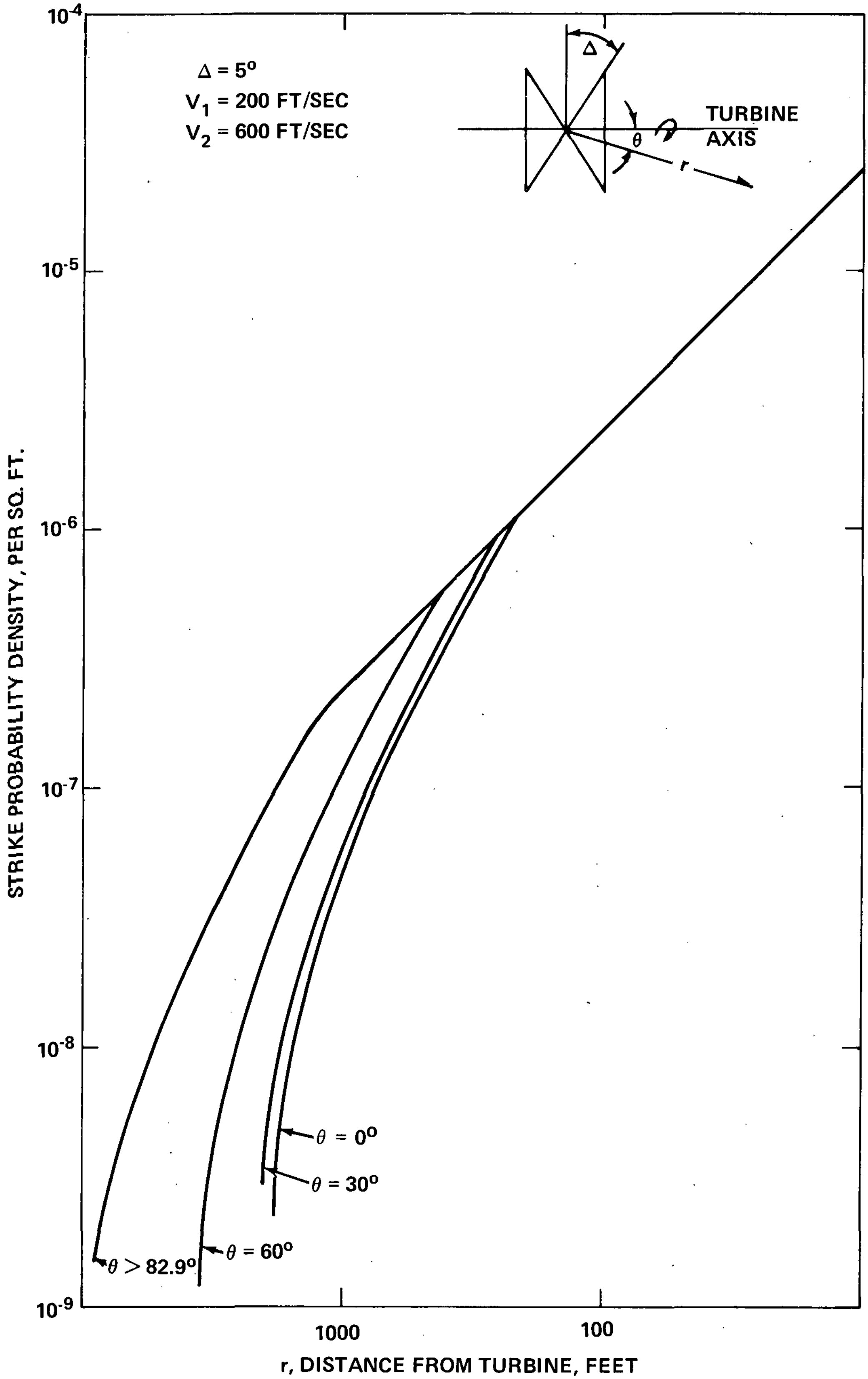

Figure 4. Strike Probability Density Verus Distance from Turbine

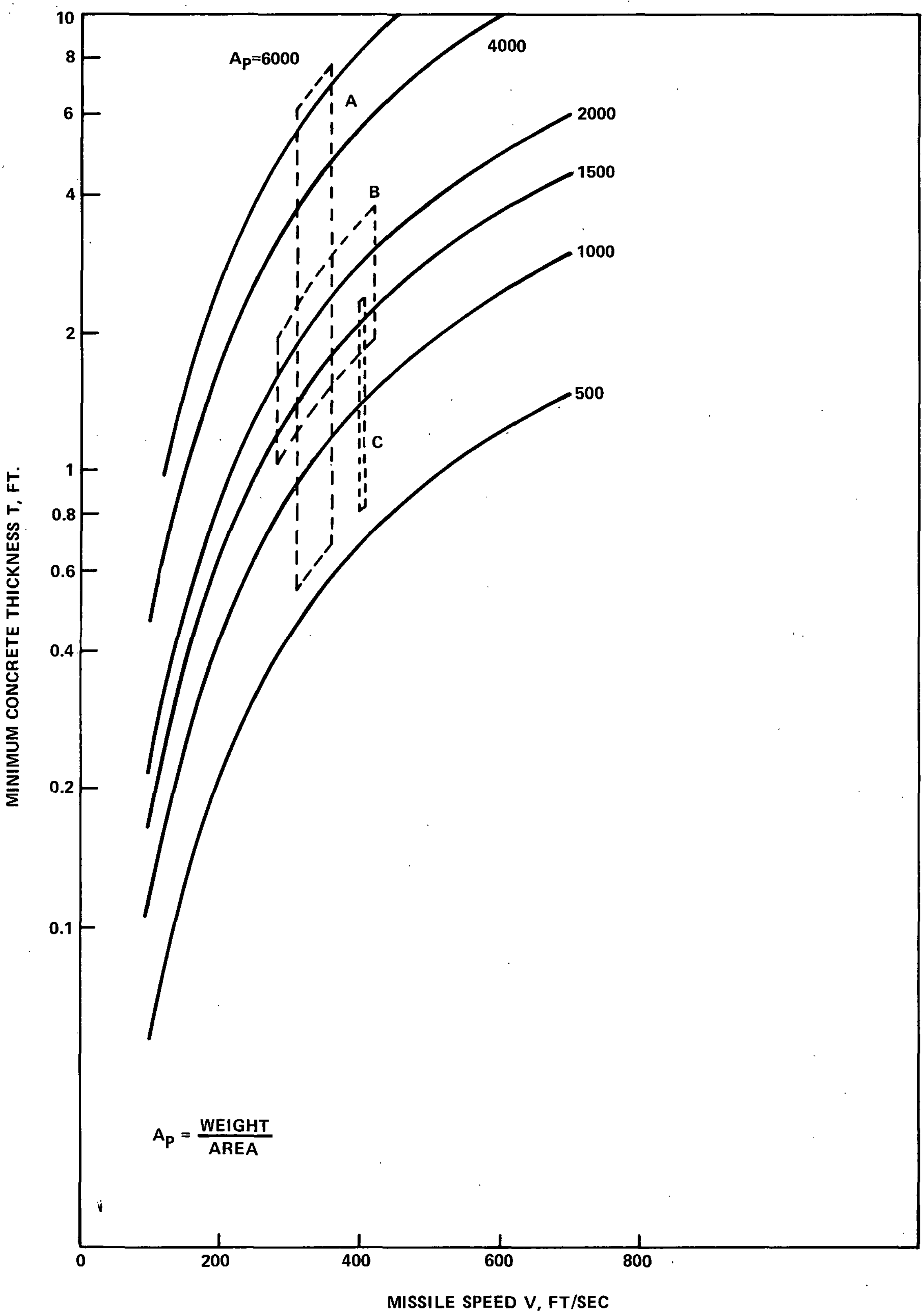

Figure 5. Minimum Concrete Thickness Versus Missile Speed for Various Missiles

The closed boundary areas indicated in Figure 5 represent the variation in missile speeds and missile orientations corresponding to several different examples of turbine missiles (Areas A, B, and C). It can be seen that for a given missile speed, the variation in sectional pressure A_p can be considerable, so that a considerable concrete thickness can be required to eliminate any possibility of penetration.

Considering the randomness of missile orientation, it is possible to introduce the concept of penetration probability, P_3, by assuming that the variation in A_p, and thus in T, is uniformly distributed between the minimum and maximum values for a particular turbine. We may write with respect to each type of turbine that

$$P_3 = \frac{T\ max - T}{T\ max - T\ min} \tag{1}$$

where T_{min} and T_{max} correspond to concrete thicknesses defined by the extreme values of the closed boundaries in Figure 5. Application of Equation (24) to each of the three turbine examples in Figure 5 yields penetration probability curves such as those shown in Figure 6. (Note that this represents an example where measures have been taken to preclude spalling.

III. REFERENCES

1. Bush, S. H., "Probability of Damage to Nuclear Components Due To Turbine Failure," Nuclear Safety, Vol. 14, No. 3, May-June 1973.

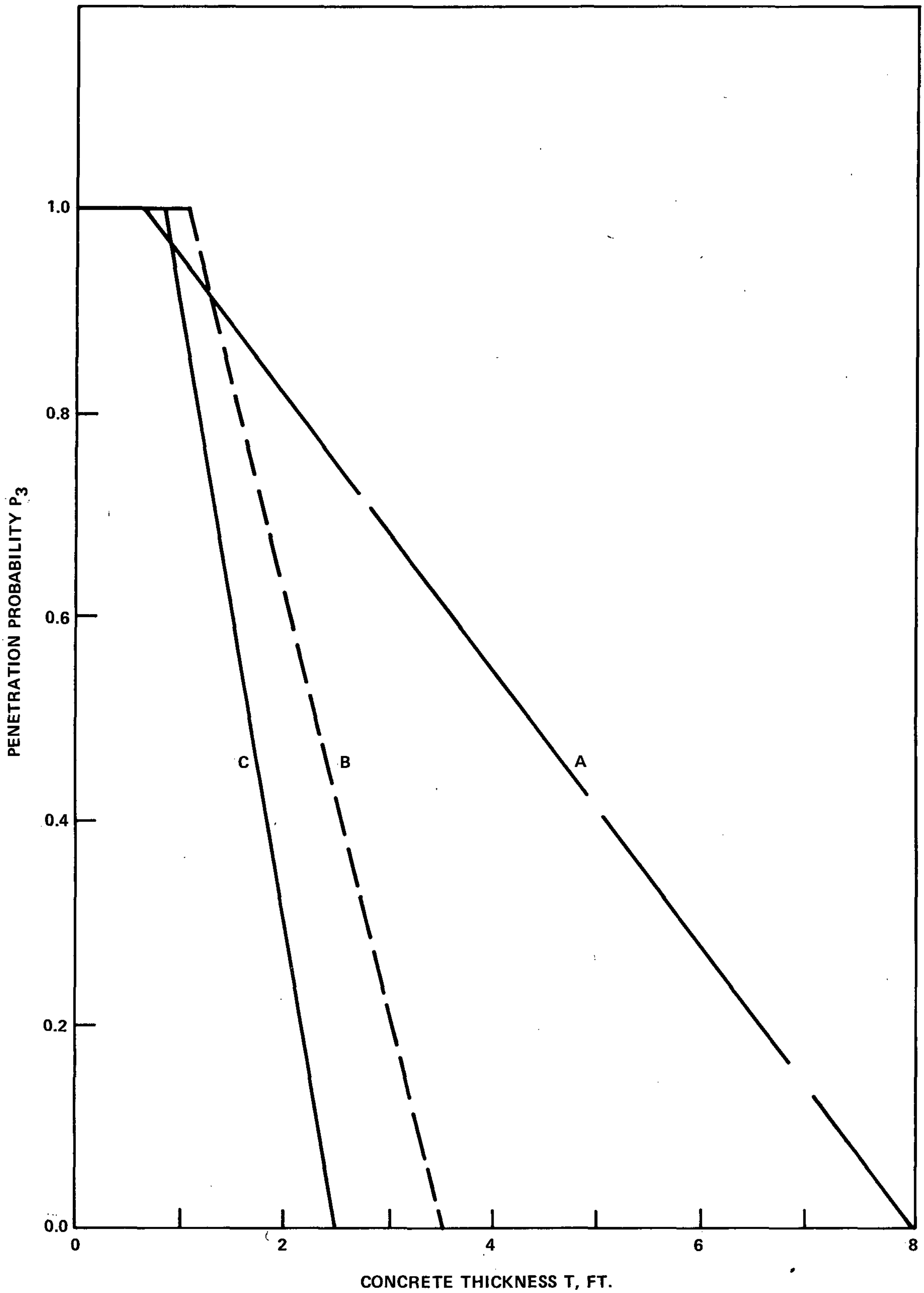

Figure 6. Penetration Probability P_3 Versus 5500 PSI Concrete Thickness Based on a Uniform Distribution in Missile Orientation

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.5.1.4 MISSILES GENERATED BY NATURAL PHENOMENA

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The applicant's assement of possible hazards due to missiles generated by the design basis tornado, flood, and any other natural phenomena identified in Section 2.2.3 of the safety analysis report (SAR) is reviewed. The purpose of the review is to assure that hazards due to these missiles are acceptably small so that they need not be included in the plant design basis, or that appropriate design basis missiles have been chosen and properly characterized. Currently, only missiles from the design basis tornado (Ref. 1) are considered in plant design bases.

The APCSB, under Standard Review Plan (SRP) 3.5.2 identifies those structures, systems, and components that should be protected against missile impact and the SEB, under SRP 3.5.3, assures that adequate protection is provided by structures and missile barriers.

II. ACCEPTANCE CRITERIA

1. The identification of appropriate design basis missiles generated by natural phenomena is considered acceptable if the methodology is consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in SRP 2.2.3 (Ref. 2).

2. The staff's position regarding the systems to be protected against tornado missiles is covered in Branch Technical Position AAB 3-2 (Ref. 3). A representative spectrum of tornado missiles is described in WASH-1361 (Ref. 4) and currently acceptable impact velocities are listed in item 4 under Review Procedures (Section III, below).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the area covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is to be based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The reviewer obtains from SAR Section 2.2.3 the identification of the design basis natural phenomena which could possibly generate missiles.

2. The total probability per year of missiles generated by a specific design basis phenomena striking a critical area of the plant is estimated. This total probability per year (P_T) may be estimated by using the following expression:

$$P_T = P_{NP} \times P_{MR} \times P_{SC} \times N$$

where

P_{NP} = frequency of occurrence (per year) of the design basis phenomenon (as calculated in SAR Section 2.2.3),

P_{MR} = probability of the generated missiles reaching the plant,

P_{SC} = probability of missiles that reach the plant striking a critical area of the plant, and

N = number of missiles generated by the design basis natural pnenomenon.

P_{MR} and P_{SC} are assumed to be equal to 1 unless analyses demonstrate lower values.

3. If P_T is greater than about 10^{-7} per year the reviewer should verify that the proper design basis events have been chosen and the missiles properly characterized.

4. All plants are required to be designed against tornado-generated missiles (i.e., the probability of a tornado strike is between 10^{-3} and 10^{-4} per year and therefore P_T is assumed greater than 10^{-7} per year). The following missiles (described in Ref. 4) and associated impact velocities are presently accepted as an adequate design basis until more definitive guidelines, based on the review of several topical reports and independent analytical work under way by the staff, are developed.

		Fraction of total tornado velocity
A.	Wood plank, 4 in. x 12 in. x 12 ft, weight 200 lb.	0.8
B.	Steel pipe, 3 in. diameter, schedule 40, 10 ft long, weight 78 lb.	0.4
C.	Steel rod, 1 in. diameter x 3 ft long, weight 8 lb.	0.6
D.	Steel pipe, 6 in. diameter, schedule 40, 15 ft long, weight 285 lb.	0.4
E.	Steel pipe, 12 in. diameter, schedule 40, 15 ft long, weight 743 lb.	0.4
F.	Utility pole, 13-1/2 in. diameter, 35 ft long, weight 1490 lb.	0.4
G.	Automobile, frontal area 20 ft^2, weight 4000 lb.	0.2

These missiles are considered to be capable of striking in all directions. Missiles A, B, C, D, and E are to be considered at all elevations and missiles F and G at elevations up to 30 feet above all grade levels within 1/2 mile of the facility structures.

The staff has, as an interim position, accepted the "no-tumbling" horizontal missile velocities presented in the Topical Report TVA-TR74-1 (Refs. 5 and 6) provided that a 4000-lb automobile at 70 mph and elevations up to 30 feet above grade level is added. These velocities are:

		Horizontal Velocity ft/sec
A.	Wood plank, 4 in. x 12 in. x 12 ft, weight 200 lb.	368
B.	Steel pipe, 3 in. diameter, schedule 40, 15 ft long, weight 115 lb.	268
C.	Steel rod, 1 in. diameter x 3 ft long, weight 8 lb.	259
D.	Steel pipe, 6 in. diameter, schedule 40, 15 ft long, weight 285 lb.	230
E.	Steel pipe, 12 in. diameter, schedule 40, 30 ft long weight 1500 lb.	205
F.	Utility pole, 14 in. diameter, 35 ft long, weight 1500 lb.	241
G.	Automobile, frontal area 20 ft^2, weight 4000 lb.	100

Vertical velocities equal to 80% of the TVA horizontal velocities are also acceptable on an interim basis.

At the operating license stage, applicants who were not required at the construction permit stage to design to one of the above missile spectra and the corresponding velocity set, should show the capability of the existing structures and components to withstand at least missiles "C" and "F." The adequacy of existing protection and any requirements for improvements will be determined on a case-by-case basis in conjunction with APCSB. The AAB Branch Chief should be consulted in making such determinations.

5. The capability of structures to withstand the postulated missile impacts is reviewed by the SEB and vital target areas are defined by the APCSB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

"These analyses result in a probability of missiles generated by ______________ having consequences worse than the design basis accident of less than 10^{-7} per year. We, therefore, conclude that the probability of missile impacts due to __________ causing radiological consequences greater than the design basis events analyzed is so small that it does not present an undue risk to the health and safety of the public.

"These analyses verify that design basis missiles have been properly chosen and characterized."

V. REFERENCES

1. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

2. Standard Review Plan 2.2.3, "Evaluation of Potential Accidents."

3. Branch Technical Position AAB 3-2, "Tornado Design Classification," attached to this plan.

4. "Safety-Related Site Parameters for Nuclear Power Plants," WASH-1361, U. S. Atomic Energy Commission (1975).

5. "The Generation of Missiles by Tornadoes," TVA-TR74-1, Tennessee Valley Authority (1974). (Topical report under review by the staff.)

6. Regulatory Staff, "Preliminary Evaluation of Topical Report TVA-TR74-1," U. S. Nuclear Regulatory Commission, February 1975.

BRANCH TECHNICAL POSITION AAB 3-2

TORNADO DESIGN CLASSIFICATION

A. BACKGROUND

General Design Criterion 2 requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as tornadoes without loss of capability to perform their safety functions. Criterion 2 also requires that the design bases for these structures, systems, and components reflect (1) appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena and (2) the importance of the safety functions to be performed.

General Design Criterion 4 requires, in part, that structures, systems, and components important to safety, be protected against the effects of missiles from events and conditions outside the plant.

Nuclear power plants should be designed so that the plants can be placed and maintained in a safe shutdown condition in the event of the most severe tornado that can reasonably be predicted to occur at a site as a result of severe meteorological conditions. Protection of structures, systems, and components necessary to place and maintain the plant in a cold shutdown condition may generally be accomplished by designing protective barriers to preclude missile strikes. For example, the primary containment, reactor building, auxiliary building, and control structures should be designed against collapse and should provide an adequate barrier against missiles. However, the primary containment need not necessarily maintain its leak-tight integrity under pressure loadings due to the pressure differentials developed by the tornado. If protective barriers are not installed, the structures and components themselves should be designed to withstand the effects of the tornado, including tornado missile impacts.

It is not necessary to maintain the functional capability of all seismic Category I structures, because the combined probability of a joint occurrence of low probability events (loss-of-coolant accident with design basis or smaller tornado, or earthquake and design basis or smaller tornado) is so small as to not warrant consideration in the plant design basis. However, a source of water should be available to provide long-term core cooling.

Similarly, it is not necessary to protect radioactive liquid waste holdup tanks since even in the event of gross failure, the spills would be limited to small amounts of waste and would be expected to be collected in the building foundations, which are designed for that purpose.

Structures, systems, and components important to safety which should be designed to withstand the effects of a design basis tornado are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary.

2. The capability to shut down the reactor and maintain it in a safe shutdown condition.

3. The capability to prevent accidents which could result in potential offsite exposures that are a significant fraction of the guideline values of 10 CFR Part 100. Designs which differ substantially from those now in use may require reevaluation with respect to this objective.

The physical separation of redundant or alternative structures or components required for the safe shutdown of the plant is generally not considered an acceptable method for protecting against tornado effects, including tornado-generated missiles.

This branch position describes a method acceptable to the staff for identifying those structures, systems, and components of light-water reactors which should be designed to withstand the effects of the design basis tornado (as defined by Regulatory Guide 1.76), including tornado missiles, and to remain functional.

B. BRANCH TECHNICAL POSITION

1. Those structures, systems, and components, including foundations and supports, which should be designed to withstand the effects of a design basis tornado (as defined in Regulatory Guide 1.76), including tornado missiles, without loss of capability to perform essential safety functions are listed below.

   a. The reactor coolant pressure boundary.[1/]

   b. Those portions of the main steam and main feedwater systems of pressurized water reactors (PWRs) up to and including the outermost isolation valves.

   c. The reactor core and reactor vessel internals.

   d. Systems[2/] or portions of systems, and those auxiliary systems necessary to support these systems (for example, service water, cooling water source, component cooling and auxiliary feedwater) that are required for (1) reactor shutdown, (2) residual heat removal, (3) cooling the spent fuel storage pool, or (4) makeup water for the primary system.

   e. The spent fuel storage facility to the extent necessary to preclude significant loss of watertight integrity of the storage pool and to prevent missiles from contacting fuel within the pool.

   f. The reactivity control systems, e.g., control rod drives and boron injection systems.

[1/] As defined in 10 CFR § 50.2

[2/] The system boundary includes those portions of the system required to accomplish the specified safety function and connecting piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

g. The control room, including its associated vital equipment, cooling systems for the vital equipment and life support systems, and any structures or equipment inside or outside of the control room whose failure could result in an incapacitating injury to individuals occupying the control room.

h. Those portions of the gaseous radwaste treatment systems which by design are intended to store or delay gaseous radioactive waste and portions of structures housing these systems including isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system (e.g., charcoal delay tanks in a boiling water reactor (BWR) plant and waste gas storage tanks in a PWR plant).

i. Systems or portions of systems that are required for (1) monitoring systems important to safety and (2) actuating and operating systems important to safety.

j. All electric and mechanical devices and circuits between the process sensors and the input terminals of the actuator systems involved in generating signals that initiate protective action.

k. Those portions of the long-term emergency core cooling system that would be required to maintain the plant in a safe condition for an extended time after a loss-of-coolant accident.

l. Primary reactor containment and other safety-related structures, such as the control room building and auxiliary building, should be protected against collapse. The primary containment need not necessarily maintain its leak-tight integrity under pressure loadings due to pressure differentials developed by the tornado, tornado-borne missiles which could jeopardize contained safety-releated systems and components.

m. Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies that provide emergency electric power needed for functioning of plant features included in items a through k above.

2. Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce to an unacceptable safety level the functional capability of any feature included in the items listed above should be designed and constructed so that the effects of the design basis tornado would not cause failure (for example, of the containment walls).

C. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.5.1.5 SITE PROXIMITY MISSILES (EXCEPT AIRCRAFT)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The staff reviews the applicant's assessment of possible hazards due to missiles generated by the design basis explosions identified in Section 2.2 of the safety analysis report (SAR). The purpose of the review is to assure that hazards due to there missiles are acceptably small so that they need not be included in the plant design basis, or that appropriate design basis missiles have been chosen and properly characterized. The APCSB determines those systems and components that should be protected against missile impacts, and the SEB assures that adequate protection is provided.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against site proximity missiles if the resulting probability of a missile affecting the safety-related features of the plant is within the guidelines established in Section II of Standard Review Plan 2.2.3.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The identification of accidents which could possibly generate missiles is obtained from Section 2.2 of the SAR.

2. The total probability of the missiles striking a critical area of the plant is estimated. The total probability per year (P_T) may be estimated by using the following expression:

$$P_T = P_E \times P_{MR} \times P_{SC} \times P_P \times N$$

where:

P_E = probability per year of design basis explosion calculated in Section 2.2,

P_{MR} = probability of missiles reaching the plant,

P_{SC} = probability of missiles striking a critical area of the plant,

P_P = probability of missiles exceeding the energies required to penetrate to vital areas (e.g., based on wall thickness provided for tornado missiles), and

N = number of missiles generated by the design basis explosion.

P_{MR}, P_{SC} and P_P are assumed to be equal to 1 unless the analyses in this section demonstrate lower values.

3. If P_T is greater than about 10^{-7} per year, the reviewer should verify that the proper design basis events have been chosen and the missiles properly characterized.

4. The capability of structures to withstand the postulated missile impacts will be reviewed by the SEB, and the vital target areas will be defined by the APCSB.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

1. "The staff analysis shows that the probability of an accident having serious radiological consequences is extremely remote and is within the guidelines established for low probability events of site proximity missiles. We conclude, therefore, that the probability of a missile impact causing radiological consequences of the order of 10 CFR Part 100 guidelines is so small that such an event does not present an undue risk to the health and safety of the public."

or

2. "The staff analyses verify that a design basis missile impact has been properly chosen and characterized."

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

2. Regulatory Guide 1.76, "Design Bases Tornado for Nuclear Power Plants."

3. Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites."

4. Standard Review Plan 2.2.3, "Evaluation of Potential Accidents."

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SECTION 3.5.1.6 AIRCRAFT HAZARDS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Structural Engineering Branch (SEB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

The staff reviews the applicant's assessment of aircraft hazards to the plant. The purpose of the review is to assure that either aircraft hazards are eliminated as a design basis concern or appropriate design basis aircraft have been chosen and properly characterized as to impact and fire hazards. The review also involves a determination of adequate protection against fire hazards for design basis events. Some information relating to this review is contained in Section 2.2 of the applicant's safety analysis report (SAR), e.g., facility locations, projected traffic, and accident statistics.

The APCSB determines which structures and components are to be protected, and the SEB assures that adequate protection has been provided.

II. ACCEPTANCE CRITERIA

1. The plant is considered adequately designed against aircraft hazards if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10^{-7} per year (see Standard Review Plan 2.2.3).

2. The probability is generally considered acceptable by inspection if the level of aircraft activity near the site falls below the criteria given in Section 2.2.3 of Regulatory Guide 1.70 (Ref. 2) for analysis of hazards due to commercial, experimental, and general aviation aircraft. For military airspace, a minimum distance of five miles from the reactor is adequate for low level training routes except those associated with usage greater than 1000 flights per year or activities (such as practice bombing) where an unusual stress situation exists.

3. Aircraft accidents which could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than about 10^{-7} per year should be considered in the design of the plant.

4. The evaluation of fire hazards will be done on an individual case basis. Concrete structures are generally assumed to withstand fire, but protection must be provided to prevent fire, smoke, or flammable mixtures from entering safety-related ventilation intakes, such as those for the control room, areas housing shutdown equipment, and the diesel generators.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The staff's review of the aircraft hazard assessment consists of the following steps:

1. Data describing aviation uses in the airspace near the proposed site, including airports and their approach paths, federal airways, Federal Aviation Administration (FAA) restricted areas, and military uses is obtained from Section 2.2 of the SAR. For many cases, no detailed analysis need be made as the probability can be judged adequately low based on a comparison with analyses previously performed. In such cases the conclusion reached and a citation of the cases used for comparison should be transmitted by buck slip to the AAB site analyst for retention in the case workbook.

2. For situations where federal airways or aviation corridors pass through the vicinity of the site, the probability per year of an aircraft crashing into the plant (P_{FA}) should be estimated. This probability will depend on a number of factors such as the altitude and frequency of the flights, the width of the corridor, and the corresponding distribution of past accidents.

   One way of calculating P_{FA} is by using the following expression:

   $$P_{FA} = C \times N \times A / w$$

   where:

   C = inflight crash rate per mile for aircraft using airway,

   w = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles,

   N = number of flights per year along the airway, and

   A = effective area of plant in square miles.

   This gives a conservative upper bound on aircraft impact probability if care is taken in using values for the individual factors that are meaningful and conservative. For commercial aircraft a value of $C = 3 \times 10^{-9}$ per aircraft mile has been used. For heavily traveled corridors (greater than 100 flights per day), a more detailed analysis may be required to obtain a proper value for this factor.

3. The probability of an aircraft crashing into the site should be estimated for cases where either of the following apply:

   a. An airport is located within five miles of the site.

   b. An airport with projected operations greater than 500 d^2 movements per year is located within ten miles of the site, or an airport with projected operations greater than 1000 d^2 movements per year is located beyond ten miles from the site, where "d" is the distance in miles from the site.

   The probability per year of an aircraft crashing into the site for these cases (P_A) may be calculated by using the following expression:

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} C_j \; N_{ij} \; A_j$$

   where:

   M = number of different types of aircraft using the airport,

   L = number of flight paths affecting the site,

   C_j = probability per square mile of a crash per aircraft movement, for the jth aircraft,

   N_{ij} = number (per year) of movements by the jth aircraft along the ith flight path, and

   A_j = effective plant area (in square miles) for the jth aircraft.

   As noted earlier, the choice of values for the parameters should be made judiciously in order to arrive at a meaningful result. The manner of interpreting the individual factors may vary on a case-by-case basis because of the specific conditions of each case or because of changes in aircraft accident statistics.

   Values for C_j currently being used are taken from the data summarized in the following table:

Distance From End of Runway (miles)	Probability (x 10^8) of a Fatal Crash per Square Mile for Aircraft Movements			
	U.S. Air Carrier[1]	General Aviation[2]	USN/USMC[1]	USAF[1]
0-1	16.7	84	8.3	5.7
1-2	4.0	15	1.1	2.3
2-3	0.96	6.2	0.33	1.1
3-4	0.68	3.8	0.31	0.42
4-5	0.27	1.2	0.20	0.40
5-6	0	NA[3]	NA	NA
6-7	0	NA	NA	NA
7-8	0	NA	NA	NA
8-9	0.14	NA	NA	NA
9-10	0.12	NA	NA	NA

[1] Reference 2.

[2] Reference 4.

[3] NA indicates that data was not available for this distance.

4. For military installations or any other airspace usages, a detailed quantitative modeling of all operations should be verified. The result of the model should be the total probability (C) of an aircraft crash per unit area and time in the vicinity of the proposed site.

   The probability per year of a potentially damaging crash at the site due to operations at the facility under consideration (P_M) is then given for this case by the following expression:

   $$P_M = C \times A$$

   where:

   C = total probability of an aircraft crash per square mile per year in the vicinity of the site, and

   A = effective area of the plant in square miles.

5. The total aircraft hazard probability at the site equals the sum of the individual probabilities obtained in the preceding steps.

6. The effective plant areas used in the calculations should include the following:

   a. A shadow area of the plant elevation upon the horizontal plane based on the assumed crash angle for the different kinds of aircraft and failure modes.

b. A skid area around the plant as determined by the characteristics of the aircraft under consideration. Artificial berms or any other man-made and natural barriers should be taken into account in calculating this area.

c. Areas of the plant susceptible to structural damage as a result of aircraft impact.

d. Areas of the plant susceptible to fire hazards resulting from aircrift accidents on the site.

For those classes of aircraft hazard having a probability of occurrence of causing radiological consequences in excess of 10 CFR Part 100 guidelines greater than about 10^{-7} per year, the reviewer should verify that the proper design basis events have been chosen and the aircraft properly characterized in terms of impact and fire parameters.

The capability of structures to withstand the postulated aircraft impacts will be reviewed by the SEB, and the vital target areas will be defined by the APCSB. In the past, external fire effects have been evaluated by the AAB with assistance from consultants (Ref. 3), but the APCSB will review this area for future applications.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and drafts an introductory paragraph for the evaluation findings indicating those facilities described in SAR Section 2.2 for which an aircraft hazards analysis was performed. A brief description of the methods used in the analysis should be provided, together with references to any sources of statistical data utilized.

The reviewer also verifies that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

1. "The applicant's assessment of aircraft hazards at the site has been independently verified by the staff and results in a probability less than about 10^{-7} per year of an accident having radiological consequences worse than the exposure guidelines of 10 CFR Part 100. We conclude, therefore, that operation of the ____________ plant in the vicinity of ____________ does not present an undue risk to the health and safety of the public."

2. "Plant sites reviewed in the past which had equivalent aircraft traffic in equal or closer proximity were, after careful examination, found to present no undue risk to the safe operation of those plants. Based upon this experience, in the staff's judgment, no undue risk is present from aircraft hazard at the plant site now under consideration."

3. "The applicant's assessment of aircraft hazards at the site has been independently verified by the staff and we corroborate that if the plant (or appropriate parts of

the plant) is designed to withstand the aircraft selected as the design basis aircraft, the probability of an aircraft strike causing radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 is less than about 10^{-7} per year. We conclude, therefore, that the operation of the ___________ plant in the vicinity of ___________ does not present an undue risk to the health and safety of the public."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. D. G. Eisenhut, "Reactor Siting in the Vicinity of Airfields." Paper presented at the American Nuclear Society Annual Meeting, June 1973.

3. I. I. Pinkel, "Appraisal of Fire Effects from Aircraft Crash at Zion Power Reactor Facility," July 17, 1972 (Docket No. 50-295).

4. D. G. Eisenhut, "Testimony on Zion/Waukegan Airport Interaction" (Docket No. 50-295).

5. USAEC Regulatory Staff, "Safety Evaluation Report," Appendix A, "Probability of an Aircraft Crash at the Shoreham Site" (Docket No. 50-322).

6. "Addendum to the Safety Evaluation by the Division of Reactor Licensing, USAEC, in the Matter of Metropolitan Edison Company (Three Mile Island Nuclear Station Unit 1, Dauphin County, Pennsylvania)," April 26, 1968 (Docket No. 50-289).

7. Letter to Honorable J. R. Schlesinger from S. H. Bush, Chairman, Advisory Committee on Reactor Safeguards, "Report on Rome Point Nuclear Generating Station," November 18, 1971 (Project No. 455).

8. Letter to Mr. Joseph L. Williams, Portland General Electric Company, from R. C. DeYoung (in reference to Mr. Williams' letter of May 7, 1973), November 23, 1973 (Project No. 485).

9. "Aircraft Considerations-Preapplication Site Review by the Directorate of Licensing, USAEC, in the Matter of Portland General Electric Company, Boardman Nuclear Plant, Boardman, Oregon," October 12, 1973 (Project No. 485).

10. Letter to Mr. J. H. Campbell, Consumers Power Company, from Col. James M. Campbell, Dep. Chief, Strategic Division, Directorate of Operations, U. S. Air Force, May 19, 1971 (Docket No. 50-155).

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SECTION 3.5.2 STRUCTURES, SYSTEMS, AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - None

I. AREAS OF REVIEW

The APCSB review of the structures, systems, and components (SSC) to be protected from externally generated missiles includes all safety-related SSC on the plant site that have been provided to support the reactor facility. These include such elements as service water intakes, buried components (e.g., service water piping, storage tanks), and access openings and penetrations in structures. The intent of the review is to verify that the applicant's list of SSC requiring protection against externally generated missiles is complete.

The APCSB reviews the functional operations or performance requirements for structures, systems, and components and identifies which of these are necessary for the safety of the reactor during all operating conditions including normal operations and operational transients, and during accidents. Safety-related SSC are so designated if their function is required for attaining and maintaining a safe shutdown condition during normal or accident conditions, mitigating the consequences of an accident, or preventing the occurrence of an accident.

Based on their relation to safety, structures or areas of structures, systems or portions of systems, and components are identified as requiring protection from externally generated missiles if a missile could prevent the intended safety function.

II. ACCEPTANCE CRITERIA

Acceptability of the list of structures, systems, and components to be protected against externally generated missiles, presented in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for the acceptability of the listing of structures, systems, and components that require missile protection will be the similarity of the design with those of previously approved plants.

The identification of structures, systems, and components to be protected against externally generated missiles is acceptable if it is in accordance with the following criteria:

1. General Design Criterion 4, with respect to protection of structures, systems, and components against the effects of externally generated missiles to maintain their essential safety functions.

2. Regulatory Guide 1.13, as related to the spent fuel pool systems and structures being capable of withstanding the effects of externally generated missiles and preventing missiles from contacting stored fuel assemblies.

3. Regulatory Guide 1.27, as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of externally generated missiles.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the applicant's list of SSC that require protection from externally generated missiles is complete and meets the acceptance criteria given in Section II of this plan. For operating license (OL) applications, the procedures are used to verify that the CP-stage list continues to be applicable and complete, or has been supplemented as appropriate. The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

The first step in the review under this plan is to determine the safety-related SSC. Some structures and systems are considered safety-related in their entirety, others have only portions that are safety-related, and others are classified as non-safety-related. In order to determine the safety category of the SSC, the APCSB evaluates the SSC of the facility with respect to their necessity for achieving and maintaining safe reactor shutdown, or for performing accident prevention or mitigation functions. The information provided in the SAR pertaining to SSC design bases, design criteria, and descriptions and safety evaluations, together with the system and component characteristics tables and safety classification tables are reviewed to identify safety functions performed during all operating conditions. The safety functions to be performed by the SSC in various designs remain essentially the same. However, the location of the SSC and the methods used vary from plant to plant depending upon the individual designer. The reviewer identifies variations in designs and evaluates them on a case-by-case basis.

The second step in the review is to determine the SSC, or portions of SSC, that require protection against externally generated missiles. The reviewer uses engineering judgment and the results of failure modes and effects analyses in conjunction with the results of reviews under other plans of specific SSC in determining the need for missile protection. Most safety-related systems are located within structures that are resistant to external missiles by virtue of design for other purposes (e.g., primary containment), or because the structures are constructed specifically to withstand missiles. Systems and components located within such structures are considered adequately protected. The reviewer concentrates his attention on safety-related SSC located outside such structures and on penetrations and access openings in the structures. Essential service water piping and components, storage tanks, and ultimate heat sink components are examples of SSC typically located outside missile-resistant structures. Depending on the nature and source of the externally generated missiles, protection may be provided by missile barriers or by suitable separation of independent redundant systems. Specific missile sources and the protection needed are considered in other standard review plans in the 3.5.1 series.

The reviewer compares his evaluation of SSC to be protected against externally generated missiles with the applicant's list of such SSC.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The review of the structures, systems, and components to be protected from externally generated missiles included all safety-related structures, systems, and components provided to support the reactor facility. The scope of review of the structure, systems, and components to be protected from externally generated missiles for the _______ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for all structures, systems, and components that are essential to the safe operation and safe shutdown of the plant. The review has included the applicant's proposed design criteria, design bases, and safety classifications for all systems, structures, and components.
>
> "The basis for acceptance of the list of structures, systems, and components to be protected from externally generated missiles has been conformance of the applicant's designs, design criteria, and design bases for structures, systems, and components to the Commission's regulations as set forth in General Design Criterion 4, and to applicable regulatory guides, staff technical positions, and industry standards.
>
> "The staff concludes that the designation of structures, systems, and components requiring external missile protection conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

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SECTION 3.5.3 BARRIER DESIGN PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to procedures utilized in the design of seismic Category I structures, shields, and barriers to withstand the effects of missile impact are reviewed.

1. Procedures utilized for the prediction of local damage in the impacted area. This includes estimation of the depth of penetration and, in case of concrete barriers, the potential for generation of secondary missiles by spalling or scabbing effects.

2. Procedures utilized for the prediction of the overall response of the barrier or portions thereof due to the missile impact. This includes assumptions on acceptable ductility ratios where elasto-plastic behavior is relied upon, and procedures for estimation of forces, moments, and shears induced in the barrier by the impact force of the missile.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. For Local Damage Prediction

a. In Concrete

Among the empirical equations available to estimate missile penetration into concrete barriers, the one most commonly used is the modified Petry equation, as given by A. Amirikian (Ref. 1). The use of this equation is acceptable. However, other equations may be used provided the results obtained are either comparable to those obtained from the modified Petry equation, or penetration tests are conducted to validate the equation used. Sufficient thickness of concrete should be provided to prevent perforation and to prevent spalling or scabbing, when protection from spalling or scabbing is required. To prevent perforation, a concrete thickness of at least twice the penetration thickness determined for an infinitely thick slab is acceptable. When spalling or scabbing is critical, the procedures used to determine the required thickness are reviewed on a case-by-case basis.

b. In Steel

The results of tests conducted by the Stanford Research Institute on the penetration of missiles into steel plates are summarized by W. B. Cottrell and A. W. Savolainen in "U.S. Reactor Containment Technology" (Ref. 2). The equations presented in reference 2 are acceptable. Other equations may be used provided the results are either comparable to those referenced above, or are validated by penetration tests.

c. In Composite Sections

For composite or multi-element missile barriers, procedures for prediction of local damage are acceptable if the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. For determining this residual velocity, the equations presented by Recht and Ipson (Ref. 3) are acceptable when the first barrier of a multi-element missile barrier is steel. When the first barrier is concrete, procedures are reviewed on a case-by-case basis.

2. For Overall Damage Prediction

The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., mid-span of a slab or near a support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. In general, the assumption of plastic collisions is acceptable, where all of the missile initial momentum is transferred to the target and only a portion of its kinetic energy is absorbed as strain energy within the target. However, where elastic impacts are expected, the additional momentum transferred to the target by missile rebound should be included.

After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area should then be determined, from which the structural response, in conjunction with other design loads, can be evaluated using conventional design methods. An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in a paper by Williamson and Alvy (Ref. 4). Other procedures may be used provided the results obtained are comparable to those referenced above.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. For the prediction of local damage, the equations proposed by the applicant for estimation of missile penetration are reviewed in the following manner:

   a. For missile penetration in concrete, the reviewer verifies that the applicant has made a commitment to utilize the modified Petry formula. If other equations are selected, the applicability and validity of such equations are reviewed to determine that the results are at least as conservative as those obtained from the modified Petry formula. If sufficient justification for the use of alternate equations is not provided, additional information is requested from the applicant at the first stage of the review. The reviewer also verifies that the applicant has made a commitment to provide sufficient barrier thickness to prevent perforation and to prevent spalling or scabbing when protection from spalling or scabbing is considered necessary.

b. For missile penetration in steel, the reviewer verifies that the applicant has made a commitment to utilize the Stanford equations. If other equations are selected, the applicability and validity of such equations are reviewed to assure that the results are at least as conservative as those obtained from the Stanford equations. If sufficient justification for the use of alternate equations is not provided, additional information is requested from the applicant at the first stage of the review.

c. For missile penetration in composite or multi-element barriers, the reviewer verifies that the applicant has made a commitment to utilize the criteria delineated in Section II.1.c of this plan. If other criteria are proposed, the justification provided is reviewed to assure that such equations give results which are at least as conservative as those referenced above.

2. For the prediction of overall damage and response of the barrier, the reviewer verifies that the applicant has made a commitment to utilize the criteria delineated in Section II.2 of this plan. If other criteria are selected, the applicant's justification is reviewed to assure that the results obtained are at least as conservative as those delineated in Section II.2. If sufficient justification is not provided, additional information is requested from the applicant at the first stage of the review.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The procedures utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structures or barriers are adequately resistant to and will withstand the effects of such forces.

"The use of these procedures provides reasonable assurance that in the event of design basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. A. Amirikian, "Design of Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D.C. (August 1950).

2. W. B. Cottrell and A. W. Savolainen, "U.S. Reactor Containment Technology," ORNL-NSIC-5, Vol. 1, Chapter 6, Oak Ridge National Laboratory.

3. R. F. Recht and T. W. Ipson, "Ballistic Perforation Dynamics," Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963.

4. R. A. Williamson and R. R. Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver, Inc., Revised November 1973.

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

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SECTION 3.6.1 PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The plant design for protection against piping failures outside containment is reviewed to assure that such failures would not cause the loss of needed functions of safety-related systems and to assure that the plant could be safely shut down in the event of such failures. The review includes high energy and moderate energy fluid system piping located outside of containment. If such a system penetrates containment (except for the auxiliary feedwater system) the review starts with the first isolation valve outside of containment. The review boundary for auxiliary feedwater systems extends either to the steam generator or to the feedwater (or steam) line, as appropriate. The specific areas of review are as follows:

1. APCSB reviews the general layout of high and moderate energy piping systems with respect to the plant arrangement criteria of Section B.1 of Branch Technical Position (BTP) APCSB 3-1, which is attached to this plan. Three arrangement situations are covered by the criteria and all three may be encountered in a single plant. They are:

   a. Arrangements where protection of safety-related plant features is provided by separation of high and moderate energy systems from essential systems and components.

   b. Arrangements where protection of safety-related plant features is provided by enclosing either the high and moderate energy systems or the safety-related features in protective structures.

   c. Arrangements where neither separation nor protective enclosures are practical and special protective measures are taken to ensure the operability of safety-related features.

2. APCSB, in conjunction with the secondary review branches as detailed below, reviews design features recommended in Section B.2 of BTP APCSB 3-1 as follows:

a. APCSB confirms with the RSB the seismic design classifications of systems and components defined as essential safety-related features in Appendix A of BTP APCSB 3-1.

b. APCSB identifies protective structures, piping restraints, and other measures used for protection against pipe breaks outside containment. Review of the specific aspects of these elements recommended in B.2.b of BTP APCSB 3-1 is done by the SEB and MEB as follows:

(1) SEB reviews the design of protective structures in connection with the review of other Category I structures under Standard Review Plan (SRP) 3.8.4.

(2) MEB reviews the design of piping restraints and other protective measures in connection with the review of break locations and dynamic effects of piping failures under SRP 3.6.2.

c. APCSB identifies portions of high and moderate energy fluid system piping between containment isolation valves that are subject to the recommendations of B.2.c of BTP APCSB 3-1. MEB reviews the design of these portions of piping in connection with the review of break locations and dynamic effects of piping failures under SRP 3.6.2.

d. MTEB reviews inservice inspection aspects of piping within protective structures or guard pipes, between containment isolation valves, or subject to other protective measures, with regard to the recommendations of B.2.d of BTP APCSB 3-1. This review is done in connection with the review of inservice inspection of Class 2 and 3 components under SRP 6.6.

3. APCSB reviews analyses of postulated piping failures with respect to the guidelines of Section B.3 of BTP APCSB 3-1. The locations and types of failures to be considered and the dynamic effects associated with the failures are reviewed by the MEB under SRP 3.6.2.

a. APCSB reviews analyses of piping failures in high and moderate energy fluid systems postulated according to the guidelines of B.3.a of BTP APCSB 3-1.

b. APCSB reviews the assumptions made in the analyses with regard to:

(1) The availability of offsite power.

(2) The failure of a single active component in systems used to mitigate the consequences of the piping failure.

(3) The special provisions applicable to certain dual purpose systems.

(4) The use of available systems to mitigate the consequences of the piping failure.

c. APCSB reviews the effects of postulated failures on the habitability of the control room and access to areas important to safe control of post-accident operations.

d. APCSB reviews the effects of piping failures in systems not designed to seismic Category I standards on essential systems and components.

4. Other secondary review evaluations are performed as required. These include:

   a. EICSB verifies, on request, the capability of power supplies, instrumentation, and controls to initiate, actuate, and complete needed safety actions, considering the effects of a nearby piping failure such as the release of steam, water, or gases.

   b. CSB verifies, on request, the magnitudes of any differential pressures in structures in which piping failures may be postulated.

II. ACCEPTANCE CRITERIA

Acceptance of the plant design for protection against postulated piping breaks outside containment, as described in the applicant's safety analysis report (SAR), will be based on conformance to Branch Technical Position APCSB 3-1, attached to this plan.

III. REVIEW PROCEDURES

All the systems of concern in this section have been reviewed under other standard review plans with respect to design functions for normal operation and for the prevention or mitigation of accidents. The review under this plan does not deal with individual system design requirements necessary to assure that each system performs as intended, but rather considers the protection necessary to assure the operation of such systems in the event of nearby piping failures. The reviewer will select and emphasize material in the review, as may be appropriate for a particular case.

1. APCSB reviews the information presented in the SAR identifying all high and moderate energy fluid systems, and verifies by comparison with individual system temperatures and pressures that they have been correctly identified. The reviewer will then, by reviewing system descriptions of the high and moderate energy piping runs, and by reviewing the appropriate system arrangement and piping drawings, examine the plant arrangement measures that were taken to assure protection from the effects of postulated pipe breaks of high energy systems, or of leakage cracks for moderate energy systems. The reviewer will determine from SAR that the following means either by themselves or in combination have been used by the applicant to achieve this protection:

   a. High and moderate energy fluid systems are separated from essential systems and components, as defined in Appendix A to BTP APCSB 3-1. The reviewer inspects plant arrangement drawings and other information to verify that this is the case.

b. High and moderate energy fluid systems, or portions thereof, are enclosed within structures or compartments designed to protect nearby essential systems or components. Or, the essential systems and components are enclosed in protective structures. The reviewer traces the routing of the systems identified in the SAR as high or moderate energy systems on appropriate plant arrangement drawings, locates the postulated break locations specified in the applicant's analyses, and determines all locations where the effects from the breaks or leaks interface with safety-related equipment. The reviewer then determines that at these locations, enclosures have been provided that protect the safety-related equipment. Where questions as to break locations arise, the reviewer consults the MEB for a determination on the proper locations.

c. For cases where neither physical separation nor protective enclosures are considered practical by the applicant, the APCSB will review the SAR information to verify the following:

(1) The reasons for which the applicant judged both physical separation and system enclosure to be impractical as means of protection are consistent with Subsection B.1.c of BTP APCSB 3-1.

(2) Redundant design features or additional protection have been provided in these situations and are such that failure modes and effects analyses for all failure situations show that the performance of safety features will be assured, assuming a single active failure in any required system. These analyses are done under the criteria and assumptions of Section B.3 of BTP APCSB 3-1. Special measures taken to provide additional protection are reviewed on an individual case basis, with assistance from the secondary review branches as needed.

2. APCSB reviews the information presented in the SAR that identifies the principal design features. The reviewer performs his evaluation by comparing the design basis information given in the SAR with that described in Section B.2 of BTP APCSB 3-1. By this comparison of individual design features, the reviewer verifies as follows that the necessary measures have been provided by the applicant in his design.

a. APCSB reviews the seismic design classification of plant systems and checks with RSB to verify that essential systems and components, as defined in Appendix A of BTP APCSB 3-1, have been designed to meet the seismic requirements of Section B.2.a of BTP APCSB 3-1.

b. APCSB, with assistance from SEB and MEB, reviews the design features provided for protective structures or compartments, fluid system piping restraints, and other protective measures as described in Section B.2.b of BTP APCSB 3-1. The reviewer compares the design features and bases given in the SAR with the stated section in BTP APCSB 3-1. The comparative review may include the use of plant arrangement and layout drawings as necessary to clarify the design intentions and implementation. In the majority of case reviews, SAR statements and drawings indicating that the design meets the intent of the acceptance criteria are accepted. However, there may be cases where engineering judgment and independent staff analyses are needed to verify the capability of structures and components to withstand the dynamic pressure and mechanical effects of a pipe rupture.

c. APCSB reviews the SAR information, as supplemented by engineering sketches or drawings where necessary, to determine that fluid system piping between containment isolation valves conforms to Section B.2.c of BTP APCSB 3-1. This includes piping penetrations between single and dual barrier containments that may have enclosing protective structures. The review is mainly performed on a comparative basis by APCSB. MEB reviews these piping details to verify the design limits, break locations, and dynamic effects, in accordance with BTP MEB 3-1.

d. APCSB reviews the broad aspects of the applicant's inservice inspection program by comparison of the items included in the program with the provisions given in Section B.2.d of BTP APCSB 3-1. The review of the actual inservice inspection program is performed by MTEB.

3. APCSB reviews the results of the applicant's evaluation of the consequences of postulated piping failures of high and moderate energy fluid piping systems. The type and location of each postulated piping failure (i.e., longitudinal or circumferential) in either a high or moderate energy system will be reviewed by MEB on the basis of BTP MEB 3-1. The review by APCSB will be based upon the information provided by applicants in the SAR concerning the effects of postulated failures on essential equipment and the ability of the plant to be safely shut down, as described in Section B.3 of BTP APCSB 3-1.

The reviewer verifies that the applicant's evaluation has properly considered the following points, and in certain cases, as necessary, performs an independent evaluation especially with regard to single failure analyses.

a. APCSB reviews the applicant's plant arrangements and design features using layout drawings to assure that all potentially affected essential systems and components have been considered with respect to the effects of an assumed pipe break.

b. APCSB reviews the effects of postulated piping failures as determined from the information given in the SAR. The reviewer will confirm the results of the applicant's evaluations by performing a comparative, but abbreviated as appropriate, failure modes and effects analysis that includes the considerations given in Section B.3.d of BTP APCSB 3-1 for the following effects:

(1) The availability of offsite power.

(2) The effects of a single active component failure in systems necessary to mitigate consequences of the postulated piping break.

(3) Permissible exclusions to (2) above based upon the provision given in Section B.3.b(3) of BTP APCSB 3-1 for certain dual purpose moderate energy systems.

(4) The considerations involved in to the selection of available systems to mitigate the consequences of the piping failure.

c. The reviewer will verify from a review of arrangement drawings that control room habitability or access to necessary surrounding areas is not jeopardized as a consequence of the postulated piping failure.

d. APCSB evaluates the applicant's analysis of the postulated failure of non-seismic Category I piping systems by performing a failure modes and effects analysis using SAR information and engineering sketches as necessary.

4. Systems defined in Appendix A to BTP APCSB 3-1 as "essential systems" are those that are needed to shut down the reactor and mitigate the consequences of the pipe break for a given postulated piping break. However, depending upon the type and location of the postulated pipe break, certain safety equipment may not be classed as "essential" for that particular event (e.g., emergency power system or high and low pressure core spray systems). On the other hand, some safety equipment will be "essential" for almost all cases (e.g., service water to ultimate heat sink). Table 3.6.1-1 is a list of those essential systems generally in the latter category.

TABLE 3.6.1-1

SYSTEMS USUALLY REQUIRED FOR SAFE SHUTDOWN

PWR	BWR
Service Water System	Service Water System
Auxiliary Feedwater System	Reactor Coolant Injection System
Volume Control System	
Decay Heat Removal System	Residual Heat Removal System
Component Cooling Water System (if provided)	Component Cooling Water System (if provided)

Table 3.6.1-2 is a listing of systems typically classified as either high or moderate energy systems that are located outside the primary containment in pressurized water reactor (PWR) and boiling water reactor (BWR) plants.

TABLE 3.6.1-2

TYPICAL HIGH ENERGY SYSTEMS OUTSIDE CONTAINMENT

PWR	BWR
Main Steam Line System	Main Steam Line System
Main Feedwater Line System	Main Feedwater Line System
Auxiliary Feedwater System	High Pressure Core Spray System
Volume Control System	Process Sampling System
Process Sampling System	Condensate System
Condensate System	Reactor Cleanup System
Steam Generator Blowdown Line	Standby Liquid Control System

TYPICAL MODERATE ENERGY SYSTEMS OUTSIDE CONTAINMENT

PWR	BWR
Service Water System	Service Water System
Decay Heat Removal System (outside of reactor coolant pressure boundary)	Residual Heat Removal System (outside of reactor coolant pressure boundary)
Circulating Water System	Circulating Water System
Fire Protection System	Fire Protection System
Component Cooling Water System	Component Cooling Water System

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The review of the plant design for protection against postulated piping failures outside containment included all high and moderate energy piping systems located outside containment. The review of these high and moderate energy systems for the __________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information. [The review has included the applicant's proposed design criteria and design bases for the systems, structures, and components of interest, the adequacy of those criteria and bases, and the functions necessary to maintain the capability for a safe plant shutdown during any failure of high or moderate energy system piping. (CP)] [The review has included the applicant's analysis of the manner in which the design of all structures, systems, and components conforms to the design criteria and design bases and demonstrates the ability to perform a safe plant shutdown after any postulated piping failure of a high or moderate energy system. (OL)]

"The staff concludes that the facility design for protection against postulated piping failures outside containment conforms to the Commissions's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. References

1. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to this plan, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

A. BACKGROUND

General Design Criterion 4, "Environmental and Missile Design Bases," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that systems and components important to safety "...shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit." Guidance on acceptable design approaches to meet General Design Criterion 4 for existing plants and for plants for which applications for construction permits were then under review was provided in letters to applicants and licensees from A. Giambusso, Deputy Director of Licensing for Reactor Projects, most of which were dated in December 1972. The guidance document from these letters is attached as Appendix B to this position. Similar interim guidance for new plants was provided in a letter to applicants, prospective applicants, reactor vendors, and architect-engineers from J. F. O'Leary, Director of Licensing, dated July 12, 1973. This document is attached as Appendix C to this position.

Guidance is available for protection against pipe whipping and other effects of postulated fluid system piping failures (e.g., a break or rupture resulting in a loss-of-coolant accident) of systems and components important to safety located within primary reactor containment. As an example, this problem is addressed by Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

Reviews of nuclear power plant designs have indicated that the functional or structural integrity of systems and components required for safe shutdown of the reactor and maintenance of cold shutdown conditions could be endangered by fluid system piping failures at locations outside containment. The staff has evolved an acceptable approach for the design, including the arrangement, of fluid systems located outside of containment to assure that the plant can be safely shut down in the event of piping failures outside containment. This approach is set forth in this position and in the companion Branch Technical Position (BTP) attached to Standard Review Plan 3.6.2, BTP MEB 3-1.

It is the intent of this design approach that postulated piping failures in fluid systems should not cause a loss of function of essential safety-related systems and that nuclear plants should be able to withstand postulated failures of any fluid system piping outside containment, taking into account the direct results of such failure and the further failure of any single active component, with acceptable offsite consequences.

The detailed provisions of the position below and of BTP MEB 3-1 are intended to implement this intent with due consideration of the special nature of certain dual purpose systems and the need to define and to limit to a finite number the types and locations of piping failures to be analyzed. Although various measures for the protection of safety-related systems and components are outlined in this position, the preferred method of protection is based upon separation and isolation by plant arrangement.

B. BRANCH TECHNICAL POSITION

1. Plant Arrangement

Protection of essential systems and components[1/] against postulated piping failures in high or moderate energy fluid systems that operate during normal plant conditions and that are located outside of containment should be provided by one of the following plant arrangement considerations:

a. Plant arrangements should separate fluid system piping from essential systems and components. Separation should be achieved by plant physical layouts that provide sufficient distances between essential systems and components and fluid system piping such that the effects of any postulated piping failure therein (e.g., pipe whip, jet impingement, and the environmental conditions resulting from the escape of contained fluids as appropriate to high or moderate-energy fluid system piping) cannot impair the integrity or operability of essential systems and components.

b. Fluid system piping or portions thereof not satisfying the provisions of B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems.

c. Plant arrangements or system features that do not satisfy the provisions of either B.1.a or B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by the staff as a base plant under the Commission's standardization and replication policy; or for other substantive reasons such as particular design features of the fluid systems. Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut down the reactor). In these cases, redundant design features that are separated or otherwise protected from postulated piping failures, or additional protection, should be provided so that the effects of postulated piping failures are shown by the analyses and guidelines of B.3 to be acceptable. Additional protection may be provided by restraints and barriers or by designing or testing essential systems and components to withstand the effects associated with postulated piping failures.

2. Design Features

a. Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide 1.29.

[1/] See Appendix A for definitions of underlined phrases.

b. Protective structures or compartments, fluid system piping restraints, and other protective measures should be designed in accordance with the following:

(1) Protective structures or compartments needed to implement B.1 should be designed to seismic Category I requirements. The protective structures should be designed to withstand the effects of a postulated piping failure (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with loadings associated with the operating basis earthquake and safe shutdown earthquake within the respective design load limits for structures. Piping restraints, if used, may be taken into account to limit effects of the postulated piping failure.

(2) High-energy fluid system piping restraints and protective measures should be designed such that a postulated break in one pipe cannot, in turn, lead to rupture of other nearby pipes or components if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break. An unrestrained whipping pipe should be considered capable of (a) rupturing impacted pipes of smaller nominal pipe sizes and (b) developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

c. Fluid system piping in containment penetration areas should meet the following design provisions:

(1) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of single barrier containment structures (including any rigid connection to the containment penetration) that connect, on a continuous or intermittent basis, to the reactor coolant pressure boundary, or the steam and feedwater systems of PWR plants, should be designed to the stress limits specified in B.1.b or B.2.b of Branch Technical Position (BTP) MEB 3-1, attached to Standard Review Plan 3.6.2.

These portions of high-energy fluid system piping should be provided with pipe whip restraints that are capable of resisting bending and torsional moments produced by a postulated piping failure either upstream or downstream of the containment isolation valves. The restraints should be located reasonably close to the containment isolation valves and should be designed to withstand the loadings resulting from a postulated piping failure beyond these portions of piping so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.

(2) Portions of fluid system piping between the required restraints located inside and outside containment beyond the isolation valves of dual barrier

containment structures should also meet the design provisions of B.2.c.(1). In addition, those portions of piping that pass through the containment annulus, and whose postulated failure could affect the leaktight integrity of the containment structure or result in pressurization of the containment annulus beyond design limits should be provided with an enclosing protective structure.

For the purpose of establishing the design parameters (i.e., pressure, temperature) of the enclosing protective structure, a full flow area opening should be assumed in that portion of piping within the enclosing structure and vent areas should be taken into account, if provided, in the enclosing structure. Where guard pipes for individual process pipes are used as an enclosing protective structure, such guard pipes should be designed to meet the requirements specified in B.1.b(6) of BTP MEB 3-1.

(3) Terminal ends of the piping runs extending beyond these portions of high-energy fluid system piping should be considered to originate at a point adjacent to the required pipe whip restraints located inside and outside containment.

(4) Piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable.

d. Inservice examination and related design provisions should be in accordance with the following:

(1) The protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, "Rules for Inspection and Testing of Components in Light-Water Cooled Plants."

(2) For those portions of fluid system piping identified in B.2.c, includes piping running from inboard to outboard restraints in containment penetration areas, the extent of inservice examinations completed during each inspection interval (IWA-2400, ASME Code, Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

(3) For those portions of fluid systems piping enclosed in guard pipes, inspection ports should be provided in guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

(4) The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Tables IWC-2520.

3. Analyses and Effects of Postulated Piping Failures

a. To show that the plant arrangement and design features provide the necessary protection of essential systems and components, piping failures should be postulated in accordance with BTP MEB 3-1, attached to Standard Review Plan 3.6.2. In applying the provisions of BTP MEB 3-1, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping should be considered separately as a single postulated initial event occurring during normal plant conditions. An analysis should be made of the effects of each such event, taking into account the provisions of BTP MEB 3-1 and of the system and component operability considerations of B.3.b. below. The effects of each postulated piping failure should be shown to result in offsite consequences within the guidelines of 10 CFR Part 100 and to meet the provisions of B.3.c and d below.

b. In analyzing the effects of postulated piping failures, the following assumptions should be made with regard to the operability of systems and components:

(1) Offsite power should be assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure.

(2) A single active component failure should be assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in B.3.b.(3) below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

(3) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure, single failures of components in the other train or trains of that system only need not be assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that may, in some plant designs, qualify as dual-purpose essential systems are service water systems, component cooling systems, and residual heat removal systems.

(4) All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account should be taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions should be judged on the basis of ample time and adequate access to equipment being available for the proposed actions.

c. The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, should not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure.

d. A postulated failure of piping not designed to seismic Category I standards should not result in any loss of capability of essential systems and components to withstand the further effects of any single active component failure and still perform all functions required to shut down the reactor and mitigate the consequences of the postulated piping failure.

4. Implementation

a. Designs of plants for which construction permit applications are tendered after July 1, 1975 should conform to the provisions of this position.

b. Designs of plants for which construction permit applications are tendered after July 1, 1973 and before July 1, 1975 should conform to the provisions of either (a) the letter of July 12, 1973 from J. F. O'Leary, Appendix C to this position, or (b) this position, at the option of the applicants.

c. Designs of plants for which construction permit applications were tendered before July 1, 1973 and operating licenses are issued after July 1, 1975 should follow the guidance provided in the December 1972 letter from A. Giambusso, Appendix B to this position and provide analyses of moderate energy lines made in conformance with B.3 of this position, as part of the operating license application for these plants to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided. Alternately, this position may be used in its entirety as an acceptable basis for this finding.

For plants in this category for which construction permits are not issued as of February 1, 1975, a commitment by the applicant to either (a) follow the guidance of Appendix B and submit B.3 analyses of moderate energy lines with the plant final safety analysis report (FSAR), or (b) conform the plant design to the provisions of this position, should provide an acceptable basis for issuance of the construction permit with regard to effects of piping failures outside containment.

d. Designs of plants for which operating licenses are issued before July 1, 1975 are considered acceptable with regard to effects of piping failures outside containment on the basis of the analyses made and measures taken by applicants and licensees in response to the December 1972 letter from A. Giambusso, and the staff review and acceptance of these analyses and measures.

For plants in this category for which the staff review and acceptance of protection against the effects of piping failures outside containment is not substantially complete as of February 1, 1975, a commitment by the applicant to carry out analyses according to B.3 of this position, to submit them for staff review, and to carry out any system modifications found necessary before extended operation of the plant at power levels above one-half the license power level, should provide an acceptable basis for issuance of the operating license.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.29, "Seismic Design Classification."

3. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

4. Letter from A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, to applicants and licensees, December 1972, and attachment entitled "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment." The corrected attachment is Appendix B to this position.

5. Letter from J. F. O'Leary, Director of Licensing, to applicants, reactor venders, and architect-engineers, July 12, 1972, and attachment entitled "Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures." The letter and attachment is Appendix C to this position.

6. ASME Boiler and Pressure Vessel Code, Sections III and XI, American Society of Mechanical Engineers.

APPENDIX A

BRANCH TECHNICAL POSITION APCSB 3-1

DEFINITIONS

Essential Systems and Components. Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

Fluid Systems. High and moderate energy fluid systems that are subject to the postulation of piping failures outside containment against which protection of essential systems and components is needed.

High-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

a. maximum operating temperature exceeds 200°F, or

b. maximum operating pressure exceeds 275 psig.

Moderate-Energy Fluid Systems. Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. maximum operating temperature is 200°F or less, and

b. maximum operating pressure is 275 psig or less

Normal Plant Conditions. Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

Upset Plant Conditions. Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

Postulated Piping Failures. Longitudinal and circumferential breaks in high-energy fluid system piping and through-wall leakage cracks in moderate-energy fluid system piping postulated according to the provisions of BTP MEB 3-1, attached to SRP 3.6.2.

S_h and S_A. Allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

S_m. Design stress intensity as defined in Article NB-3600 of the ASME Code, Section III.

Single Active Component Failure. Malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends. Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping thermal expansion. A branch connection to a main piping run is a terminal end of the branch run.

Intersections of runs of comparable size and fixity need not be considered terminal ends when so justified in the analysis. Terminal ends for the purpose of postulating breaks should be selected at points located immediately outside or beyond the required pipe whip restraints located inside and outside containment at penetration areas.

In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.

APPENDIX B

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the attachment to the letters sent by A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, in December 1972 to applicants and licensees on the subject of postulated piping failures outside containment. The attachment provided guidance on measures to be taken and on information to be submitted. An errata sheet for the attachment was sent in January 1973 to recipients of the original letters. The attachment as given here has been corrected for the errata.

General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:
   (a) Both of the following piping system conditions are met:
      (1) the service temperature is less than 200°F; and
      (2) the design pressure is 275 psig or less; or
   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or
   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or
   (d) The internal energy level[1/] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system or component to an unacceptable level.

[1/] Footnotes are collected at the end of this appendix.

2. Design basis break locations should be selected in accordance with the following pipe whip protection criteria: however, where pipes carrying high energy fluids are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2/] breaks should be postulated to occur at the following locations in each piping run[3/] or branch run:
   (1) The terminal ends;
   (2) Any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4/] exceeds 2.0 S_m[5/] for ferritic steel, and 2.4 S_m for austenitic steel;
   (3) Any intermediate locations between terminal ends where the cumulative usage factor (U)[6/] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and
   (4) At intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:
   (1) The terminal ends;
   (2) Any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 $(S_h + S_A)$[7/] or the expansion stresses exceed 0.8 S_A; and
   (3) Intermediate locations in addition to these determined by (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under (2) above should be equivalent to the following:

(a) Longitudinal[8/] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

(b) Circumferential[9/] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:
   (a) The locations and number of design basis breaks on which the dynamic analyses are based.
   (b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.
   (c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.
   (d) Diagrams of mathematical models used for the dynamic analysis.
   (e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structures, systems, or components important to safety, such as the control room.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet, and reactive forces including:
   (a) Pipe restraint design to prevent whip impact;
   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;
   (c) Separation of redundant features;
   (d) Provisions to separate physically piping and other components of redundant features; and
   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:
   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;
   (b) The allowable design stresses and/or strains; and
   (c) The load factors and the load combinations.

7. The structural design loads, including the pressure and temperature transients, the dead, live and equipment loads, and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations, and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including non-seismic Category I structures,

caused by the accident, will not cause failure of any other structure in a manner to adverseley affect:

(a) Mitigation of the consequences of the accidents; and

(b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:

    (a) Loss of required redundancy in any portion of the protection system (as defined in IEEE Std 279), Class IE electric system (as defined in IEEE Std 308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of that accident and place the reactor(s) in a cold shutdown condition; or

    (b) Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted; but a loss of function is not permitted. For such situations, plant shutdown is required.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown on the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy line break. The information required for our review should include the following:

    (a) Identification of all electrical equipment necessary to meet requirements of (11) above. The time after the accident in which they are required to operate should be given.

    (b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

    (c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingement from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

    (d) An evaluation of the capability for safety-related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

    (e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety-related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety-related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

1/ The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

2/ Piping is a pressure-retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

3/ A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

4/ Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

5/ S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

6/ U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

7/ S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

8/ Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

9/ Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

APPENDIX C

BRANCH TECHNICAL POSITION APCSB 3-1

This appendix consists of the letter and attachment sent by J. F. O'Leary, Director of Licensing, to applicants, reactor vendors, and architect-engineers on the subject of postulated piping failures outside containment. The letter was dated July 12, 1973.

Late last year, the Atomic Energy Commission's Regulatory staff requested those utilities that operate nuclear power plants, have applied for operating licenses, or have plants whose construction permit review was essentially complete, to assess the effects and consequences of a postulated rupture of piping containing high-energy fluids and located outside of the containment structure. These requests were issued by Mr. A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing, in letters, most of which were dated in December 1972.

Because these plants were either in operation or in advanced stages of engineering design and construction, the request included guidance for corrective modifications that could be implemented by in-situ measures. Such modifications included relocation or rerouting of piping, installation of impingement barriers and encapsulation sleeves around high stressed piping regions, provisions for venting of compartments subject to pressurization, addition of piping restraints, and strengthening of structural components of buildings.

From our review of responses submitted to the Regulatory staff, and from discussions with architect-engineering firms, we have learned that some of these organizations have inferred that the criteria contained in Mr. A. Giambusso's letter pertaining to corrective modifications for plants in advanced stages of construction and operation are applicable for the design of high-energy fluid systems outside the containment in new designs of nuclear power plants. It was not our intent that the criteria for corrective plant modifications be applied to new power plants that are in the initial design stages. We believe that a more direct approach, involving a re-arrangement of the physical plant layout with a view to relocation of essential safety systems and components is appropriate for the new plants.

For the present, pending issuance of a planned AEC Regulatory Guide - "Protection Against Postulated Events and Accidents Outside Containment," an acceptable implementation of Criterion 4 of the Commission's General Design Criteria listed in Appendix A of 10 CFR Part 50, as applied to new plants with respect to the design of structures, systems and components important to safety and located outside of containment is as follows:

I. PIPING SYSTEMS CONTAINING HIGH-ENERGY FLUIDS* DURING NORMAL REACTOR OPERATION

(a) The piping systems are isolated by adequate physical separation and remotely located from safety systems and components that are required to shut down the reactor safely and maintain the plant in a cold shutdown condition.

*Refer to Appendix A for identification of high-energy fluid systems.

(b) Where isolation by remote location is impracticable, systems containing high-energy fluids, or portions of the systems, are enclosed within the structures suitably designed to protect adjoining safety systems and components required to shut down the reactor safely and maintain the plant in a cold shutdown condition from postulated pipe failures within the enclosure.

(c) Where both isolation by remote location (as specified in I.a) and enclosure in protective structures (as specified in I.b) are impracticable, systems containing high-energy fluids, or portions of the systems, are provided with restraints and protective measures such that the operability and integrity of structures, safety systems and components that are required to shut down condition are not impaired.

(d) Protective enclosures for the piping systems containing high-energy fluids are designed as Seismic Category I structures to withstand the combined effects of a postulated pipe break, the dynamic effects of pipe whipping, the jet impingement forces, and the compartment pressurization as a consequence of discharging fluids in combination with the specified seismic event of the Safe Shutdown Earthquake and normal operating loads.

(e) Piping systems containing high-energy fluids are designed so that the effects of a single postulated pipe break cannot, in turn, cause failures of other pipes or components with unacceptable consequences.

In addition, any systems, or portions of systems, that are designed to mitigate the consequences of a postulated pipe failure, and to place the reactor in the cold shutdown condition, are provided with design features that will assure the performance of their safety function, assuming a single active component failure.

(f) For a postulated pipe failure, the escape of steam, water, and heat from structures enclosing the high-energy fluid containing piping does not preclude: 1) the accessibility to surrounding areas important to the safe control of reactor operations, 2) the habitability of the control room, 3) the ability of instrumentation, electric power supplies, and components and controls to initiate, actuate and complete a safety action. In this regard, a loss of redundancy is permissible but not the loss of function.

(g) The criteria for determination of postulated break locations are contained in the attached Appendix A, "Criteria for Determination of Postulated Pipe Break or Leakage Locations in Fluid Piping Systems Outside Containments."

II. PIPING SYSTEMS CONTAINING MODERATE-ENERGY FLUIDS* DURING REACTOR OPERATION

(a) Piping systems containing moderate-energy fluids are designed to comply with the criteria applied to high-energy fluid piping systems as listed under I., above, except that the piping is postulated to develop a limited-size through-wall leakage crack instead of a pipe break.

(b) For each postulated leakage, design measures are included that provide protection from the effects of the resulting water spray and flooding to the same extent required to satisfy criterion I(e).

(c) The criteria for determination of postulated leakage locations are contained in Appendix A.

The measures taken for the protection of structures, systems and components important to safety should not preclude the conduct of inservice examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME Boiler and Pressure Vessel Code - Section XI, "Inservice Inspection of Nuclear Power Plant Components."

*Refer to Appendix A for identification of moderate-energy fluid systems.

Although compliance with the design criteria listed above should be accomplished by plant arrangement and layouts utilizing the separation concept to the extent practicable, special consideration will be necessary to provide adequate protection where interconnection is unavoidable between high-energy fluid containing piping and piping of systems important to safety.

We are prepared to discuss with you these guidelines for the design of new nuclear power plants with regard to protection required against postulated breaks of high and moderate energy piping outside of containment, particularly for those plants with construction permit applications currently under consideration.

Sincerely,

John F. O'Leary, Director
Directorate of Licensing

Enclosure:
Appendix A

APPENDIX A TO J.F. O'LEARY LETTER OF JULY 12, 1972

CRITERIA FOR DETERMINATION OF POSTULATED BREAK AND LEAKAGE LOCATIONS IN HIGH[1/] AND MODERATE[2/] ENERGY FLUID PIPING SYSTEMS OUTSIDE OF CONTAINMENT STRUCTURES[10/]

A. High-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated by remote location for structures, systems, and components important to safety[3/], pipe breaks[4/] need not be postulated provided the requirements of A.4 are satisfied.

2. For piping systems that are enclosed in suitably designed concrete structures or compartments to protect structures, systems, and components important to safety, pipe break should be postulated at the following locations in each piping or branch run within the protective structure:

   a. the terminal ends[9/] of the piping or branch run (except as exempted by the provisions of A.4), if located within the protective structure or compartment, and
   b. each fitting (i.e., elbow, tee, cross, non-standard fitting), and
   c. a minimum of one break selected in each piping or branch run within the protective structure or compartment at a location that results in the maximum loading from the impact of the postulated ruptured pipe and jet discharge force on wall, floor, and roof of the structure or compartment, including internal pressurization, and taking into account any piping restraints provided to limit pipe motions.

3. For portions of piping systems that can neither be isolated as specified A.1, nor enclosed in protective structures as specified in A.2, pipe breaks should be postulated at the following locations in each piping or branch run within the confines of the structures or compartments that enclose or adjoin areas containing systems and components important to safety:

   a. the terminal ends[9/] of piping or branch run (except as exempted by A.4), if located within the boundary of the confining structure or each compartment within the structure; and
   b. any intermediate location within the boundary of the confining structure or each compartment within the structure where the stresses[5/] under the loadings associated with specified seismic events[6/] and operational plant conditions[7/] exceed 0.8 $(S_h + S_A)$[8/] or, in lieu of these calculated stress-related locations, at each fitting (i.e., elbow, tree, cross, non-standard fitting); and
   c. a minimum of two separated locations within the boundary of the confining structure or each compartment within the structure in piping or branch runs exceeding twenty pipe diameters in length; a minimum of one location in piping or branch runs twenty pipe-diameters or less in length except that no intermediate locations need to be postulated in branch runs that are three pipe-diameters or less in length. Intermediate break locations should be selected such that the maximum pipe whip and jet impingement will result, assuming for this purpose an unrestrained ruptured pipe.

4. For those portions of the piping passing through primary containment penetrations and extending to the first outside isolation valve, pipe breaks need not be postulated provided such piping is conservatively reinforced and restrained beyond the valve such that, in the event of a postulated pipe break outside containment, the transmitted pipe loads will neither impair the operability of the valve nor the integrity of the piping or the containment penetration. (A terminal end of such piping is considered to originate at this restraint location.)

B. Moderate-Energy Fluid Systems

1. For piping systems that by plant arrangement and layout are isolated and physically separated and remotely located from systems and components important to safety, through-wall leakage cracks need not be postulated.

2. For piping systems that are located in the same areas as high-energy fluid systems which, by the criteria of A.1 to A.3 have postulated pipe break locations, through-wall leakage cracks need not be postulated.

3. For piping systems that are located in areas containing systems and components important to safety, but where no high-energy fluid systems are present, through-wall leakage cracks should be postulated at the most adverse location to determine the protection needed to withstand the effects of the resulting water spray and flooding.

C. Size and Types of Pipe Breaks and Cracks

1. The following types of breaks should be postulated at the locations specified by the criteria listed under A. High-Energy Fluid Systems:

   a. longitudinal breaks in piping runs and branch runs with nominal pipe sizes of 4 inches and larger,

   b. circumferential breaks in piping runs and branch runs exceeding a nominal pipe size of 1 inch.

2. The following leakage cracks are postulated at the locations specified by the criteria listed under B. Moderate-Energy Fluid Systems:

   a. through-wall leakage cracks in piping and branch runs exceeding a nominal pipe size of 1 inch, where the crack opening is assumed as 1/2 the pipe diameter in length and 1/2 the pipe wall thickness in width.

FOOTNOTES

1/ High-energy systems include those systems where either of the following conditions are met:

a) the maximum operating temperature exceeds $200^{\circ}F$, and
b) the maximum operating pressure exceeds 275 psig.

2/ Moderate energy systems include those systems where both of the following conditions are met:

a) the maximum operating temperature is $200^{\circ}F$ or less, and
b) the maximum operating pressure is 275 psig or less.

3/ Structures, systems, and components important to safety, as specified herein refer to those plant features required to shut down the reactor safely and maintain the plant in the cold shutdown condition.

4/ Break in piping means (a) a complete circumferential pipe severance and, (b) a longitudinal split opening an area equal to the pipe area, but without pipe severance. Such breaks are assumed to occur at each specified break location, but not concurrently.

5/ Either circumferential or longitudinal stresses derived on an elastically-calculated basis.

6/ Specified seismic events are earthquakes that produce at least 50 percent of the vibratory motion of the Safe Shutdown Earthquake (SSE).

7/ Operational plant conditions include normal reactor operation, upset conditions, (e.g., anticipated operational occurrences) and testing conditions.

8/ S_h is the allowable stress at maximum temperature, and S_A is the allowable stress range for expansion stresses for Class 2 and 3 piping as permitted by the rules of ASME Code Section III.

9/ Terminal ends of pipe runs originate at points of maximum constraint (e.g., Connections to vessels, pumps, valves, fittings that are rigidly anchored to structures) terminal ends of branch runs originate at pipe intersections and components that act as rigid constraints.

10/These criteria are intended for the purpose of designing piping restraints and do not preclude consideration of other aspects of the AEC General Design Criteria, such as single failure criteria and other additional protective measures required to provide protection against environmental conditions incident to postulated accidents.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.6.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

Information concerning break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high and moderate energy fluid system piping, including "field run" piping, inside and outside of containment should be provided in the applicant's safety analysis report (SAR). This information is reviewed by the MEB in accordance with this plan, to confirm that requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break are met. At the construction permit (CP) stage, the staff review covers the following specific areas:

1. The criteria used to define break and crack locations and configurations.

2. The analytical methods used to define the forcing functions, including the jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related structures, systems, and components.

3. The dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices, under postulated pipe rupture loads.

At the operating license (OL) stage, the staff review covers the following specific areas:

1. The implementation of criteria for defining pipe break and crack locations and configurations.

2. The implementation of criteria dealing with special features, such as augmented inservice inspection programs or the use of special protective devices such as pipe whip restraints, including diagrams showing final configurations, locations, and orientations in relation to break locations in each piping system.

3. The acceptability of the analysis results, including the jet thrust and impingement forcing functions and pipe whip dynamic effects.

4. The design adequacy of systems, components, and component supports to assure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loadings.

Secondary reviews related to the areas of this plan are performed by other branches and the results are used by MEB to complete its evaluation. The secondary reviews are as follows:

1. The APCSB reviews plant arrangements where separation of high and moderate energy systems is the method of protection for essential systems and components, Sections B.1.a and B.2.a of BTP MEB 3-1. The APCSB identifies high and moderate energy systems outside containment and the essential systems and components that must be protected from postulated piping failures in these high and moderate energy systems.

2. The SEB reviews loading combinations and other design aspects of protective structures or compartments used to protect essential systems and components.

3. The MTEB reviews inservice inspection and related design provisions of high and moderate energy systems.

4. The RSB identifies high and moderate energy systems inside containment and the essential systems and components that must be protected from postulated piping failures in these high and moderate energy systems.

5. The EICSB reviews the environmental effects of pipe rupture, such as temperature, humidity, and spray-wetting with respect to the functional performance of essential electrical equipment and instrumentation.

II. <u>ACCEPTANCE CRITERIA</u>

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. <u>Postulated Pipe Break Locations Inside Containment</u>

   Acceptable criteria to define postulated pipe break locations and configurations inside containment are specified in Regulatory Guide 1.46 (Ref. 2). If the criteria specified in Regulatory Guide 1.46 are inpractical to implement for a specific application, the criteria of Branch Technical Position (BTP) MEB 3-1 (Ref. 5) will be considered.

2. Postulated Pipe Break Locations Outside Containment

For protection against postulated pipe ruptures outside containment, References 3 and 4, and BTB MEB 3-1 provide acceptable criteria to define postulated rupture locations and plant layout considerations.

Reference 3 includes the area of concern in this plan and may be used for those plants for which construction permit applications were tendered before July 1, 1973, as specified in Section B.4 of BTP APCSB 3-1 (Ref. 6).

Reference 4 specifically emphasizes protection via plant arrangement and layouts utilizing the concept of physical separation to the extent practical, and may be used for those plants for which construction permit applications are tendered after July 1, 1973 and before July 1, 1975, as specified in Section B.4 of BTP APCSB 3-1.

BTP MEB 3-1 may be used for all applications, in lieu of References 3 and 4, at the option of applicants. After July 1, 1975, only BTP MEB 3-1 will be used by the staff in the review of all new construction permit applications.

3. Methods of Analysis

Detailed acceptance criteria covering pipe whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement, are included in Section III, "Review Procedures," of this plan. The general bases and assumptions of the analysis are given in BTP MEB 3-1, Section B.3.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. The locations and configurations of breaks in high energy piping and leakage cracks in moderate energy piping are reviewed.

   a. At the CP stage, the applicant's criteria for determining break and crack locations are reviewed for conformance with the acceptance criteria referenced in Section II of this plan.

      Exceptions taken by the applicant to the referenced pipe break location and configuration criteria must be identified and the basis clearly justified so that evaluation is possible. Deviations from approved criteria and the justifications provided are reviewed to determine acceptability.

   b. At the OL stage, the following are reviewed to ensure that the pipe break criteria have been properly implemented:

      (1) Sketches showing the locations of the resulting postulated pipe ruptures, including indentification of longitudinal and circumferential breaks, structural barriers, if any, restraint locations, and the constrained directions in each restraint. APCSB reviews this information for fluid systems outside

containment and RSB for systems inside containment, with regard to identification of the systems in which failures should be postulated.

(2) A summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range as delineated in References 3 and 4 and Branch Technical Position MEB 3-1.

2. Analyses of pipe motion caused by the dynamic effects of postulated failures are reviewed. These analyses should show that pipe motions will not be such as to result in unacceptable impact upon, or overstress of, any structure, system, or component important to safety to the extent that essential functions would be impaired or precluded. The analysis methods used should be adequate to determine the resulting loadings in terms of the kinetic energy or momentum induced by the impact of the whipping pipe, if unrestrained, upon a protective barrier or a component important to safety and to determine the dynamic response of the restraints induced by the impact and rebound, if any, of the ruptured pipe.

At the CP stage, the staff reviews the applicant's criteria, methods, and procedures used or proposed for dynamic analyses by comparing them to the criteria which follow. At the OL stage, the analyses are reviewed in accordance with these criteria.

a. <u>Dynamic Analysis Criteria</u>
An analysis of the dynamic response or static equivalent thereof of the pipe run or branch should be performed for each longitudinal and circumferential postulated piping break.

The loading condition of a pipe run or branch, prior to the postulated rupture, in terms of internal pressure, temperature, and inertial effects should be consistent with the limiting upset plant operating condition.

In the case of a circumferential rupture, the need for a pipe whip dynamic analysis may be governed by considerations of the available driving energy as discussed in position 4.c of Reference 2.

Dynamic analysis methods used for calculating piping and restraint system responses to the jet thrust developed following the postulated rupture should adequately account for the following effects: (a) mass inertia and stiffness properties of the system, (b) impact and rebound, (c) elastic and inelastic deformation of piping and restraints, and (d) support boundary conditions.

The design strain limit for restraints should not exceed 0.5 of the ultimate uniform strain of the materials of the restraints. The method of dynamic analysis used should be capable of determining the inelastic behavior of the piping and restraint system within these design limits.

A 10% increase of minimum specified design yield strength (S_y) may be used in the analysis to account for strain rate effects.

Dynamic analysis methods and procedures presented should include:

(1) A representative mathematical model of the piping system or piping and restraint system.

(2) The analytical method of solution selected.

(3) Solutions for the most severe responses among the piping breaks analyzed.

(4) Solutions with demonstrable accuracy or justifiable conservatism.

The extent of mathematical modeling and analysis should be governed by the method of analysis selected.

b. Dynamic Analysis Models for Piping Systems

Acceptable models for the analysis of ASME Class 1, 2, and 3 piping systems and other piping systems which must be designed to seismic Category I standards include the following:

(1) Lumped Parameter Analysis Model: Lumped mass points are interconnected by springs to take into account inertia and stiffness properties of the system, and time histories of responses are computed by numerical integration, taking into account clearances at restraints and inelastic effects. In the calculation, the maximum possible initial clearance should be used to account for the most adverse dynamic effects of pipe whip.

(2) Energy Balance Analysis Model: Kinetic energy generated during the first quarter cycle movement of the ruptured pipe and imparted to the piping and restraint system through impact is converted into equivalent strain energy. In the calculation, the maximum possible initial clearance at restraints should be used to account for the most adverse dynamic effects of pipe whip. Deformations of the pipe and the restraint should be compatible with the level of absorbed energy. The energy absorbed by the pipe deformation may be deducted from the total energy imparted to the system. For applications where pipe rebound may occur upon impact on the restraint, an amplification factor of 1.2 should be used to establish the magnitude of the forcing function in order to determine the maximum reaction force of the restraint beyond the first quarter cycle of response. Amplification factors other than 1.2 may be used if justified by more detailed dynamic analysis.

(3) Static Analysis Model: The jet thrust force is represented by a conservatively amplified static loading, and the ruptured system is analyzed statically. An amplification factor can be used to establish the magnitude of the forcing

function. However, the factor should be based on a conservative value obtained by comparison with factors derived from detailed dynamic analyses performed on comparable systems.

(4) Other models may be considered if justified.

c. Dynamic Analysis Models for Jet Thrust Forces

(1) The time-dependent function representing the thrust force caused by jet flow from a posulated pipe break or crack should include the combined effects of the following: the thrust pulse resulting from the sudden pressure drop at the initial moment of pipe rupture; the thrust transient resulting from wave propagation and reflection; and the blowdown thrust resulting from build-up of the discharge flow rate, which may reach steady state if there is a fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval. Alternately, a steady state jet thrust function may be used, as outlined in (4), below.

(2) A rise time not exceeding one millisecond should be used for the initial pulse, unless longer crack propagation times or rupture opening times can be substantiated by experimental data or analytical theory.

(3) The time variation of the jet thrust forcing function should be related to the pressure, enthalpy, and volume of fluid in the upstream reservoir, and the capability of the reservoir to supply a high energy flow stream to the break area for a significant interval. The shape of the transient function may be modified by considering the break area and the system flow conditions, the piping friction losses, the flow directional changes, and the application of flow limiting devices.

(4) The jet thrust force may be represented by a steady state function if the energy balance model or the static model is used in the subsequent pipe motion analysis. In either case, a step function amplified as indicated in 2.b(2) or 2.b(3), above, is acceptable. The function should have a magnitude not less than

$$T = KpA$$

where

p = system pressure prior to pipe break

A = pipe break area, and

K = thrust coefficient.

To be acceptable, K values should not be less than 1.26 for steam-saturated water, or steam-water mixtures, or 2.0 for subcooled, nonflashing water.

3. Analyses of jet impingement forces are reviewed. These analyses should show that jet impingement loadings on nearby safety-related structures, systems, and components will not be such as to impair or preclude essential functions. Assumptions that are acceptable in modeling jet impingement forces are:

a. The jet area expands uniformly at a half angle not exceeding 10 degrees.

b. The impinging jet proceeds along a straight path.

c. The total impingement force acting on any cross-sectional area of the jet is time and distance invariant, with a total magnitude equivalent to the jet thrust force as defined in 2.c(4), above.

d. The impingement force is uniformly distributed across the cross-sectional area of the jet, and only the portion intercepted by the target is considered.

e. The break opening may be assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

f. Jet expansion within a zone of five pipe diameters from the break location is acceptable if substantiated by a valid analysis or testing, i.e., Moody's expansion model (Ref. 7). However, jet expansion is applicable to steam or water-steam mixtures only, and should not be applied to cases of subcooled water blowdown.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The proposed design of piping restraints and measures to deal with jet impingement effects upon the reactor coolant pressure boundary and other safety-related systems provide adequate protection for the containment structure, reactor coolant pressure boundary elements, and other systems important to safety.

"The provisions for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid provide adequate assurance that design basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and emergency core cooling system performance will not be degraded by these dynamic effects.

"The proposed piping arrangement and applicable design considerations for high and moderate energy fluid systems inside and outside of containment, other than the reactor coolant pressure boundary, will provide adequate assurance that the unaffected system components, and those systems important to safety which are in close proximity to the systems in which postulated pipe failures are assumed to occur, will be protected. The design will be of a nature to mitigate the consequences of a pipe break so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated failure of a pipe carrying a high or moderate energy fluid inside or outside of containment."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

3. Attachment to letter from A. Giambusso, December 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," Appendix B to BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

4. Letter from J. F. O'Leary, July 12, 1973, and attachment entitled,"Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures," Appendix C to BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

5. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to this plan.

6. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

7. F. J. Moody, "Prediction of Blowdown and Jet Thrust Forces," ASME Paper 69 HT-31, August 6, 1969.

BRANCH TECHNICAL POSITION MEB 3-1

POSTULATED BREAK AND LEAKAGE LOCATIONS IN FLUID SYSTEM
PIPING OUTSIDE CONTAINMENT

A. BACKGROUND

This position is intended to be used in conjunction with Branch Technical Position APCSB 3-1, attached to Standard Review Plan 3.6.1. The two positions together form an acceptable design approach for assuring that a plant can be safely shut down in the event of a piping failure outside containment. The background for this position is, therefore, the same as that for BTP APCSB 3-1 and reference should be made to that BTP for background information.

B. BRANCH TECHNICAL POSITION

1. High-Energy Fluid System Piping

a. Fluid Systems Separated from Essential Systems and Components

For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of Branch Technical Position (BIP) APCSB 3-1, a review of the piping layout and plant arrangement drawings should clearly show the effects of postulated piping breaks at any location are isolated or physically remote from essential systems and components.[1] At the designer's option, break locations as determined from I.C. and I.D of this postion may be assumed for this purpose.

b. Fluid System Piping In Containment Penetration Areas

Breaks need not be postulated in those portions of piping identified in B.2.c of BTP APCSB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the following additional design requirements:

(1) The following design stress and fatigue limits should not be exceeded:

For ASME Code, Section III, Class 1 Piping

(a) The maximum stress range should not exceed $2.4S_m$

(b) The maximum stress range between any two load sets (including the zero load set) should be calculated by Eq. (10) in Paragraph NB-3653, ASME Code, Section III, for normal and upset plant conditions and an operating basis earthquake (OBE) event transient.

If the calculated maximum stress range of Eq. (10) exceeds the limit of B.1.b(1)(a) but is not greater than $3S_m$, the limit of B.1.b(1)(c) should be met.

[1] Definitions of underlined phrases are given in Appendix A to Branch Technical Position APCSB 3-1, attached to Standard Review Plan 3.6.1.

If the calculated maximum stress range of Eq. (10) exceeds $3S_m$, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should meet the limit of B.1.b(1)(a) and the limit of B.1.b(1)(c).

(c) The cumulative usage factor should be less than 0.1 if consideration of fatigue limits is required according to B.1.b(1)(b).

(d) The maximum stress, as calculated by Eq. (9) in Paragraph NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping should not exceed $2.25S_m$ except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirements specified in SRP 3.9.3. Primary loads include those which are deflection limited by whip restraints.

<u>For ASME Code, Section III, Class 2 Piping</u>

(e) The maximum stress ranges as calculated by the sum of Eq. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering <u>normal and upset plant conditions</u> (i.e., sustained loads, occasional loads, and thermal expansion) and an OBE event should not exceed $0.8(1.2S_h + S_A)$.

(f) The maximum stress, as calculated by Eq. (9) in Paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping should not exceed $1.8S_h$.

Primary loads include those which are deflection limited by whip restraints. The exceptions permitted in (d) may also be applied provided that when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1 (see APCSB 3-1 B-2.C.4), the piping shall either be of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds shall be fully radiographed.

(2) Welded attachments, for pipe supports or other purposes, to these portions of piping should be avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of B.1.b(1).

(3) The number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used, the enclosed portion of <u>fluid system</u> piping should be seamless construction unless specific access provisions are made to permit inservice volumetric examination of the longitudinal welds.

(4) The length of these portions of piping should be reduced to the minimum length practical.

(5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) should not require welding directly to the outer surface of the piping (e.g., flued integrally-forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of B.1.b(1).

(6) Guard pipes provided for those portions of piping identified in B.2.c(2) of BTP APCSB 3-1 should be constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the guard pipe is part of the containment boundary. In addition, the entire guard pipe assembly should be designed to meet the following requirements and tests:

(a) The design pressure and temperature should not be less than the maximum operating pressure and temperature of the enclosed pipe under <u>normal plant conditions</u>.

(b) The design stress limits of Paragraph NE-3131(c) should not be exceeded under the loading associated with containment design pressure and temperature in combination with the safe shutdown earthquake.

(c) Guard pipe assemblies should be subjected to a single pressure test at a pressure not less than its design pressure.

c. <u>Fluid Systems Enclosed Within Protective Structures</u>

(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run within a protective structure or compartment designed to satisfy the plant arrangement provisions of B.1.b or B.1.c of BTP APCSB 3-1.

(a) At <u>terminal ends</u> of the run if located within the protective structure. Terminal ends are identified in APCSB 3-1 B.2.C.(3).

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping within the protective structure. (A terminal end, as determined by B.1.c(1)(a), may be considered as one of these extremes.)

(ii) At each location where the stresses[2/] exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress.[3/] Where the piping consists of a straight run without fittings, welded attachments, and valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.

(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run:

(a) At terminal ends of the run if located within the protective structure.

(b) At each intermediate pipe fitting, welded attachment, and valve.

If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.

(3) Applicable to (1) and (2) above:
If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of he fact that the above criteria might not require such a break location to be postualted.

d. Fluid Systems Not Enclosed Within Protective Structures

(1) With the exceptions of those portions of piping identified in B.1.b, breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run routed outside of, but alongside, above, or below, a protective structure or compartment containing essential systems and components and designed to satisfy the plant arrangement provisions of B.1.b or B.1.c of BTP APCSB 3-1.

Such piping should be considered as located adjacent to a protective structure if the distance between the piping and structure is insufficient to preclude impairment of the integrity of the structure from the effects of a postulated piping failure assuming the piping is unrestrained.

[2/] Stresses under normal and upset plant conditions, and an OBE event as calculated by Eq. (9) and (10), Para. NC-3652 of the ASME Code, Section III.

[3/] Select two locations with at least 10% difference in stress, or, if stresses differ by less than 10%, two locations separated by a change of direction of the pipe run.

(a) At terminal ends of the run if located adjacent to the protective structure. Terminal ends are identified in APCSB 3-1 B.2.C. (3).

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

(ii) At each location where the stresses[4/] exceed $0.8(1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress.[5/] Where the piping consists of a straight run without fittings, welded attachment, or valves, and all stresses are below $0.8(1.2S_h + S_A)$, a minimum of one location chosen on the basis of highest stress.

(2) Breaks in non-nuclear class piping should be postulated at the following locations in each piping or branch run:

(a) At terminal ends of the run if located adjacent to the protective structure.

(b) At each intermediate pipe fitting, welded attachment, and valve.

(3) Applicable to (1) and (2) above:
If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.

e. The designer should identify each piping run he has considered to postulate the break locations required by B.1.c and B.1.d above. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within a designated run in order to postulate the number of breaks required by these criteria.

2. Moderate-Energy Fluid System Piping

a. Fluid Systems Separated from Essential Systems and Components
For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of BTP APCSB 3-1, a review of the piping layout and plant

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4/ op. cit., p. 3.6.2-11, Footnote 2.
5/ op. cit., p. 3.6.2-11, Footnote 3.

arrangement drawings should clearly show that the effects of through-wall leakage cracks at any location in piping designed to seismic and non-seismic standards are isolated or physically remote from essential systems and components.

b. Fluid System Piping Between Containment Isolation Valves

Leakage cracks need not be postulated in those portions of piping identified in B.2.c. of BTP APCSB 3-1 provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120, and are designed such that the maximum stress range does not exceed $0.4(1.2S_h + S_A)$ for ASME Code, Section III, Class 2 piping.

c. Fluid Systems Within, or Outside and Adjacent to, Protective Structures

i. Through-wall leakage cracks should be postulated in seismic Category I fluid system piping located within, or outside and adjacent to, protective structures designed to satisfy the plant arrangement provisions of B.1.b. or B.1.c of BTP APCSB 3-1, except (1) where exempted by B.2.b and B.2.d, or (2) where the maximum stress range in these portions of Class 2 or 3 piping (ASME Code, Section III), or non-nuclear piping is less than $0.4(1.2S_h + S_A)$. The cracks should be postulated to occur individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed.

ii. Through-wall leakage cracks should be postulated in fluid system piping designed to non-seismic standards as necessary to satisfy B.3.d of BTP APCSB 3-1.

d. Moderate-Energy Fluid Systems in Proximity to High-Energy Fluid Systems

Cracks need not be postulated in moderate-energy fluid system piping located in an area in which a break in high-energy fluid system piping is postulated, provided such cracks would not result in more limiting environmental conditions than the high-energy piping break. Where a postulated leakage crack in the moderate-energy fluid system piping results in more limiting environmental conditions than the break in proximate high-energy fluid system piping, the provisions of B.2.c should be applied.

e. Fluid Systems Qualifying as High-Energy or Moderate-Energy Systems

Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods[6] but qualify as moderate-energy fluid systems for the major operational period.

[6] An operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., systems such as the reactor decay heat removal system qualify as moderate-energy fluid systems; however, systems such as auxiliary feedwater systems operated during PWR reactor startup, hot standby, or shutdown qualify as high-energy fluid systems).

3. Type of Breaks and Leakage Cracks in Fluid System Piping

a. Circumferential Pipe Breaks

The following circumferential breaks should be postulated in high-energy fluid system piping at the locations specified in B.1 of this position:

(1) Circumferential breaks should be postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range[7/] exceeds the limits specified in B.1.c(1)(b)(ii) and B.1.d(1)(b)(ii) but the circumferential stress range is at least 1.5 times the axial stress range. Instrument lines, one inch and less nominal pipe or tubing size should meet the provisions of Regulatory Guide 1.11.

(2) Where break locations are selected without the benefit of stress calculations, breaks should be postulated at the piping welds to each fitting, valve, or welded attachment. Alternatively, a single break location at the section of maximum stress range may be selected as determined by detailed stress analyses (e.g., finite element analyses) or tests on a pipe fitting.

(3) Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).

(4) The dynamic force of the jet discharge at the break location should be based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

(5) Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

b. Longitudinal Pipe Breaks

The following longitudinal breaks should be postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in B.3.a:

[7/] op. cit., p. 3.6.2-11, Footnote 2.

(1) Longitudinal breaks in fluid system piping and branch runs should be postulated in nominal pipe sizes 4-inch and larger, except where the maximum stress range[8/] exceeds the limits specified in B.1.c(1)(b)(ii) and B.1.d(1)(b)(ii) but the axial stress range is at least 1.5 times the circumferential stress range.

(2) Longitudinal breaks need not be postulated at:

(a) Terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds (if longitudinal welds are used, the requirements of B.3.b(1) apply).

(b) At intermediate locations where the criterion for a minimum number of break locations must be satisfied.

(3) Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diametrically-opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

(4) The dynamic force of the fluid jet discharge should be based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

(5) Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

c. Through-Wall Leakage Cracks

The following through-wall leakage cracks should be postulated in moderate-energy fluid system piping at the locations specified in B.2 of this position:

(1) Cracks should be postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

[8/] op. cit., p. 3.6.2-11, Footnote 2.

(2) Fluid flow from a crack should be based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one half pipe wall thickness in width.

(3) The flow from the crack should be assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period required to effect corrective actions.

C. REFERENCES

The references for this position are the same as for BTP APCSB 3-1, attached to Standard Review Plan 3.6.1.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.7.1 SEISMIC INPUT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The following areas relating to the seismic input are reviewed:

1. Design Response Spectra

The seismic input, as defined by the design response spectra corresponding to the specified ground acceleration for a site is reviewed. A response spectrum is a plot of the maximum response of a family of single-degree-of-freedom damped oscillators with different frequency characteristics when the base of the oscillator is subjected to vibratory motion indicated by an appropriate time motion record. The response spectra are usually displayed on tripartite log-log graph paper. When obtained from a recorded earthquake record, the response spectrum tends to be irregular, with a number of peaks and valleys. A design response spectrum is a relatively smooth plot, obtained from a number of individual response spectra derived from records of past earthquakes. For high frequencies, spectral acceleration approaches the bound set by the maximum ground acceleration. For intermediate frequencies, spectral velocity is amplified relative to the ground velocity. For low frequencies, spectral displacement is amplified relative to the ground displacement.

Design response spectra for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) (Ref. 1) are reviewed. The design response spectra in the free field applied at the finished grade or at the various foundation levels of Category I structures are reviewed. Where applicable, the basis for any response spectra that differ from those of Regulatory Guide 1.60 (Ref. 2) is reviewed.

Site Analysis Branch (SAB) is responsible for reviewing the proposed values of the SSE and OBE ground acceleration appropriate for the site (see Standard Review Plan 2.5.2).

2. Design Time History

For the time history analyses, a comparison of the response spectra obtained in the free field at the finished grade level and at the foundation level (obtained from an appropriate time history at the base of the soil-structure interaction system) with the design response spectra is reviewed for each of the damping values to be used in the design of structures, systems, and components. Alternatively if the design response spectra for the OBE and SSE are applied at the foundation levels of Category I structures in the free field, a comparison of the free field response spectra at the foundation level derived from an actual or synthetic time history, applied at the base of the soil-structure interaction system, with the design response spectra is reviewed for each of the damping values to be used in the analysis.

The current practice in the design of nuclear power plants is to use the response spectrum technique for seismic design of buildings and structures. In this technique, the input for the dynamic analysis of the model of major building structural elements is usually given in terms of a design spectrum appropriate for the seismic characteristics of the plant site. The analysis of interior equipment or component may also be based on the design response spectrum. However, such analysis requires an integrated model of the building and interior equipment which may not be practical for the many components which must be considered. In addition, care is required in modeling to assure accuracy of results where orders of magnitude differences exist in stiffness and mass characteristics between the building and component elements of the model.

For the analysis of interior equipment, where the equipment analysis is decoupled from the building, a compatible time history is needed for computation of the time-history response of each floor. The design floor spectra for equipment are obtained from this time history information.

In addition to the comparison of the response spectra derived from the time-history with the design response spectra, the period intervals at which the spectra values are calculated are also reviewed.

3. Critical Damping Values

The specific percentage of critical damping values used for Category I structures, systems, components, and soil are reviewed for both the OBE and the SSE. Critical damping is the amount of damping that would completely eliminate vibration. Although the use of critical damping is of little practical importance in itself, it assumes great significance as a measure of the damping capacity of a structure. Damping is conveniently expressed in the form of some percentage of critical damping.

Vibrating structures have energy losses which depend on numerous factors, such as material characteristics, stress levels, and geometric configuration. This dissipation of energy, or damping effect, occurs because a part of the excitation input is transformed into heat, sound waves, and other energy forms. The response of a system to dynamic loads is a function of the amount and type of damping existing in the system. A knowledge of appropriate values to represent this characteristic is essential for obtaining realistic results in dynamic analysis.

In practical seismic analysis, which usually employs linear methods of analysis, damping is also used to account for many nonlinear effects such as changes in boundary conditions, joint slippage, plastic hinges, concrete cracking, gaps, and other effects which tend to alter response amplitude. In real structures, it is often impossible to separate "true" material damping from system damping, which is the measure of the energy dissipation, from the nonlinear effects. Overall structural damping used in design is normally determined by observing experimentally the total response of the structure.

Only the overall damping used for Category I structures, systems, components, and soil are reviewed. Where applicable, the basis for any damping values that differ from those given in Regulatory Guide 1.61 (Ref. 3) is reviewed.

4. Supporting Media for Category I Structures

The description of the supporting media for each Category I structure is reviewed, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, width of the structural foundation, total structural height, and soil properties to permit evaluation of the applicability of finite element or lumped spring approaches for soil-structure interaction analysis.

SAB is responsible for evaluating the physical properties of foundation soil and rock (see Standard Review Plans 2.5.1 and 2.5.4).

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I are as follows:

1. Design Response Spectra

Design response spectra for the OBE and SSE are considered to be acceptable if the associated amplification factors are in accordance with Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants," for all damping values.

As noted in Regulatory Guide 1.60, there are site circumstances where the design response spectra are more appropriately developed to suit the particular site characteristics. Design response spectra based upon site-dependent analysis must be derived considering in situ variable soil properties, a representative number of site earthquake records, vertical amplification, possible slanted soil layers, and the influence of any predominant soil layers. The finite element approach or equivalent should be used to consider variable soil properties and nonlinear stress-strain relations in the soil media. The procedures used to obtain site-dependent design response spectra are reviewed on a case-by-case basis.

It should be noted that to be acceptable the design response spectra should be specified for three mutually orthogonal directions; two horizontal and one vertical. Since the two horizontal spectra have an equal probability of occurrence in any horizontal direction, current practice is to assume that the maximum accelerations in the two horizontal directions are equal, while the maximum vertical acceleration is 2/3 of the maximum horizontal acceleration.

2. Design Time History

In developing the design time history to be used at the base of the soil-structure interaction system, the following represents an acceptable procedure:

a. The design response spectra are defined for the free field and applied at the proposed finished grade level of the site.

b. Using an appropriate analysis method, with appropriate soil properties, obtain a time history at the base of the idealized soil profile.* One acceptable method for deconvolution analysis of the design response spectra at finished grade is a combined application of the SHAKE and LUSH computer codes (Refs. 4, 5). Use of other equivalent computer codes and analysis techniques is also acceptable. When the time history obtained from these methods is applied at the base of the idealized soil profile and the soil-structure interaction system, the resulting free field vibratory ground motion at finished grade level should give response spectra that envelop the design response spectra. This time history should appropriately account for variation in the soil properties at the site. In addition, when the time history obtained is applied at the base of the idealized soil profile, using appropriate soil properties, the vibratory motion calculated at the elevation of Category I structural foundations should, in general, give response spectra at all frequencies (.2 cps to 50 cps), not less than 60% of the design response spectra. The same limitation applies to the response spectra obtained at the foundation level in the free field for the soil-structure interaction system. Response spectral values in the idealized soil profile at the foundation level and those at the foundation level of the interaction system that are less than 60% of the corresponding design response spectral values may be acceptable provided they can be justified. The justification will be reviewed on a case-by-case basis.

c. The time history developed in item 2.b. above should be used at the base of the soil-structure interaction system, with appropriate soil properties, for subsequent soil-structure interaction analysis. The analysis method used should account for the strain dependency of soil modulus and damping. The peaks in the floor response spectra obtained from such a time history need be broadened by only ± 10% of the frequencies corresponding to the peaks.

d. An alternate and acceptable procedure is to apply the design response spectra at the foundation level of Category I structures in the free field. In this case, the design time history for use in the seismic analysis is acceptable if the response spectra in the free field at the foundation level obtained from the time history envelop the design response spectra for all damping values actually used in the analyses. The peaks in the floor response spectra obtained from such a time history should be broadened by a minimum of ± 15% of the frequencies corresponding to the peaks.

*Note: The idealized soil profile is the soil-structure interaction system without the structure.

The frequency intervals at which the spectra values are calculated from the design time history are to be small enough such that any reduction in these intervals does not result in more than 10% change in the computed spectra. Table 3.7.1-1 provides an acceptable set of frequencies at which the response spectra should be calculated. Another acceptable method is to choose a set of frequencies such that each frequency is within 10% of the previous one.

The acceptance criterion for meeting the spectra-enveloping requirement is that no more than five points of the spectra obtained from the time history should fall below, and no more than 10% below, the design response spectra.

Table 3.7.1-1
Suggested Frequency Intervals for Calculation of
Response Spectra

Frequency Range (hertz)	Increment (hertz)
0.2 - 3.0	.10
3.0 - 3.6	.15
3.6 - 5.0	.20
5.0 - 8.0	.25
8.0 - 15.0	.50
15.0 - 18.0	1.0
18.0 - 22.0	2.0
22.0 - 34.0	3.0

3. Critical Damping Values

The specific percentage of critical damping values used in the analyses of Category I structures, systems, and components are considered to be acceptable if they are in accordance with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." Damping values in this guide are based upon the current (1973) state of the art. Higher damping values may be used in a dynamic seismic analysis if documented test data are provided to support them. These values would be reviewed and accepted by the staff on a case-by-case basis. The damping value for soil must be based upon actual measured values or other pertinent laboratory data considering variation in soil properties and strains within the soil.

4. Supporting Media for Category I Structures

To be acceptable, the description of supporting media for each Category I structure must include foundation embedment depth, depth of soil over bedrock, width of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, and density as a function of depth.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below as may be appropriate for a particular case.

1. Design Response Spectra

Design response spectra for the OBE and SSE for all damping values are checked to assure that the spectra are in accordance with the acceptance criteria as given in Section II. Any differences between the regulatory guide spectra and the proposed response spectra which have not been adequately justified are identified and the applicant is informed of the need for additional technical justification.

Design response spectra based upon site-dependent analyses are reviewed to assure that the procedure used to develop these spectra considers in situ variable soil properties, a representative number of site earthquake records, vertical amplification, possible slanted soil layers, nonlinear stress-strain relations, and the influence of possibly predominant soil layers.

2. Design Time History

Methods of defining the design time history are reviewed to ascertain that the acceptance criteria of Section II.2 are met.

3. Critical Damping Values

The specific percentage of critical damping values for the OBE and SSE used in the analyses of Category I structures, systems, and components are checked to assure that the damping values are in accordance with the acceptance criteria as given in Section II.3. Any differences in damping values which have not been adequately justified are identified and the applicant is informed of the need for additional technical justification.

4. Supporting Media for Category I Structures

The description of the supporting media is reviewed to verify that sufficient information, as specified in the acceptance criteria of Section II.4 is included. Any deficiency in the required information is identified and a request for additional information is transmitted to the applicant.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60, 'Design Response Spectra for Nuclear Power Plants.' The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61, 'Damping Values for Seismic Analysis of Nuclear Power Plants.' The synthetic time history used for seismic design of Category I plant structures, systems, and components is adjusted in amplitude and frequency content to obtain reponse spectra that envelop the design response spectra specified for the site. Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings."

Alternatively, if a site-dependent analysis is used to develop the shape of the design response spectra, the language of the evaluation findings should be similar to the following:

"The site-dependent analysis has used a finite element approach to develop the seismic design response spectra from site-related information, including site time histories. This approach, used in lieu of the response spectra specified in Regulatory Guide 1.60, is acceptable since the free field response spectra at finished grade level (or at the structural foundation level) include consideration of appropriate amplification factors based upon an acceptable set of site earthquake records, and the analysis has taken into account actual soil properties at the site and includes consideration of appropriate damping values corresponding to the calculated soil stress levels. The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61, 'Damping Values for Seismic Analysis of Nuclear Power Plants.'

"The use of the site-dependent analysis and the damping values of Regulatory Guide 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings."

V. <u>REFERENCES</u>

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide No. 1.60, "Design Response Spectra for Nuclear Power Plants."

3. Regulatory Guide No. 1.61, "Damping Values for Seismic Analysis for Nuclear Power Plants."

4. Per B. Schnabel, J. Lysmer, and H. B. Seed, "SHAKE - A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," EERC 72-12, Earthquake Engineering Research Center, University of California, Berkeley (1972).

5. J. Lysmer, T. Udaka, H. B. Seed, and R. Hwang, "LUSH - A Computer Program for Complex Response Analysis of Soil-Stucture Systems," Draft Report, Earthquake Engineering Research Center, University of California, Berkeley (1974).

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.7.2 SEISMIC SYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic system analysis described in the applicant's safety analysis report (SAR) are reviewed.

1. Seismic Analysis Methods

For all Category I structures, systems, and components, the applicable seismic analysis methods (response spectra, time history, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed, including the modeling of foundation torsion, rocking and translation, is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees of freedom is reviewed. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed. If tests or empirical methods are used in lieu of analysis for any Category I structure, the testing procedure, load levels, and acceptance basis are also reviewed.

2. Natural Frequencies and Response Loads

For the operating license review, significant natural frequencies and response loads for major Category I structures are reviewed. In addition, the response spectra at critical major Category I equipment elevations and points of support are reviewed.

3. Procedures Used for Analytical Modeling

The criteria and procedures used in modeling for the seismic system analyses are reviewed. The criteria and bases for determining whether a component or structure is analyzed as part of a system analysis or independently as a subsystem are also reviewed.

4. Soil-Structure Interaction

The design earthquake input is defined in the "free field," i.e., the effect of the presence of structures is not included. When plants are founded on soil deposits or soft media, the resulting motions of the base slab will differ from those defined at the same elevation in the free field, due to deformability of the foundation. This difference in the base slab motion and the free field motion is known as soil-structure interaction effect.

As applicable, the methods of soil-structure interaction analysis used in the seismic system analysis and their bases are reviewed. The factors to be considered in accepting the validity of a particular method are: (1) the extent of embedment, (2) the depth of soil over rock, and (3) the layering of the soil strata. If the finite element approach is used, the criteria for determining the location of the bottom boundary and side boundary are reviewed. The procedures by which strain-dependent soil properties (damping, shear modulus) are incorporated in the analysis are also reviewed.

If lumped spring methods are used, the parameters used in the analysis are reviewed. Also, the procedures by which strain-dependent soil properties (damping, shear modulus), layering, and variation of soil properties are incorporated in the analysis are reviewed. The applicability of a lumped spring method used for the particular site conditions is reviewed.

Any other methods used for soil-structure interaction analysis are also reviewed as is any basis for not using soil-structure interaction analysis. The procedures used to account for effects of adjacent structures on structural response in the soil-structure interaction analysis are reviewed.

5. Development of Floor Response Spectra

The procedures for developing floor response spectra considering the three components of earthquake motion are reviewed. If a modal response spectrum method of analysis is used to develop floor response spectra, the justification for its conservatism and equivalency to a time history method is evaluated.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of structures, systems, and components are reviewed.

7. Combination of Modal Responses

When a response spectrum approach is used for calculating the seismic response of structures, systems, or components, the phase relationship between various modes is lost. Only the maximum response for each mode can be determined. The maximum responses for modes do not in general occur at the same time and these responses have to be combined according to some procedure selected to approximate or bound the response of the system. When a response spectra method is used, the description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) is reviewed, including that for modes with closely spaced frequencies.

8. Interaction of Non-Category I Structures with Category I Structures

The design criteria to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Category I structures or portions thereof are reviewed. The procedures that are used to protect Category I structures from the structural failure of non-Category I structures, due to seismic effects, are reviewed.

9. Effects of Parameter Variations on Floor Responses

The procedures that are used to consider the effects of the expected variations of structural properties, dampings, soil properties, and soil-structure interaction on the floor response spectra and time histories are reviewed.

10. Use of Constant Vertical Static Factors

Where applicable, justification for the use of constant static factors as vertical response loads for designing Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is reviewed.

11. Methods Used to Account for Torsional Effects

The method employed to consider torsional effects in the seismic analysis of Category I structures is reviewed. The review includes the evaluation of the conservatism of any approximate methods to account for torsional accelerations in the seismic design of Category I structures.

12. Comparison of Responses

For the operating license review, where applicable, the comparison of seismic responses for major Category I structures using modal response spectrum and time history approaches is evaluated.

13. Methods for Seismic Analysis of Category I Dams

The analytical methods and procedures that will be used for seismic analysis of Category I dams are reviewed. The assumptions made, the boundary conditions used, and the procedures by which strain-dependent soil properties are incorporated in the analysis are reviewed.

14. Determination of Category I Structure Overturning Moments

The description of the dynamic methods and procedures used to determine design overturning moments for Category I structures are reviewed.

15. Analysis Procedure for Damping

The analysis procedure to account for the damping in different elements of the model of a coupled system is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used.

1. Seismic Analysis Methods

The seismic analysis of all Category I structures, systems, and components should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.

a. Dynamic Analysis Method

A dynamic analysis (e.g., response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items:

(1) Use of either the time history method or the response spectrum method.

(2) Use of appropriate methods of analysis to account for effects of soil-structure interaction.

(3) Consideration of the torsional, rocking, and translational responses of the structures and their foundations.

(4) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and applicable non-Category I structures and plant equipment. The number is considered adequate when additional degrees of freedom do not result in more than a 10% increase in responses. Alternately, the number of degrees of freedom may be taken equal to twice the number of modes with frequencies less than 33 cps.

(5) Investigation of a sufficient number of modes to assure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10% increase in responses.

(6) Consideration of maximum relative displacements among supports of Category I structures, systems, and components.

(7) Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.

b. Equivalent Static Load Method

An equivalent static load method is acceptable if:

(1) Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar structures may be submitted in support of the use of the simplified method.

(2) The design and associated simplified analysis account for the relative motion between all points of support.

(3) To obtain an equivalent static load of a structure, equipment, or component which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used if adequate justification is provided.

In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.

2. Natural Frequencies and Response Loads

To be acceptable for the operating license review, the following information should be provided.

a. A summary of natural frequencies, response loads, mode shapes, and modal responses for a representative number of major Category I structures, including the containment building.

b. A time history of acceleration (or equivalent parameters) or response spectrum at the major plant equipment elevations and points of support.

3. Procedures Used for Analytical Modeling

A nuclear power plant facility consists of very complex structural systems. To be acceptable, the stiffness, mass, and damping characteristics of the structural systems should be adequately incorporated into the analytical models. Specifically, the following items should be considered in analytical modeling:

a. Designation of Systems Versus Subsystems

Major Category I structures that are considered in conjunction with foundation media in forming a soil-structure interaction model are defined as "seismic systems." Other Category I structures, systems, and components that are not designated as "seismic systems" should be considered as "seismic subsystems."

b. Decoupling Criteria for Subsystems

It can be shown, in general, that the absolute frequencies of systems and subsystems have negligible effect on the error due to decoupling. It can be shown that the mass ratio, R_m, and the frequency ratio, R_f, govern the results where R_m and R_f are defined as:

$$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Mass that supports the subsystem}}$$

$$R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Frequency of the dominant support motion}}$$

The following criteria are acceptable:

(1) If $R_m < 0.01$, decoupling can be done for any R_f.

(2) If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $0.8 \geq R_f \geq 1.25$.

(3) If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

If the subsystem is comparatively rigid and also is rigidly connected to the primary system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections, e.g., pipe supported by hangers, the subsystem need not be included in the primary model. In most cases the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the reactor coolant system, which is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure.

c. Lumped Mass Considerations

The acceptance criteria given under II.1.a(4) are applicable.

d. Modeling for Three Component Input Motion

In general, three-dimensional models should be used for seismic analyses. However, simpler models can be used if justification can be provided that the coupling effects of those degrees of freedom that are omitted from the three-dimensional models are not significant.

4. Soil-Structure Interaction

Table 3.7.2-1 summarizes acceptable procedures for soil-structure interaction analyses. To be acceptable, a finite element technique (Ref. 1, 11) or equivalent is required as the analytical tool for soil-structure interaction analysis for all Category I structures where the foundations are deeply embedded in soil. This technique may also be used for all cases where soil-structure interaction is involved. For structures supported on rock, a fixed base approach is acceptable. For shallowly embedded structures on shallow soil overburden over rock, or layered soil with varying soil properties, either the finite element approach or multiple mass-spring (shear beam) approach (Ref. 2) may be used.

Table 3.7.2-1

Acceptable Methods for Soil-Structure Interaction Analysis

			Soil Foundation**		
			Shallowly Embedded Case		
Method of Soil-Structure Interaction Analysis	Rock†† Foundation	Deeply Embedded Case†	Deep Soil Found. w/Uniform Properties	Deep Soil Found. w/Layered Properties	Shallow Soil Foundation
Single Lumped Mass-Spring Approach	x		x		
Multiple Lumped Mass-Spring Approach*	x		x	x	x
Finite Element Approach*	x	x	x	x	x

*Or equivalent.
†Deep embedment: actual embedded depth >15% of the least base width or other appropriate value to be justified.
††A medium for which the soil-structure interaction effect is negligible or alternatively, a medium with a shear wave velocity greater than or equal to 3500 fps.
**Soil foundation means the depth of soil between the bottom of the foundation slab and the base rock.

The lumped mass-spring method or "compliance function method" may be used for cases where the depth of embedment is shallow and the soil foundation is relatively uniform and of sufficient depth that it can be considered as a half-space.

The acceptability of the procedures used to account for the effects of adjacent structures on structural responses in soil-structure interaction analyses will be reviewed on a case-by-case basis.

Other techniques which give an equivalent degree of conservatism as the appropriate acceptable technique and which are justified are also acceptable. Since the finite element and the lumped mass spring approaches are the most commonly used in current practice, these two approaches are discussed below.

Finite Element Approach

The finite element approach may be used for all cases where soil-structure interaction is involved. The acceptance criteria for different aspects of the finite element technique are as follows:

a. Boundary Conditions

(1) Bottom Boundary

Wherever possible the base of the model is placed at the rock level. However, if the bedrock is too deep, the bottom boundary can also be placed at a reasonable depth from the structure foundation such that the effect of soil-structure interaction below this depth is negligible. This should be justified. The nodes on the base of the finite element model are fixed and the earthquake input motion is applied there.

(2) Side Boundaries

The side boundaries should be kept at a distance away from the structures such that the motion at the boundary is not affected by the structural motion. It is acceptable if the distance of the boundaries from the edge of the structure is kept equal to or greater than three times the base slab dimension. Any other selection of the side boundaries should be justified.

b. Soil Properties

In a finite element model, different kinds of soils present should be adequately represented. Since the soil moduli and damping ratios are in general highly strain-dependent, the strain-compatible properties for each layer or element should be computed with the use of soil property curves which relate the moduli and damping ratios with strain for the soils present at the site.

Lumped Mass-Spring Approach

In the lumped mass-spring approach the compliance functions in common use are based upon the analytical solution of a rigid plate on an elastic half-space. Reference 8 gives an acceptable set of spring and damping constants for a plate supported on an elastic half-space. Compliance functions for layered sites have also been developed but their applicability for soil-structure interaction analysis for layered sites should be justified by the applicant.

As mentioned earlier, the lumped mass-spring method may be used for cases where the depth of embedment is shallow and the soil foundation is relatively uniform and of sufficient depth that it can be considered as a half-space. The justification for the sufficiency of the depth should be provided by the applicant and will be reviewed by the staff on a case-by-case basis. This approach should not be used for other cases for the following reasons:

(1) It is well known that the lumped soil spring parameters are frequency dependent. This dependence for a layered space is expected to be large. Thus, using a constant set of soil spring parameters for all situations may lead to incorrect results.

(2) The lumped parameters are usually derived without any regard to the actual embedment of the structure. Thus, for all structures, whether sitting on the ground surface or embedded to any extent, the answers obtained using lumped parameters will be the same. It is known that the earthquake motion is not the same at all elevations of the soil profile, hence the motion at the foundation and surface will, in general, be different. Effects of this

variation in the earthquake motion itself and also the inertia effort of the soil cannot be taken care of adequately by the lumped parameter method.

(3) In the lumped parameter technique the entire soil medium is assumed to be homogeneous and elastic. However, in any stratum of soil deposits there are generally many different types of soils present. The properties also are strain-dependent. These factors could have a significant effect on the structural response, which cannot be accounted for by a single set of stiffness and damping values.

(4) As stated earlier, if the site situation can be approximated as an elastic half-space, the design earthquake can be directly input in this approach. This is a relative advantage of the method. However, as Whitman (Ref. 8) points out, for a structure sitting on a stratum whose thickness is less than twice the width of the foundation, the effects of soil amplification and soil-structure interaction cannot be separated. So, he notes, the input motion at the spring support in such cases should be the rock motion, not the design motion specified at the foundation level. It is, therefore, obvious that in cases of shallow overburden, an uncertainty about the input motion exists.

Thus, the actual site conditions for a particular plant should be carefully reviewed before accepting the lumped spring approach.

5. Development of Floor Response Spectra

To be acceptable, the floor response spectra should be developed taking into consideration the three components of the earthquake motion. The individual floor response spectral values for each frequency are obtained for one vertical and two mutually perpendicular horizontal earthquake motions and are combined according to the "square root of the sum of the squares" method to predict the total floor response spectrum for that particular frequency.

In general, development of the floor response spectra is acceptable if a time history approach is used. If a modal response spectra method of analysis is used to develop the floor response spectra, the justification for its conservatism and equivalency to that of a time history method must be demonstrated by representative examples.

6. Three Components of Earthquake Motion

Depending upon what basic methods are used in the seismic analysis, i.e., response spectra or time history method, the following two approaches are considered acceptable for the combination of three-dimensional earthquake effects. (Ref. 3, 4, and 5.)

a. Response Spectra Method

When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model.

b. Time History Analysis Method

When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (1) To obtain maximum responses due to each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in (a) except that the maximum responses are calculated using the time history method instead of the spectrum method. (2) To obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable, the earthquake motions specified in the three different directions should be statistically independent.

7. Combination of Modal Responses

When the response spectrum method of analysis is used to determine the dynamic response of damped linear systems, the most probable response is obtained as the square root of the sum of the squares of the responses from individual modes. Thus, the most probable system response, R, is given by

$$R = \left(\sum_{k=1}^{N} R_k^2\right)^{1/2} \quad (1)$$

where R_k is the response for the k^{th} mode and N is the number of significant modes considered in the modal response combination.

When modes with closely spaced modal frequencies exist, an acceptable method for obtaining the system response is to take the absolute sum of the responses of the closely spaced modes and combine this sum with other remaining modal responses using the square root of the sum of the squares rule. Two modes having frequencies within 10% of each other are considered as modes with closely spaced frequencies.

This approach is simple and straightforward in all those cases where the group of modes with closely spaced frequencies is tightly bundled, i.e., the lowest and the highest modes of the group are within 10% of each other. However, when the group of closely spaced modes is spaced widely over the frequency range of interest (while the frequencies of the adjacent modes are closely spaced), the absolute sum method of combining responses tends to yield over-conservative results. To obviate this problem, a general approach applicable to all modes is considered appropriate. The following equation is merely a mathematical representation of this approach.

The most probable system response, R, is given by

$$R = \left(\sum_{K=1}^{N} R_k^2 + 2 \sum |R_\ell R_m|\right)^{1/2} \qquad (2)$$

where the second summation is to be done on all ℓ and m modes whose frequencies are closely spaced to each other.

Other approaches which give an equivalent degree of conservatism to the above methods, and which are adequately justified are also acceptable.

8. Interaction of Non-Category I Structures with Category I Structures

To be acceptable, the interfaces between Category I and non-Category I structures and plant equipment must be designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures and plant equipment. In addition, a statement indicating the fact that all non-Category I structures meet any one of the following requirements should be provided.

a. The collapse of any non-Category I structure will not cause the non-Category I structure to strike a seismic Category I structure or component.

b. The collapse of any non-Category I structure will not impair the integrity of seismic Category I structures or components.

c. The non-Category I structures will be analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.

9. Effects of Parameter Variations on Floor Response Spectra

To be acceptable, consideration should be given in the analyses to the effects on floor response spectra (e.g., peak width and period coordinates) of expected variations of structural properties, dampings, soil properties, and soil-structure interactions. An acceptable method for determining the amount of peak widening associated with the structural frequency is described below.

Let f_j be the j^{th} mode structural frequency which is determined from the structure model. The variation in the structural frequency is determined by evaluating the individual frequency variation due to the variation in each parameter that has significant effect, such as the soil shear modulus, damping, and material density. The total frequency variation, Δf_j, is then determined by taking the square root of the sum of squares of a minimum variation of $0.05f_j$ and the individual frequency variation $(\Delta f_j)_n$, that is,

$$\Delta f_j = \pm \sqrt{(0.05f_j)^2 + \Sigma(\Delta f_j)_n^2}$$

A value of $0.10 f_j$ is used if the actual computed value of Δf_j is less than $0.10f_j$.

If the above procedure is not used, the peak width should be increased by a minimum of $\pm$ 15% to be acceptable.

Time histories of floor motion may be used as excitations to the subsystems. To account for the effect of possible frequency variation of the structure, the same time history data can be used with at least three different time intervals: Δt and $(1 \pm \Delta f_j/f_j)\ \Delta t$, for the analysis of equipment, where f_j is the dominant structural frequency and Δf_j is a parameter defining the frequency variation due to uncertainties as given above. This variation of the time interval has a similar effect to widening the spectral peak when generating the smoothed response spectrum. If one of the equipment frequencies, f_e, is known to be within the range $f_j \pm \Delta f_j$, the time history can also be used with a time interval of $(1 - (f_e - f_j)/f_j)\ \Delta t$. This method of modifying the time history data is described in Reference 12. As in the case of the broadened response spectrum, the variation of time interval has little effect on those equipment response modes with frequencies outside the range of the broadened peak of the corresponding spectrum.

10. Use of Constant Vertical Static Factors

The use of constant vertical load factors as vertical response loads for the seismic design of all Category I structures, systems, and components in lieu of the use of a vertical seismic system dynamic analysis is acceptable only if it can be justified that the structure is rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction is more than 33 cps.

11. Methods Used to Account for Torsional Effects

An acceptable method of treating the torsional effects in the seismic analysis of Category I structures is to carry out a dynamic analysis which incorporates the torsional degrees of freedom. An acceptable alternative, if properly justified, is the use of static factors to account for torsional accelerations in the seismic design of Category I structures in lieu of the use of a combined vertical, horizontal and torsional system dynamic analysis.

12. Comparison of Responses

The responses obtained from both modal analysis response spectrum and time history methods at selected points in typical Category I structures should be compared to demonstrate approximate equivalency between the two methods.

13. Methods for Seismic Analysis of Category I Dams

For the analysis of all Category I dams, a finite element approach which takes into consideration the time history of forces (due to both horizontal and vertical earthquake loadings), the behavior of the soil under simulated earthquake loadings, and an evaluation of deformations should be used. Appropriate nonlinear stress-strain relations for the soil are to be used in the finite element analysis. For earth-filled dams, procedures presented in References 6 and 7 are acceptable. For rock-filled dams, the analytical procedure used will be reviewed on a case-by-case basis.

14. Determination of Category I Structure Overturning Moments

To be acceptable, the determination of the design moment for overturning should incorporate the following items:

a. Three components of input motion.

b. Conservative consideration of vertical and lateral seismic forces.

15. Analysis Procedure for Damping

Either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping.

For the composite modal damping approach, two techniques of determining an equivalent modal damping matrix or composite damping matrix are commonly used. They are based on the use of the mass or stiffness as a weighting function in generating the composite modal damping. The formulations lead to:

$$\bar{\beta}_j = \{\phi\}^T [\bar{M}] \{\phi\} \tag{3}$$

$$\bar{\beta}_j = \frac{\{\phi\}^T [\bar{K}] \{\phi\}}{\{\phi\}^T [K] \{\phi\}} \tag{4}$$

where

$[K]$ = assembled stiffness matrix,

$\bar{\beta}_j$ = equivalent modal damping ratio of the j^{th} mode,

$[\bar{K}]$, $[\bar{M}]$ = the modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix, and

$\{\phi\}$ = j^{th} normalized modal vector.

For models that take the soil-structure interaction into account by the lumped soil spring approach, the method defined by equation (4) is acceptable. For fixed base models, either equation (3) or (4) may be used. Other techniques based on modal synthesis (Ref. 9) have been developed and are particularly useful when more detailed data on the damping characteristics of structural subsystems are available. The modal synthesis analysis procedure consists of (1) extraction of sufficient modes from the structure model, (2) extraction of sufficient modes from the finite element soil model, and (3) performance of a coupled analysis using the modal synthesis technique, which uses the data obtained in steps (1) and (2) with appropriate damping ratios for structure and soil subsystems. This method is based upon satisfaction of displacement compatibility and force equilibrium at the system interfaces and utilizes subsystem eigenvectors as internal generalized coordinates. This method results in a nonproportional damping matrix for the composite structure and equations of motion have to be solved by direct integration or by uncoupling them by use of complex eigenvectors.

Another technique which is also considered acceptable for estimating the equivalent modal damping of a soil-structure interaction model is given by Tsai (Ref. 10).

II. REVIEW PROCEDURES

For each area of review, the following procedure is implemented. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Seismic Analysis Methods

   For all Category I structures, systems, and components, the applicable methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to ascertain that the techniques employed are in accordance with the acceptance criteria as given in Section 2.1 of this plan. If empirical methods or test are used in lieu of analysis for any Category I structure, these are evaluated to determine whether or not the assumptions are conservative, and whether the test procedure adequately models the seismic response.

2. Natural Frequencies and Response Loads

   For the operating license review, the summary of natural frequencies and response loads is reviewed for compliance with the acceptance criteria in Section II.2 of this plan.

3. Procedures Used for Analytical Modeling

   The procedures used for modeling for seismic system analyses are reviewed to determine whether the three-dimensional characteristics of structures are properly modeled in accordance with the acceptance criteria of Section II.3 and all significant degrees of freedom have been incorporated in the models. The criteria for decoupling of a structure, equipment, or component and analyzing it separately as a subsystem are reviewed for conformance with the acceptance criteria given in Section II.3.

4. Soil-Structure Interaction

   The methods of soil-structure interaction analysis used are examined to determine that the techniques employed are in accordance with the acceptance criteria as given in Section II.4. Typical mathematical models for soil-structure interaction analysis are reviewed to ensure the adequacy of the representation in accordance with Section II.4 of this plan. In addition, the methods used to assess the effects of adjacent structures on structural response in soil-structure interaction analysis are reviewed to establish their acceptability.

5. Development of Floor Response Spectra

   Procedures for developing the floor response spectra are reviewed to verify that they are in accordance with the acceptance criteria specified in Section II.5. If a modal response spectrum method of analysis is used to develop the floor response spectra, its conservatism compared to that of a time history approach is reviewed. The applicant is requested to provide additional technical justification for any procedure considered not adequately justified.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of structures, systems, and components are reviewed to determine compliance with the acceptance criteria of Section II.6. Any other procedures that are considered not adequately justified are so identified, and the applicant is asked to provide additional justification.

7. Combination of Modal Responses

The procedures for combining modal responses (shears, moments, stresses, deflections, and accelerations) for closely spaced modes are reviewed to determine compliance with the acceptance critieria of Section II.7, when a response spectrum modal analysis method is used.

8. Interaction of Non-Category I Structures with Category I Structures

The design and analysis criteria for interaction of non-Category I structures with Category I structures are reviewed to ensure compliance with the acceptance criteria of Section II.8.

9. Effects of Parameter Variations on Floor Response Spectra

The seismic system analysis is reviewed to determine whether the analysis considered the effects of expected variations of structural properties, dampings, soil properties, and soil-structure interaction on floor responses spectra (e.g., peak width and period coordinates) and to determine compliance with the acceptance criteria of Section II.9.

10. Use of Constant Vertical Static Factors

Use of constant static factors as response loads in the vertical direction for the seismic design of any Category I structure, system or component in lieu of a detailed dynamic method is reviewed to determine that constant vertical static factors are used only if the structure is rigid in the vertical direction.

11. Methods Used to Account for Torsional Effects

The methods of seismic analysis are reviewed to determine that the torsional effects of vibration are incorporated by including the torsional degrees of freedom in the dynamic model. Justification provided by the applicant for the use of any approximate method to account for torsional effects is judged to assure that it results in a conservative design. If such justification is deemed inadequate, it is identified and the applicant requested to provide additional justification.

12. Comparison of Responses

Where applicable, the responses obtained from both response spectrum and time history methods at selected points in major Category I structures are compared to judge the accuracy of the analyses conducted. Large differences in the results obtained by use of the two methods are identified and the applicant is asked to discuss the reasons for the large differences.

13. Methods for Seismic Analysis of Category I Dams
Methods for the seismic analysis of Category I dams are reviewed to determine compliance with the acceptance criteria of Section II.13.

14. Determination of Category I Structure Overturning Moments
Methods to determine Category I structure overturning moments are reviewed to determine compliance with the acceptance criteria of Section II.14.

15. Analysis Procedure for Damping
The analysis procedure to account for damping in different elements of the model of a coupled system is reviewed to determine that it is in accordance with the acceptance criteria of Section II.15. It is verified that composite damping based on mass weighting is not used for sites where the lumped mass spring approach is used to model the soil-structure interaction.

Any matters identified during the review of the SAR where additional information or justification are needed are included in the "Additional Technical Information Request" prepared by SEB for transmittal to the Division of Reactor Licensing. Such requests not only identify any portions of the seismic system analysis considered unacceptable without further justification, but also specify the changes that should be made in the SAR to meet the acceptance criteria. Subsequent amendments of the SAR received in response to these SEB requests are reviewed for conformance with the staff positions.

IV. EVALUATION FINDINGS

(Combined for Sections 3.7.2 and 3.7.3)

The reviewer verifies that sufficient information has been provided and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The scope of review of the seismic system and subsystem analysis for the (__________) plant included the seismic analysis methods for all Category I structures, systems, and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, seismic analysis of Category I dams, evaluation of Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I structures and piping with Category I structures and piping and the effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside containment.

"The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods form the bases for the analyses of all major Category I structures, systems, and components. When the modal response spectrum method is used, governing response parameters are combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses are used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of structures, systems, and components are generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis will be employed for all structures, systems, and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning are considered.

"The finite element (or the lumped soil spring) approach is used to evaluate soil-structure interaction and structure-to-structure interaction effects upon seismic responses. For the finite element analysis, appropriate nonlinear stress-strain and damping relationships for the soil are considered in the analysis.

"For the analysis of Category I dams, a finite element approach which takes into consideration the time history of forces, the behavior and deformation of the dam due to the earthquake, and applicable stress-strain relations is used.

"We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicant provide an acceptable basis for the seismic design."

V. <u>REFERENCES</u>

1. P. K. Agrawal, "Comparative Study for Soil-Structure Interaction Effect by the Soil Spring and Finite Element Model," Report No. SAD-082, Sargent & Lundy Engineers (1973).

2. P. Schnabel, H. B. Seed, and J. Lysmer, "Modification of Seismograph Records for Effects of Local Soil Conditions," Bulletin of the Seismological Society of America, Vol. 62, No. 6, pp. 1649-1664 (1972).

3. N. M. Newmark, J. A. Blume, and K. K. Kapur, "Design Response Spectra for Nuclear Power Plants," Journal of the Power Division, American Society of Civil Engineers, pp. 287-303, November 1973.

4. S. L. Chu, M. Amin, and S. Singh, "Spectral Treatment of Actions of Three Earthquake Components on Structures," Nuclear Engineering and Design, Vol. 21, pp. 126-136 (1972).

5. N. M. Newmark and E. Rosenblueth, "Fundamentals of Earthquake Engineering," Prentice Hall, pp. 236-237 (1971).

6. H. B. Seed, K. L. Lee, I. M. Idriss, and F. Mahdisi, "Analysis of the Slides in the San Fernando Dams During the Earthquake of Feb. 9, 1971," Report, Earthquake Engineering Research Center, Univ. of Calif., Berkeley, March 1973.

7. H. B. Seed, K. L. Lee, and I. M. Idriss, "An Analysis of the Sheffield Dam Failure," Journal of the Soil Mechanics and Foundation Division, American Society of Civil Engineers, Vol. 95, No. SM 6 (1969).

8. R. V. Whitman, "Soil Structure Interaction," in "Seismic Design for Nuclear Power Plants," The MIT Press, Cambridge (1970).

9. P. Koss, "Element Associated Damping by Modal Synthesis," in "Proceedings of the Water Reactor Safety Conference, Salt Lake City, March 1973," National Technical Information Service, U. S. Dept. of Commerce.

10. N. C. Tsai, "Modal Damping for Soil-Structure Interaction," Journal of Engineering Mechanics Division, American Society of Civil Engineers, April 1974.

11. H. B. Seed, J. Lysmer, and R. Hwang, "Soil-Structure Interaction Analyses for Evaluating Seismic Response," presented at the Specialty Conference on Structural Design of Nuclear Plant Facilities, Dec. 1973.

12. N. C. Tsai, "Transformation of Time Axes of Acceleragrams," Journal of Engineering Mechanics Division, American Society of Civil Engineers, Vol. 95, No. EM 3, pp. 807-812, (1969).

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SECTION 3.7.3 SEISMIC SUBSYSTEM ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic subsystem analysis are reviewed:

1. Seismic Analysis Methods

   The information reviewed is similar to that described in Section I.1 of Standard Review Plan (SRP) 3.7.2, but as applied to seismic Category I subsystems.

2. Determination of Number of Earthquake Cycles

   Criteria or procedures used to establish the number of earthquake cycles during one seismic event and the maximum number of cycles for which applicable Category I subsystems and components are designed are reviewed.

3. Procedures Used for Analytical Modeling

   The criteria and procedures used for modeling for the seismic subsystem analysis are reviewed.

4. Basis for Selection of Frequencies

   As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed.

5. Use of Equivalent Static Load Method of Analysis

   The basis for the use of the equivalent static load method of analysis and the procedures used for determining the equivalent static loads are reviewed.

6. Three Components of Earthquake Motion

   The information reviewed is similar to that described in Section I.6 of SRP 3.7.2, but as applied to Category I subsystems.

7. Combination of Model Responses

   The information reviewed is similar to that described in Section I.7 of SRP 3.7.2, but as applied to Category I subsystems.

8. Analytical Procedures for Piping Systems
The analytical procedures applicable to seismic analysis of piping systems, including methods used to consider differential piping support movements at different support points located within a structure and between structures, are reviewed.

9. Multiply-Supported Equipment and Components with Distinct Inputs
The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are reviewed.

10. Use of Constant Vertical Static Factors
The information reviewed is similar to that described in Section I.10 of SRP 3.7.2, but as applied to Category I subsystems.

11. Torsional Effects of Eccentric Masses
The criteria and procedures that are used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic subsystem analyses are reviewed.

12. Category I Buried Piping Systems and Tunnels
For Category I buried piping and tunnels, the seismic criteria and methods which consider the compliance of soil media, settlement due to earthquake, and differential movements at support points, penetrations, and entry points into other structures provided with anchors are reviewed.

13. Interaction of Other Piping With Category I Piping
The seismic analysis procedures to account for the seismic motion of non-Category I piping systems in the seismic design of Category I piping are reviewed.

14. Seismic Analyses for Reactor Internals
The seismic subsystem analyses that are utilized in establishing seismic design adequacy of the reactor internals including fuel elements, control rod assemblies, and control rod drive mechanisms are reviewed. The information reviewed includes the following:

a. Typical diagrams of mathematical dynamic modeling of reactor internal structures.

b. Damping values and their justification.

c. A description of the methods and procedures that will be used to compute seismic responses.

d. For the operating license review, a summary of the results of the dynamic seismic analysis.

15. Analysis Procedure for Damping

The information reviewed is similar to that described in Section I.15 of SRP 3.7.2, but as applied to Category I subsystems.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Other approaches which can be justified to be equivalent to or more conservative than the stated acceptance criteria may be used.

1. Seismic Analysis Methods

The acceptance criteria provided in SRP 3.7.2, Section II.1, are applicable.

2. Determination of Number of Earthquake Cycles

During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) should be assumed. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed.

3. Procedures Used for Analytical Modeling

The acceptance criteria provided in SRP 3.7.2, Section II.3, are applicable.

4. Basis for Selection of Frequencies

To avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than 1/2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.

5. Use of Equivalent Static Load Method of Analysis

The acceptance criteria provided in SRP 3.7.2, Section II.1, are applicable.

6. Three Components of Earthquake Motion

The acceptance criteria provided in SRP 3.7.2, Section II.6, are applicable.

7. Combination of Modal Responses

The acceptance criteria provided in SRP 3.7.2, Section II.7, are applicable.

8. Analytical Procedures for Piping Systems

The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are as given in SRP 3.7.2, Section II.1.

9. Multiply-Supported Equipment and Components With Distinct Inputs

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, by using the floor response spectra. For the latter option, the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration in "g's" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra involves determination of dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors and applied identically to all the floors, provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination using static analysis procedures.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the subsystems (Ref. 3). Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems.

10. <u>Use of Constant Vertical Static Factors</u>
The acceptance criteria provided in SRP 3.7.2, Section II.10, are applicable.

11. <u>Torsional Effects of Eccentric Masses</u>
For seismic Category I subsystems, if the torsional effect of an eccentric mass such as a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined on a case-by-case basis.

12. <u>Category I Buried Piping Systems and Tunnels</u>
For Category I buried piping systems and tunnels the following items should be considered in the analysis:

a. The inertial effects due to an earthquake upon buried piping systems and tunnels should be adequately accounted for in the analysis. Use of the procedures described in References 1 and 2 is acceptable.

b. The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in Reference 4 is acceptable.

c. When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis.

13. Interaction of Other Piping with Category I Piping

To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I piping system. If it is not feasible or practical to isolate the Category I piping system, adjacent non-Category I piping should be analyzed according to the same seismic criteria as applicable to the Category I piping system. For non-Category I piping systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the Category I piping.

14. Seismic Analyses for Reactor Internals

To be acceptable, the seismic responses of the reactor pressure vessel and internals must be determined by a dynamic analysis. The analysis should comply with the applicable acceptance criteria provided in Sections II.1 and II.6 of SRP 3.7.2. In addition, the effects of piping-vessel interactions, externally applied structural restraints, hydrodynamic masses, etc., should be considered in the analysis.

15. Analysis Procedure for Damping

The acceptance criteria provided in SRP 3.7.2, Section II.15, are applicable.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Seismic Analysis Methods

The seismic analysis methods are reviewed to determine that these are in accordance with the acceptance criteria of SRP 3.7.2, Section II.1.

2. Determination of Number of Earthquake Cycles

Criteria or procedures used to establish the number of earthquake cycles are reviewed to determine that they are in accordance with the acceptance criteria as given in Section II.2. Justification for deviating from the acceptance criteria is requested from the applicant, as necessary.

3. Procedures Used for Analytical Modeling

The criteria and procedures used for modeling for the seismic subsystem analysis are reviewed to determine that these are in accordance with the acceptance criteria of SRP 3.7.2, Section II.3.

4. Basis for Selection of Frequencies

As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed to determine compliance with the acceptance criteria of Section II.4.

5. Use of Equivalent Static Load Method of Analysis

The criteria for the use of the equivalent static load method of analysis are reviewed to determine that these are in accordance with the acceptance criteria of Section II.5.

6. Three Components of Earthquake Motion

The procedures by which the three components of earthquake motion are considered in determining the seismic response of subsystems are reviewed to determine compliance with the acceptance criteria of SRP 3.7.2, Section II.6.

7. Combination of Modal Responses

The procedures for combining modal responses are reviewed to determine compliance with the acceptance criteria of SRP 3.7.2, Section II.7, when a response spectrum modal analysis method is used.

8. Analytical Procedures for Piping Systems

For all Category I piping and applicable non-Category I piping, the methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to determine that the techniques employed are in accordance with the acceptance criteria of Section II.8. Typical mathematical models are reviewed to judge whether all significant degrees of freedom have been included.

9. Multiply-Supported Equipment and Components With Distinct Inputs

The criteria for the seismic analysis of multiply-supported components and equipment with distinct inputs are reviewed to determine that the criteria are in accordance with the acceptance criteria of Section II.9.

10. Use of Constant Vertical Static Factors

Use of constant static factors as response loads in the vertical direction for the seismic design of any Category I subsystems in lieu of a detailed dynamic method is reviewed to determine that constant static factors are used only if the structure is rigid in the vertical direction.

11. Torsional Effects of Eccentric Masses

The procedures for seismic analysis of Category I piping subsystems are reviewed to determine compliance with the acceptance criteria of Section II.11.

12. Category I Buried Piping Systems and Tunnels

The analysis procedures for Category I buried piping and tunnels are reviewed to determine that they are in accordance with the acceptance criteria of Section II.12. This includes review of the procedures used to consider the inertial effects of soil media and the differential displacements at structural penetrations, etc. Any procedures that are not adequately justified are so identified and the applicant is requested to provide additional justification.

13. Interaction of Other Piping with Category I Piping

The criteria used to design the interfaces between Category I and non-Category I piping are reviewed to determine compliance with the acceptance criteria of Section II.13.

14. Seismic Analyses for Reactor Internals

The applicable methods of seismic analysis for reactor internals are reviewed to determine that the techniques employed are in accordance with the acceptance criteria of Section II.14. Typical mathematical models are reviewed to judge whether the characteristics of the reactor pressure vessel and internals are properly modeled and that all significant degrees of freedom have been incorporated, including any hydrodynamic effects. The number of modes used in the analysis are reviewed to assure that all significant modes have been included in the analysis.

15. Analysis Procedure for Damping

The analysis procedure to account for damping in different elements of the model of a coupled system is reviewed to determine that it is in accordance with the acceptance criteria of SRP 3.7.2, Section II.15.

IV. EVALUATION FINDINGS

Evaluation findings for SRP 3.7.3 have been combined with those of SRP 3.7.2 and are given under SRP 3.7.2, Section IV.

V. REFERENCES

1. N. M. Newmark, and E. Rosenblueth, "Fundamentals of Earthquake Engineering," Prentice Hall (1971).

2. N. M. Newmark, "Earthquake Response Analysis of Reactor Structures," Nuclear Engineering and Design, Vol. 20, pp. 303-322 (1972).

3. R. P. Kassawara, and D. A. Peck, "Dynamic Analysis of Structural Systems Excited at Multiple Support Locations," 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Chicago, Dec. 17-18, 1973.

4. M. Hetenyi, "Beams on Elastic Foundation," The University of Michigan Press (1946).

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SECTION 3.7.4 SEISMIC INSTRUMENTATION

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas related to the seismic instrumentation program are reviewed:

1. Comparison with Regulatory Guide 1.12

A comparison of the proposed seismic instrumentation with the seismic instrumentation guidelines of Regulatory Guide 1.12 (Ref. 2) is made. In addition, the bases for elements of the program that differ from Regulatory Guide 1.12 are reviewed.

2. Location and Description of Instrumentation

The locations for the installation of seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected Category I structures and components are reviewed. The bases for selection of the instrumentation and the locations and a discussion of the extent to which the seismic instrumentation will be employed to verify the seismic analyses following an earthquake are reviewed.

3. Control Room Operator Notification

The procedures that will be followed to inform the control room operator of the peak acceleration level and the input response spectra values shortly after occurrence of an earthquake are reviewed. Also reviewed are the bases for establishing predetermined values for activating the readout of the seismic instrumentation to the control room operator.

4. Comparison of Measured and Predicted Responses

The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are given below. Any other seismic instrumentation program which can be justified to be equivalent to the acceptance criteria may be used.

1. Comparison with Regulatory Guide 1.12

   The seismic instrumentation program is considered to be acceptable if it is in accordance with Regulatory Guide 1.12 (see also Table 3.7.4-1). This guide recommends provision of a triaxial time history accelerograph and a triaxial response spectrum recorder to measure the input time history and response spectra directly. Additional time history accelerographs, response spectrum recorders, peak accelerographs, and seismic switches are recommended to measure the responses of structures, equipment, and components at selected locations. The bases for elements of the proposed seismic instrumentation program that differ from Regulatory Guide 1.12 must be provided.

2. Location and Description of Instrumentation

   For the construction permit review there should be a commitment by the applicant to provide the following instruments at the given locations:

   a. A triaxial time history accelerograph in the free field or at the containment foundation, with readout in the control room.

   b. A seismic switch on the containment foundation, with readout in the control room.

   c. A triaxial response spectrum recorder on the containment foundation, with readout in the control room.

   In addition, a commitment to provide the recommended additional instrumentation at the various response locations should be made without providing details of actual locations.

   For the operating license review, a detailed seismic instrumentation plan including locations and descriptions of the remaining instrumentation should be provided. To be acceptable, the remaining instrumentation locations are related to the locations of the output vibratory motions used in the seismic design. Typical general locations are:

   a. Containment structure or reactor building.

   b. Reactor piping.

   c. Reactor equipment.

   d. Other category I structures, equipment, and piping.

Instrumentation should be provided depending upon the plant safe shutdown earthquake acceleration as given in Regulatory Guide 1.12. The specific locations are determined by the plant designer so as to obtain the most pertinent information (Ref. 3). A possible approach to the specification of the seismic instrumentation system is given in Regulatory Guide 1.12. Other desirable combinations of instruments which may prove to be as useful as the instrumentation plan outlined in the guide may be utilized.

The criteria for selection of Category I structures, components, and equipment to be instrumented and the location of instrumentation, as well as the extent to which this instrumentation is employed to verify the seismic analyses following an earthquake, should be specified. The criteria will be reviewed on a case-by-case basis.

3. Control Room Operator Notification

To be acceptable, the seismic switch located at the foundation of the containment should be connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level (OBE acceleration level) resulting from the earthquake is exceeded. Also both audio and visual signals should be provided to the control room operators in the event of an earthquake.

In addition, the triaxial time history accelerograph located in the containment foundation or in the free field should be connected to the control room, so that peak acceleration level experienced in the basement of the reactor containment structure or in the free field is indicated to the control room operator. The response spectrum recorder in the reactor containment foundation or in the free field is also connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake.

4. Comparison of Measured and Predicted Responses

In the event of an earthquake, the control room operator should be immediately informed through the event indicators. If the instrumentation shows that the peak acceleration or the response spectra experienced at the foundation of the containment building or in the free field exceed the OBE acceleration level or response spectra, the plant should be shut down (Ref. 1) pending permission to resume operations. To help predict the capability of the plant for resuming operations, field inspection of safety-related items should be implemented and the measured responses from both the peak-recording and strong motion accelerographs should be compared with those assumed in the design.

The procedures for comparison of measured and predicted responses are acceptable if a commitment is made to provide detailed comparisons, as outlined below, between measured seismic responses of Category I structures and equipment with calculated responses determined from dynamic analysis. First, the time history records are digitized and corrected for time signal variations and baseline variations. The time history records from the triaxial sensors located in the free field or at the foundation of the containment building are used to calculate response spectra at appropriate critical damping

TABLE 3.7.4-1 SEISMIC INSTRUMENTATION REQUIREMENTS

Instrumentation		Triaxial Time-History Accelerograph		Triaxial Response Spectrum Recorder		Triaxial Peak Accelerograph		Seismic Switch	
Location	SSE	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g	0.3 g or less	over 0.3 g
I. Free Field		1*#	1*#						
II. Inside Containment									
Basement		1*	1*	1*	1*			1*	1*
At Elevation		1	1						
Reactor Equip. Sup.				}1	}1				}1*
Reactor Piping Sup.									
Reactor Equipment						1	1		
Reactor Piping						1	1		
III. Outside Containment									
Cat. I Structure			1	1	1				
Cat. I Equip. Sup.					1				
Cat. I Piping Sup.				}1	1				
Cat. I Equipment							1		
Cat. I Piping						}1	1		

*Control room readout.
#May be omitted if soil-structure interaction is negligible.
}Denotes either of the two locations.

values. The response spectra thus obtained, or the response spectra from the response spectrum recorder, are compared with the design response spectra. In addition, the time history records from the free field triaxial sensor are used as input ground motion for the reactor building dynamic model, including soil where applicable. Amplified response spectra are then calculated at the locations of the other sensors in the reactor building for comparison and correlation with the response spectra directly measured. Structural responses and amplified response spectra are calculated using the free field time history records with the dynamic model for comparison with the original design and analysis parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for remodeling, detailed analyses, and physical inspection.

III. REVIEW PROCEDURES

For each area of review, the following review procedure is followed. The reviewer will select and emphasize material from the procedures given below, as may be appropriate for a particular case.

1. Comparison with Regulatory Guide 1.12

   The seismic instrumentation program is checked to assure that the instrumentation is in accordance with the guidelines of Regulatory Guide 1.12. Any differences between the proposed and the guide seismic instrumentation, which have not been adequately justified, are identified and the applicant is informed of the need for additional technical justification.

2. Location and Description of Instrumentation

   At the operating license stage, the locations and descriptions of the seismic instrumentation are reviewed to determine that these are in accordance with the acceptance criteria of Section II.2. If the instrumentation provided is judged to be insufficient, the need for additional instrumentation is transmitted to the applicant.

3. Control Room Operator Notification

   The seismic instrumentation is checked to verify that the seismic switch located at the foundation of the containment structure or in the free field is connected to event indicators that are located in the control room, so that a signal is given when the preset threshold level is exceeded. If there is no provision for both audio and visual signals in the applicant's seismic instrumentation plan, the applicant is so informed with a request for compliance.

4. Comparison of Measured and Predicted Responses

   The criteria and procedures that will be used to compare measured responses of Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses are checked to verify that sufficient information as specified in Section II.4 is included. Any deficiency in the required information is identified and the applicant is requested to provide further information.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The installation of the specified seismic instrumentation in the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12."

V. REFERENCES

1. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1.

3. K. Kapur, "Seismic Instrumentation for Nuclear Power Plants," in "Proceedings of the Topical Meeting on Water Reactor Safety, Salt Lake City, March 1973," CONF-730304, U. S. Atomic Energy Commission Technical Information Center (1973).

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.1 CONCRETE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to concrete containments or to concrete portions of steel/concrete containments, as applicable, are reviewed.

1. Description of the Containment

The descriptive information, including plans and sections of the structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. In particular, the type of concrete containment is identified and its structural and functional characteristics are examined. Among the various types of concrete containments reviewed are:

a. Reinforced and prestressed concrete BWR containments utilizing the pressure-suppression concept, including the Mark I (modified lightbulb/torus), the Mark II (over/under) and the Mark III (with horizontal venting between a centrally-located cylindrical drywell and a surrounding suppression pool).
b. Reinforced concrete PWR containments utilizing the pressure-suppression concept with ice-condenser elements.
c. Reinforced concrete PWR containments designed to function under sub-atmospheric conditions.
d. Reinforced and prestressed concrete PWR dry containments designed to function at atmospheric conditions.
e. Reinforced and prestressed concrete PWR or BWR containments utilizing special features or modifications of the above-listed types.

Various geometries have been utilized for these containments. The geometry most commonly encountered is an upright cylinder topped with a dome and supported on a flat concrete base mat. Although applicable to any geometry, the specific provisions of this review plan are best suited to the cylindrical type containment topped by a dome. If containments with other types of geometry are reviewed, the necessary modifications to this plan are made on a case-by-case basis.

The geometry of the containment is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The arrangement of the containment and the relationship and interaction of the shell with its surrounding structures and with its interior compartment walls and floors are reviewed to determine the effect which these structures could have upon the design boundary conditions and expected structural behavior of the containment when subjected to design loads.

General information related to the containment shell is reviewed including the following:

a. The base foundation slab, including the main reinforcement; the floor liner plate and its anchorage and stiffening system; the methods by which the interior structures are anchored through the liner plate and into the slab, if applicable.
b. The cylindrical wall, including the main reinforcement and prestressing tendons, if any; the wall liner plate and its anchorage and stiffening system; the major penetrations and the reinforcement surrounding them including the equipment and personnel hatches and major pipe penetrations; major structural attachments to the wall which penetrate the liner plate such as beam seats, pipe restraints and crane brackets; and external supports, if any, attached to the wall to support external structures such as enclosure buildings.
c. The dome and the ring girder, if any, including the main reinforcement and prestressing tendons; the liner plate and its anchorage and stiffening systems; and any major attachments to the liner plate made from the inside.
d. Steel components of concrete containments that resist pressure and are not backed by structural concrete are covered by Standard Review Plan 3.8.2.

2. Applicable Codes, Standards, and Specifications

Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design fabrication, construction, testing, and in-service surveillance of the containment, is reviewed. The specific edition, date, or addenda identified for each document are reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various combinations thereof is reviewed with emphasis on the extent of compliance with Article CC-3000 of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359), (Ref. 1), (hereafter "the Code"). The loads normally applicable to concrete containments include the following:

a. Those loads encountered during preoperational testing.
b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those present in pressure-suppression containments utilizing water.

c. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating basis earthquake specified for the plant site.
d. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the safe shutdown earthquake specified for the plant site.
e. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCA). The main abnormal plant condition for containment design is the design basis LOCA. Also considered are other accidents involving various high-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possibly localized loads such as jet impingement and associated missile impact.
f. Those loads to be sustained, if applicable, after abnormal plant conditions including flooding of the containment subsequent to a LOCA for fuel recovery.

The various combinations of the above loads that are normally postulated and reviewed include the following:

Testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; normal operating loads with extreme environmental and abnormal loads; and post-LOCA flooding loads with severe environmental loads, if applicable.

The loads and load combinations described above are generally applicable to all containments. However, other site-related design loads might be applicable also. Such loads, which are not normally combined with abnormal loads, are reviewed on a case-by-case basis. They include those loads induced by floods, potential aircraft crashes, explosive hazards in proximity to the site and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures

The design and analysis procedures utilized for the containment are reviewed with emphasis on the extent of compliance with Article CC-3000 of the Code, particularly with respect to the following:

a. Assumptions on boundary conditions.
b. Treatment of axisymmetric and non-axisymmetric loads.
c. Treatment of transient and localized loads.
d. Treatment of the effects of creep, shrinkage, and cracking of the concrete.
e. A description of the computer programs utilized in the design and analyses.
f. The treatment of the effects of seismically-induced tangential (membrane) shears.
g. The evaluation of the effects of variations in specified physical properties of materials on analytical results.
h. The treatment of the large, thickened penetration regions.

i. The treatment of the steel liner plate and its anchors. Steel penetration closures are covered by Standard Review Plan 3.8.2.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment are reviewed, with emphasis on the extent of compliance with Article CC-3000 of the Code, specifically with respect to allowable stresses, strains, gross deformations, and other parameters that identify quantitatively the margins of safety. For each load combination specified, the proposed allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan. Included in these allowable limits are the following major parameters:

a. Compressive stresses in concrete, including membrane, membrane plus bending, and localized stresses.
b. Shear stresses in concrete, particularly those tangential (membrane) stresses induced by lateral loads.
c. Tensile stresses in reinforcement.
d. Tensile stresses in prestressing tendons.
e. Tensile or compressive stress/strain limits in the liner plate, including membrane and membrane plus bending.
f. Force/displacement limits in the liner plate anchors, including those induced by strains in the adjacent concrete.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on materials that are used in construction of the containment is reviewed with emphasis on the extent of compliance with Article CC-2000 of the Code. Among the major materials of construction that are reviewed are the following:

a. The concrete ingredients.
b. The reinforcing bars and splices.
c. The prestressing system.
d. The liner plate.
e. The liner plate anchors and associated hardware.
f. The structural steel used for embedments such as beam seats and crane brackets.
g. The corrosion-retarding compounds used for the prestressing tendons.

The quality control program that is proposed for the fabrication and construction of the containment is reviewed with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the Code, including the following:

Examination of the materials including tests to determine the physical properties of concrete, reinforcing steel, mechanical splices, the liner plate and its anchors, and the prestressing system, if any; placement of concrete; and erection tolerances of the liner plate, reinforcement, and prestressing system.

Special, new or unique construction techniques, if proposed, such as slip forming, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and In-service Surveillance Requirements

The preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, is reviewed including the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with Article CC-3000 of the Code. In-service surveillance programs such as the periodic surveillance and inspection of the prestressing tendons, if any, are also reviewed, including the applicable Technical Specifications, at the operating license stage. Special testing and in-service surveillance requirements proposed for new or previously untried design approaches are also reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The Regulatory acceptance criteria for the areas of review are as follows:

1. Description of the Containment

The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Section 3.8.1.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 2). If the concrete containment has new or unique features that are not specifically covered in the "Standard Format...," the reviewer determines that the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented, as appropriate.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and in-service surveillance of concrete containments are covered by codes, standards, specifications, and guides that are either applicable in their entirety or in part. The following codes and guides are acceptable.

Code	Title
ACI/ASME (ACI-359)*	Proposed Standard Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2, Issued for Interim Trial-Use and Comment, April 1973

Regulatory Guides	Title
1.10	Mechanical (Cadweld) Splices for Reinforcing Bars of Category I Concrete Structures
1.15	Testing of Reinforcing Bars for Category I Concrete Structures
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments
1.19	Nondestructive Examination of Primary Containment Liner Welds
1.35	Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures
1.55	Concrete Placement in Category I Structures

*Issued as an interim code for trial use and comments and subject to revisions prior to publication as a mandatory code. Any proposed use of amendments to the April version of the interim code will be reviewed on a case-by-case basis.

3. Loads and Loading Combinations

The specified loads and load combinations are acceptable if found to be in accordance with Article CC-3000 of the Code with the exceptions listed below taken to the requirements specified in Table CC-3200-1.

a. Y_j, jet impingement loads, and Y_m, impact loads of missiles associated with accidents, should be included.

b. The 6th combination, representing "abnormal" load conditions, need not include Y_r in combination with 1.5P.

c. In the 7th, 8th,and 9th combinations, representing "abnormal/severe environmental" and abnormal/extreme environmental" load conditions, the "and/or" between R_a and Y_r should be deleted and, in addition to R_a and Y_r, the combinations should include Y_j and Y_m.

d. The maximum values of P_a, T_a, R_a, Y_r, Y_j,and Y_m should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise.

Where post-LOCA flooding is a design consideration, the following combination should also be considered in the factored load category:

1.0 D + 1.0 L + 1.0 F + 1.0 Feqo, where D, L, Feqo are as defined in the Code and F is the load generated by the post-LOCA flooding of the containment.

4. Design and Analysis Procedures

The procedures of design and analysis utilizes for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in Article CC-3300 of the Code. In particular, for the areas of review outlined in Section I.4 of this plan, the following procedures are, in general, acceptable:

a. Assumptions on boundary conditions

The boundary conditions depend on the methods of analysis to be used and the portions of the containment shell to be separately analyzed. If the analysis is to be accomplished through the use of the finite element technique, and is to include the foundation media, the boundary would be the demarcation lines separating the foundation mass taken into consideration in the analysis from the surrounding media. The boundaries of the foundation mass considered have to be so selected that any further extension of the boundaries will not affect the results by more than 15 percent.

If only the containment shell and its foundation mat are taken into consideration in the analysis, then the bottom of the foundation slab is the boundary of the analytical model. The foundation media should be represented by appropriate soil springs.

If separate analyses of the containment shell and the base mat are to be used, it is considered acceptable if strain compatibility of the bottom portion of the shell with the base mat is maintained.

b. Axisymmetric and non-axisymmetric loads

Even with the large penetrations and buttresses that may be utilized in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable if such an assumption is made with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the non-axisymmetric effect of these loads should be considered.

c. Transient and localized loads

During normal operation, a linear temperature gradient across the containment wall thickness may develop. After the loss-of-coolant accident (LOCA), however, the sudden increase in temperature in the steel liner and the adjacent concrete may produce a non-linear transient temperature gradient across the containment wall thickness. Effects of such transient loads should be considered.

In a PWR ice-condenser containment, non-axisymmetric and transient pressure loads resulting from compartmentation inside the containment will develop after a LOCA.

For the effects of such localized and transient loads, the overall behavior of the containment structure should first be determined. A portion of the containment shell, within which the localized or transient load is located, should then be analyzed, using the results obtained from the analysis of the overall vessel behavior as boundary conditions.

d. Creep, shrinkage, and cracking of concrete

Creep and shrinkage values for concrete should be established by tests performed on the concrete which is to be used in the containment structure, or from data obtained on completed containments constructed of the same kind of concrete. In establishing these values, consideration should be given to the differences in the environment between the test samples and the actual concrete in the structure. Cracking of the concrete may be considered in either of the following two ways: (i) the moments, forces, and shears under load may be obtained on the basis of an uncracked section for all loading combinations. In sizing the reinforcing steel required, however, the concrete shall not be relied upon for resisting tension. Thermal moments may be modified to take creep and cracking into consideration. (ii) For axisymmetrical loadings, cracking of the concrete may be considered through the use of computer programs which are capable of treating such cracking by an iterative process. However, for non-axisymmetric loadings, most of the computer programs available do not have the capability of considering cracking, since the structure itself becomes non-axisymmetric when concrete cracking is to be considered iteratively. Accordingly, if the concrete is cracked under any load combination involving axisymmetric and non-axisymmetric loadings, a method should be described for considering cracking. Such methods are reviewed on a case-by-case basis.

e. Computer programs

The computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria:

(i) The computer program is a recognized program in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

(ii) The computer program solution to a series of test problems has been demonstrated to be substantially identical to those obtained by a similar and independently-written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program.

(iii) The computer program solution to a series of test problems has been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests, or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

A summary comparison should be provided for the results obtained in the validation of each computer program.

f. Tangential shear

Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in Article CC-3000 of the Code. The exceptions taken by the Regulatory staff to the provisions of this Article, as contained in Section II.5 of this plan, are to be noted.

g. Variation in physical material properties

For considering the effects of possible variations in the physical properties of materials on the analytical results, the upper and lower bounds of these properties should be used in the analysis, wherever critical. Among the physical properties that may be critical include the soil modulus, and modulus of elasticity and Poisson's ratio of concrete.

h. Thickened penetrations

The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated in the same manner as for localized loads discussed in item (c).

i. Steel liner plate and anchors

For the design and analysis of the liner plate and its anchorage system, the procedures furnished are found adequate and acceptable if in accordance with the provisions of Subarticle CC-3600 of the Code. In general, the liner plate analysis should consider deviations in geometry due to fabrication and erection tolerances, and variations of the assumed physical properties of the liner and anchor material. Since the liner plate is usually anchored at relatively closely spaced intervals, the analysis procedures are acceptable if based on either the classical plate or beam theory. Since the concrete shell is much stiffer than the liner plate, the strains in the liner will essentially follow those in the concrete. The strains in the concrete under the various load combinations as obtainable from the analysis of the shell are thus imposed on the liner plate and the resulting strains and stresses in the liner and its anchors should be lower than the allowable limits defined in Tables CC-3700-1 and CC-3700-2 of the Code.

5. Structural Acceptance Criteria

a. For the structural portions of the containment, the specified allowable limits for stresses and strains are acceptable if they are in accordance with Subsection CC-3400 of the Code but with the following exceptions:

CC-3411.5

- Under no conditions shall the tangential shear stress carried by the concrete, v_c, exceed 40 psi and 60 psi for the 7th and 9th combinations of Table CC-3200-1, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively.
- For prestressed concrete, the principal tensile stress shall not exceed $4\sqrt{f'c}$.

C-3421.1

- The footnote on page 196 indicates that the 33-1/3% increase in allowable stresses is permitted only temperature loads and not for seismic or wind loads.

CC-3422.1

- Item (c) should be deleted.

CC-3422.2

- The footnote on page 197 should be deleted.

b. For the liner plate and its anchorage system, the specified limits for stresses and strains are acceptable if in accordance with Tables CC-3700-1 and CC-3700-2 of the Code, respectively.

6. Materials, Quality Control, and Special Construction Techniques

a. The specified materials of construction are acceptable if in accordance with Article CC-2000 of the Code with the following exceptions:

(i) CC-2243.3 permits the chloride and nitrate content of constituents of cement grout for prestressing tendons to reach maximum limits of 300 ppm and 100 ppm, respectively. It is understood that these are total allowable limits in the grout, not just for the mixing water, and ACI 318 (commentary 3.4.1) is the apparent source of the 300 ppm chloride limit. However, in view of the cautions approach taken in the wording of 3.4.1 and 3.6.1 of the ACI 318-71 commentary, the potential sensitivity of the prestressed steels to chloride ions, and since local municipal treated potable water contains in the order of only 12 ppm each of chlorides and nitrates, reconsideration of the 300 ppm and 100 ppm limits are recommended. Consideration should be given to the possibility of chloride and nitrate content of all the grout constituents should be individually determined and the possible concentration levels evaluated.

(ii) In Tables I-1.1 and I-1.2, the inclusion of deformed bars as acceptable materials for prestressing systems should be deleted, since neither ASTM acceptance nor sufficient user justification data have been secured, as per the requirements of the Code.

b. Quality control programs are acceptable if in accordance with applicable portions of Articles CC-4000 and CC-5000 of the Code as augmented by Regulatory Guides 1.10 for Cadweld reinforcement splicing (Ref. 3), 1.15 for testing of reinforcing bars (Ref. 4), 1.19 for the nondestructive examination of the liner plate welds (Ref. 5), and 1.55 for concrete placement (Ref. 6).

c. Special construction techniques, if any, are reviewed on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

a. Procedures for the post-construction preoperational structural proof test proposed for the containment are acceptable if found in accordance with those delineated in Article CC-6000 of the Code as augmented by the provisions delineated in Regulatory Guide 1.18 (Ref. 7).

b. For prestressed concrete containments, in-service surveillance requirements for the tendons, as presented in the Technical Specifications of the Operating License, are acceptable if in accordance with Regulatory Guides 1.35 for ungrouted tendons (Ref. 8) and 1.___ for grouted tendons (Ref. 9), respectively.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Description of the Containment

After the type of containment and its functional characteristics are identified, information on similar and previously licensed applications is obtained for reference. Such information, which is available in safety analysis reports and amendments of previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided in the SAR. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications are checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and stated effective addenda are utilized.

3. Loads and Loading Combinations

The reviewer verifies that the loads and load combinations, as described by the applicant, are as conservative as those referenced in Section II.3 of this plan. Loading conditions that are unique to the site, such as potential aircraft crashes, and that are not specifically covered in Section II.3, are treated on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and this information is transmitted to the applicant for further consideration.

4. Design and Analysis Procedures

The reviewer assures himself that the applicant has committed to utilize design and analysis procedures delineated in Article CC-3000 of the Code. Any exceptions to these procedures are reviewed and evaluated on a case-by-case basis. In particular, the areas of review contained in Section I.4 of this plan are evaluated for conformance with the acceptance criteria.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, liner plate and its anchors, and in components of the prestressing system, if any, are reviewed and compared with the acceptable limits referenced in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification provided to show that the structural integrity of the containment will not be affected is evaluated. If such justification is unacceptable, the applicant is required to submit additional justification or otherwise comply with the acceptance criteria delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that referenced in Section II.6 of this plan. If a material not used in previously licensed applications is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of the material. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the containment, the liner plate, and its anchorage system.

7. Testing and In-service Surveillance Requirements

The initial structural overpressure test program is reviewed and compared with that indicated as acceptable in Section II.7 of this plan. Proposed deviations are considered on a case-by-case basis. In-service surveillance programs, particularly for the prestressing tendons, if any, as presented in the Technical Specifications of the Operating License, are similarly reviewed.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's Safety Evaluation Report:

"The criteria used in the analysis, design, and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, guides, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function. Conformance with these criteria constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2, 4, 16, and 50."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard Code for Concrete Reactor Vessels and Containments," issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

3. Regulatory Guide 1.10, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures."

4. Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category I Concrete Structures."

5. Regulatory Guide 1.19, "Nondestructive Examination of Primary Containment Liner Welds."

6. Regulatory Guide 1.55, "Concrete Placement in Category I Structures."

7. Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments."

8. Regulatory Guide 1.35, "In-service Surveillance of Ungrouted Tendons in Prestressed Concrete Containments."

9. Regulatory Guide 1.__, "In-service Surveillance in Prestressed Concrete Containments with Grouted Tendons," (in preparation).

10. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

12. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

13. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.2 STEEL CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The following areas relating to steel containments or to other Class MC steel portions of steel/concrete containments, as applicable, are reviewed.

1. Description of the Containment

   The descriptive information, including plans and sections of the structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. In particular, the type of steel containment is identified and its structural and functional characteristics are examined. Among the various types of steel containments reviewed are:

   a. Steel BWR containments utilizing the pressure-suppression concept, including the Mark I (lightbulb/torus), the Mark II (over/under) and the Mark III (with horizontal venting between a centrally-located cylindrical drywell and a surrounding suppression pool).
   b. Steel PWR containments utilizing the pressure-suppression concept with ice-condenser elements.
   c. Steel PWR dry containments.

Various geometries have been utilized for these containments. The geometry most commonly encountered, however, is an upright cylinder topped with a dome and supported on either a flat concrete base mat covered with a liner plate, or on a concrete foundation built around the bottom portion of the steel shell, which is an inverted dome. Although applicable to any geometry, the specific provisions of this review plan are best suited to the cylindrical-type steel containment surrounded by a Category I concrete shield building. If containments with other types of geometry are reviewed, the necessary modifications to this plan are made on a case-by-case basis.

The geometry of the containment is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The arrangement of the containment and the relationship and interaction of the shell with its surrounding shield building and with its interior compartments, walls and floors, are reviewed to determine the effect which these structures could have upon the design boundary conditions and the expected behavior of the shell when subjected to the design loads.

General information related to the containment shell is reviewed including the following:

a. The foundation of the steel containment including the following:
   (i) If the bottom of the steel containment is continuous through an inverted dome, the method by which the inverted dome and its supports are anchored to the concrete foundation, which is covered by Standard Review Plan 3.8.5, is reviewed.
   (ii) If the bottom of the steel containment is not continuous, and where a concrete base slab topped with a liner plate is used for a foundation, the extent of descriptive information reviewed for the foundation is contained and is reviewed as stated in Section I.1 of Standard Review Plan 3.8.1. Further, the method of anchorage of the steel cylindrical shell walls in the concrete base slab is reviewed, particularly the connection between the floor liner plate and the steel shell.
b. The cylindrical portion of the shell is reviewed including major structural attachments such as beam seats, pipe restraints, crane brackets, and shell stiffeners, if any, in the hoop and vertical directions.
c. The dome of the steel containment including any reinforcement at the dome/cylinder junction, penetrations or attachments made on the inside such as supports for containment spray piping, and any stiffening of the dome.
d. Major penetrations or portions thereof, of steel or concrete containments, to the limits defined by Figure NE-1132.1 of Subsection NE of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Division 1 (Ref. 1), and portions of the penetrations that are intended to resist pressure but are not backed by structural concrete, including those of sleeved and unsleeved piping penetrations, mechanical systems penetrations such as fuel transfer tubes, electrical penetrations, and access openings such as the equipment hatch and personnel locks.

2. Applicable Codes, Standards, and Specifications

The information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are used in the design, fabrication, construction, testing, and in-service surveillance of the steel containment, is reviewed. The specific edition, date, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various load combinations is reviewed with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1, and with Regulatory Guide 1.57 (Ref. 2). The loads normally applicable to steel containments include the following:

a. Those loads encountered during preoperational testing.
b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those present in pressure-suppression containments utilizing water.
c. Those loads to be sustained during severe environmental conditions, including those induced by design wind (if not protected by a shield building) and the operating basis earthquake.
d. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado (if not protected by a shield building) and the safe shutdown earthquake specified for the plant site.
e. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCA). The main abnormal plant condition for containment design is the design basis LOCA. Also to be considered are other accidents involving various high-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possibly localized loads such as jet impingement and associated missile impact. Also included are external pressure loads generated by events inside or outside the containment.
f. Those loads to be sustained, if applicable, after abnormal plant conditions, including flooding of the containment subsequent to a LOCA for fuel recovery.

The various combinations of the above loads that are normally postulated and reviewed include the following: Testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with severe environmental loads and abnormal loads; normal operating loads with extreme environmental loads and abnormal loads; and post-LOCA flooding loads with severe environmental loads, if applicable. Specific and more detailed information on these combinations are delineated in Section II.3 of this plan.

Unless the steel containment is protected by a shield building, other site-related design loads might also be applicable, including those described in Section I.3 of Standard Review Plan 3.8.1.

4. <u>Design and Analysis Procedures</u>
The design and analysis procedures utilized for the steel containment are reviewed with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1. Particular emphasis is placed on the following subjects:
a. Treatment of non-axisymmetric and localized loads.
b. Treatment of local buckling effects.
c. The computer programs utilized in the design and analysis.

5. <u>Structural Acceptance Criteria</u>
The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment are reviewed, specifically with respect to allowable stresses, strains, and gross deformations, with emphasis on the extent of compliance with Subsection NE of the Code, Section III, Division 1, and with Regulatory

Guide 1.57. For each specified load combination, the proposed allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan. Included in these allowable limits are the following major parameters:

a. Primary stresses, including general membrane, local membrane, and bending plus local membrane stresses.
b. Primary and secondary stresses.
c. Peak stresses.
d. Buckling criteria.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on the materials that are to be used in the construction of the steel containment is reviewed with emphasis on the extent of compliance with Article NE-2000 of Subsection NE of the Code, Section III, Division 1. Among the major materials that are reviewed are the following:

a. Steel plates used as shell components.
b. Structural steel shapes used for stiffeners, beam seats, and crane brackets.

Corrosion and corrosion protection procedures are reviewed by the Materials Engineering Branch (MTEB).

The quality control program that is proposed for the fabrication and construction of the containment is reviewed with emphasis on the extent of compliance with Article NE-5000 of Subsection NE of the Code, Section III, Division 1, including the following:

a. Nondestructive examination of the materials, including tests to determine their physical properties.
b. Welding procedures.
c. Erection tolerances.

Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and In-service Surveillance Program

The preoperational structural test programs for the completed containment and for individual class MC components reviewed, including the objectives of the test, and the acceptance criteria with emphasis on the extent of compliance with Article NE-6000 of Subsection NE of the Code, Section III, Division 1. Structural tests for components such as personnel and equipment locks are also reviewed.

In-service surveillance programs, if any, of components relied upon for containment structural integrity, are reviewed. Any in-service surveillance required in special areas subject to corrosion is reviewed by the Materials Engineering Branch (MTEB).

Special testing and in-service surveillance requirements proposed for new or previously untried design approaches are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. Description of the Containment

   The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Section 3.8.2.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 3).

   If the steel containment has new or unique features that are not specifically covered in the "Standard Format...," the reviewer determines that the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented.

2. Applicable Codes, Standards, and Specifications

   The design, materials, fabrication, erection, inspection, testing, and in-service surveillance of steel containments are covered by codes, standards, and specifications which are either applicable in their entirety or in part. The following codes and guides are acceptable.

Code	Title
ASME	Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components"
Regulatory Guides	
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

3. Loads and Loading Combinations

   Subsection NE of the Code, Section III, Division 1 and Regulatory Guide 1.57 are not explicit with respect to the loads and load combinations which should be considered in the design of steel containments. The specified loads and load combinations are acceptable if found to be in accordance with the following:

   a. Loads

   D --- Dead loads.

   L --- Live loads.

   P_t --- Test pressure.

   T_t --- Test temperature.

   T_o --- Thermal effects and loads during startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

   R_o --- Pipe reactions during startup, normal operating or shutdown conditions, based on the most critical transient or steady state-condition.

   E --- Loads generated by the operating basis earthquake.

   E' --- Loads generated by the safe shutdown earthquake.

   P_e --- Design external pressure.

   T_e --- Thermal loads under thermal conditions during event causing external pressure.

R_e --- Pipe reactions under thermal conditions during event causing external pressure.

P_a --- Pressure equivalent static load generated by the postulated design basis accident.

T_a --- Thermal loads under thermal conditions generated by the postulated design basis accident and including T_o.

R_a --- Pipe reactions under thermal conditions generated by the postulated design basis accident and including R_o.

Y_r --- Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident.

Y_j --- Jet impingement equivalent static load on the structure generated by the broken pipe during the design basis accident.

Y_m --- Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping.

F_L --- Loads generated by the post-LOCA flooding of the containment, if any.

b. <u>Loading Combinations</u>

(1) ---- $D + L + P_t + T_t$

(2) ---- $D + L + T_o + R_o$

(3) ---- $D + L + T_o + R_o + E$

(4) ---- $D + L + T_a + R_a + P_a + E$

(5) ---- $D + L + T_e + R_e + P_e + E$

(6) ---- $D + L + T_a + R_a + P_a + E'$

(7) ---- $D + L + T_e + R_e + P_e + E'$

(8) ---- $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

(9) ---- $D + L + F_L + E$

4. <u>Design and Analysis Procedures</u>

Design and analysis procedures for steel containments are covered in Article NE-3000 of Subsection NE of the Code, Section III, Division 1. The procedures given in the Code, as augmented by the applicable provisions of Regulatory Guide 1.57, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Section I.4 of this plan, the following criteria are acceptable:

a. <u>Treatment of non-axisymmetric and localized loads</u>

For most containments, the major non-axisymmetric loads which apply are the horizontal seismic loads. Other possible non-axisymmetric and localized loads are those induced by pipe rupture such as reactions, jet impingement forces, and missiles. For the PWR ice-condenser containment, the design basis accident may result in a non-axisymmetric pressure load due to compartmentalization of the containment interior. For such localized loads, the analyses should include a determination of the local effects of the loads. These effects should then be superimposed on the overall effects. For the overall effects of non-axisymmetric loads on shells of revolution, an acceptable general procedure is to expand the load by a Fourier series. Other methods are reviewed on a case-by-case basis for applicability to a large thin shell.

b. Treatment of local buckling effects

Localized pressure loads, such as those encountered in PWR ice-condenser containments, require consideration of local buckling of the shell. An acceptable approach to the problem is to perform a non-linear dynamic analysis. If a static analysis is performed, an appropriate dynamic load factor should be used to obtain the effective static load.

c. Computer programs

The computer programs used in the design and analysis should be described and validated by any of the procedures or criteria described in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

Stresses at various locations of the shell of the containment for various design loads are determined by analysis. Total stresses for the combination of loads delineated in Section II.3 of this plan are acceptable if found to be within limits defined by various sections of the Code, Section III, Subsection NE, as augmented by Regulatory Guide 1.57. An acceptable interpretation of these limits is contained in Table 3.8.2-1 where the notation is in accordance with the Code.

6. Materials, Quality Control, and Special Construction Techniques

a. The materials of construction are acceptable if in accordance with Article NE-2000 of Subsection NE of the Code, Section III, Division 1. Acceptance criteria for corrosion protection are established by the Materials Engineering Branch (MTEB).
b. Quality control programs are acceptable if in accordance with Articles NE-4000 and NE-5000 of Subsection NE of the Code, Section III, Division 1.
c. Special construction techniques, if any, are reviewed on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

a. Procedures for the preoperational structural proof test are acceptable if found in accordance with Article NE-6000 of Subsection NE of the Code, Section III, Division 1.
b. In-service surveillance requirements for steel containments have not yet been established by the Code and they are currently under development. Acceptance criteria for in-service surveillance programs in areas subject to corrosion are established by the Materials Engineering Branch (MTEB), as required.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Description of the Containment

After the type of containment and its functional characteristics are identified, information on similar and previously-licensed applications is obtained for reference. Such information, which is available in safety analysis reports and

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TABLE 3.8.2-1

STRESS LIMITS FOR STEEL CONTAINMENTS

Section II.3.b Combination No.		Primary Stresses: Gen. Memb. P_m	Primary Stresses: Local Memb. P_L	Primary Stresses: Bend + Local Memb. $P_B + P_L$	Primary & Secondary Stresses	Peak Stresses	Buckling Note (3)
(1)		$.9S_y$	$1.25S_y$	$1.25S_y$	$3S_m$	Consider for Fatigue Analysis	125% of Allow. Given by NE-3133
(2) & (3)		S_m	$1.5S_m$	$1.5S_m$	$3S_m$	Consider for Fatigue Analysis	Allow. Given by NE-3133
(4) & (5)		S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. Given by NE-3133
(6) & (7)	Not integral and continuous	S_m	$1.5S_m$	$1.5S_m$	N/A	N/A	Allow. Given by NE-3133
	Integral and continuous	The Greater of $1.2S_m$ or S_y	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133
(8)	Not integral and continuous	The Greater of $1.2S_m$ or S_y	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133
	Integral and continuous	85% of Stress Intensity Limits of Appendix F			N/A	N/A	85% of Allow. Given by F-1325 of App. F
(9)		$1.5S_m$	The Greater of $1.8S_m$ or $1.5S_y$	The Greater of $1.8S_m$ or $1.5S_y$	N/A	N/A	120% of Allow. Given by NE-3133

NOTES:
(1) Thermal stresses need not be considered in computing P_m, P_L and P_B.
(2) Thermal effects are considered in:
(a) Specifying stress intensity limits as a function of temperature.
(b) Analyzing effects of cyclic operation (NB-3222.4).
(3) If a detailed analysis considering inelastic behavior is performed for checking instability (buckling), such an analysis should demonstrate that the applied stress is less than 50% of the critical buckling stress. Designs utilizing vertical stiffeners are permitted. The allowable axial compressive stress may be determined by considering the effects of circumferential stiffener spacing and the effects of water, if present.

amendments of previous license applications, enables identification of differences for the case under review which require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications
The list of codes, standards, guides, and specifications is checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and effective addenda are utilized.

3. Loads and Loading Combinations
The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Loading conditions that are unique and that are not specifically covered in Section II.3, are treated on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant for further consideration.

4. Design and Analysis Procedures
The reviewer assures himself that the applicant is committed to the design and analysis procedures delineated in Article NE-3000 of Subsection NE of the Code, Section III, Division 1. Any exceptions to these procedures are reviewed and evaluated on a case-by-case basis. In particular, the areas of review contained in Section I.4 of this plan are evaluated for conformance with the acceptance criteria.

5. Structural Acceptance Criteria
The limits on allowable stresses in the steel shell and its components are reviewed and compared with the acceptable limits specified in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points of the structure, the justification provided to show that the structural integrity of the containment will not be affected is reviewed and evaluated. If such justification is unacceptable, the applicant is required to comply with the acceptance criteria delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques
The information provided on materials, quality control programs, and special construction techniques, if any, is compared with that referenced in Section II.6 of this plan. If a material not covered by the Code is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of the material. Similarly, any new quality control programs or construction techniques

are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the containment and its various components.

7. Testing and In-service Surveillance Requirements
   The initial structural overpressure test program is reviewed and compared with that indicated as acceptable in Section II.7 of this plan. Any proposed deviations are considered on a case-by-case basis. In-service surveillance programs, if any, as presented in the Technical Specifications of the Operating License, are similarly reviewed.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and concludes that his evaluation is sufficiently complete to support the following type of conclusive statement to be included in the staff's safety evaluation report:

> "The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and guides acceptable to the Regulatory staff.
>
> "The use of these criteria as defined by applicable codes, standards, and guides; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occuring within and outside the containment, the structure will withstand the specified conditions without impairment of structural integrity or safety function. A Category I concrete shield building protects the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2, 4, 16, and 50."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.
2. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."
3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).
4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."
5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

6. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."
7. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - Mechanical Engineering Branch (MEB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The following areas relating to the containment internal structures are reviewed:

1. Description of the Internal Structures

The descriptive information including plans and sections of the various internal structures is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures. The internal structures have several safety-related functions for which their structural integrity is important. By providing support during normal operation and seismic disturbances, they should prevent the occurrence of a loss of coolant accident (LOCA). If such an accident does occur, however, they should act to mitigate its consequences by protecting the containment and other engineered safety features from the effects induced by the accident such as jet forces and whipping pipes.

The major containment internal structures that are reviewed, together with the primary structural function of each structure, and the extent of descriptive information required for each structure, are indicated below. For equipment supports that are not covered by this plan, reference is made to Standard Review Plan 3.9.3.

For PWR Dry Containment Internal Structures

a. Reactor Supports

The PWR vessel should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture including the loss of coolant accident. The support and restraint system should limit the movement of the vessel to within allowable limits under the applicable combinations of loadings.

The support system should nevertheless minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the reactor vessel linear supports are reviewed with emphasis on methods of transferring loads from the vessel to the support and eventually to the structure and its foundations. Shell-type supports and component standard supports are reviewed by the Mechanical Engineering Branch (MEB). The definition of linear, shell type, and standard supports is in accordance with Subsection NF of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (Ref. 1). Where uplift supports are utilized, the method of anchoring such supports in the concrete is also reviewed.

b. <u>Steam Generator Supports</u>

Steam generators should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by pipe rupture. The support system should prevent the rupture of the primary coolant pipes due to a postulated rupture in steam or feedwater pipes and vice versa. The system should nevertheless minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the steam generator linear supports are reviewed with emphasis on methods of transferring loads from the vessel to the support and eventually to the structure and its foundations. Shell-type supports, standard supports, and mechanical restraints such as hydraulic snubbers are reviewed by the Mechanical Engineering Branch (MEB).

c. <u>Reactor Coolant Pump Supports</u>

Reactor coolant pumps should be supported and restrained to prevent excessive deflections during normal operating, seismic, and pipe rupture conditions. Under LOCA loads, the pump should not become a missile and should not generate missiles that might damage other safety-related components. The pump support system should also minimize resistance to thermal movements expected during operation.

With these functional requirements in mind, the general arrangement and principal features of the pump linear supports are reviewed with emphasis on methods of transferring loads from the pump to the support and eventually to the structure and its foundations. Shell-type supports, standard supports, and mechanical restraints such as hydraulic snubbers are reviewed by the Mechanical Engineering Branch (MEB).

d. <u>Primary Shield Wall and Reactor Cavity</u>

The primary shield wall forms the reactor cavity and usually supports and restrains the reactor vessel. It is usually a thick wall that surrounds the reactor vessel and may be anchored through the liner plate to the containment base slab.

The general arrangement and principal features of the wall and cavity are reviewed including the main reinforcement and anchorage system.

e. Secondary Shield Walls

The secondary shield walls surround the primary loops, forming the steam generator compartments, and protecting the containment from the effects of pipe rupture accidents inside the compartment. They may also support intermediate floors and the operating floor. The general arrangement and principal features of these walls are reviewed with emphasis on the method of structural framing and expected behavior under compartment pressure loads and jet forces, particularly those associated with the LOCA.

f. Other Interior Structures

The other major interior structures of PWR dry containments that are reviewed in a similar manner are the pressurizer linear supports, refueling pool walls, the operating floor, other intermediate floors, and the polar crane supporting elements.

For PWR Ice-condenser Containment Internal Structures

In PWR plants where the ice-condenser containment system is utilized, in addition to the applicable structures reviewed in dry PWR containments, the following elements are also reviewed:

a. The Divider Barrier

In the PWR ice-condenser containment system, which utilizes the pressure-suppression concept, the divider barrier surrounds the reactor coolant system. The upper portion of the divider barrier is nearly surrounded by the ice-condenser which is bounded by the containment shell on the outside and by the divider barrier wall on the inside. Several venting doors connect the space inside the divider barrier to the ice-condenser.

In the event of a LOCA, the divider barrier will contain the steam released from the reactor coolant system and, temporarily acting as a pressure-retaining envelope, will channel the steam through the venting doors and into the ice-condenser. The ice will condense the steam and the energy released to the containment will thus be minimized.

Following such a LOCA and before blowdown is completed, the divider barrier will be subjected to a differential pressure and possibly jet forces, and any structural failure in its boundary may result in steam bypassing the ice-condenser and flowing directly into the containment, possibly generating a containment pressure higher than that for which it has been designed.

With this functional requirement in mind, the general arrangement and principal features of the divider barrier are reviewed with emphasis on structural framing and expected behavior when subjected to the design loads.

b. Ice-Condenser

A major feature of the ice-condenser containment is the ice-condenser which contains the baskets of ice forming the heat sink essential for pressure suppression. The structurally significant components of the ice-condenser that are reviewed are the vent doors, ice baskets, brackets, couplings and lattice framings, lower and upper supports, and insulating and cooling panels.

The general arrangment and principal features of these major components are reviewed with emphasis on the structural framing, supports, and expected behavior when subjected to design loads.

For BWR Containment Internal Structures

Since it is expected that future BWR applications will utilize the Mark III containment concept, this Standard Review Plan is oriented towards and based on this type of containment. For other types of BWR containments, modifications to this plan are made on a case-by-case basis.

Among the major Mark III containment internal structures that are reviewed, together with the primary structural function of each structure, and the extent of descriptive information required for each structure, are the following:

a. Drywell

In the BWR Mark III containment system, which utilizes the pressure-suppression concept, the drywell surrounds the reactor coolant system. The lower portion of the drywell is surrounded by the suppression pool which is bounded by the containment shell on the outside and by a weir wall located just inside the drywell wall. Several vent holes connect the drywell to the suppression pool. In the event of a loss-of-coolant accident, the drywell will contain the steam released from the reactor coolant system and, temporarily acting as a pressure-retaining envelope, will channel the steam through the vent holes and into the suppression pool. The pool water will condense the steam and the energy released to the containment will thus be minimized.

Following such a LOCA and before blowdown is completed, the drywell will be subjected to a differential pressure and possibly jet forces, and any structural failure in its boundary would result in steam bypassing the suppression pool and flowing directly into the containment, possibly generating a containment pressure higher than that for which it has been designed.

With this functional requirement in mind, the general arrangement and principal features of the drywell are reviewed with emphasis on structural framing and expected behavior under loads. Since the drywell geometrically resembles, to a certain degree, a containment, the descriptive information reviewed is similar to that reviewed for containments as delineated in Section I.1 of Standard Review Plan 3.8.1. The major components of the drywell that are so reviewed, other than the main body of the drywell, include the bottom vent region, the roof and drywell head, and major penetrations.

b. Weir Wall

The weir wall forms the inner boundary of the suppression pool and is located inside the drywell. It completely surrounds the lower portion of the reactor coolant system. The general arrangement and principal features of the weir wall are reviewed with emphasis on structural framing and behavior under loads.

c. Refueling Pool and Operating Floor

The refueling pool walls are located on top of the drywell. The outer walls form a rectangular pool that is usually subdivided by two interior crosswalls. The base slab of the pool is common to the drywell roof slab. The pool may be filled continuously with water for shielding purposes during operation.

The general arrangement and principal features of the refueling pool are reviewed with emphasis on structural framing and behavior under loads.

The operating floor is intended to provide laydown space for refueling operations and is usually a combination of reinforced concrete and structural steel framing. The containment walls and the refueling pool walls may support the floor.

The general arrangement and principal features of the operating floor are reviewed.

d. Reactor and Recirculation Pump Supports

The support systems of the BWR vessel and recirculation pumps have the same functions as the support systems for PWR vessels and pumps are similarly reviewed.

e. Reactor Pedestal

The reactor pedestal is usually a cylindrical structure located below and supporting the reactor vessel, which is anchored to the top of the pedestal.

The general arrangement and principal features of the reactor pedestal are reviewed with emphasis on structural framing, main reinforcement and the manner in which the pedestal is anchored to the containment base slab.

f. Reactor Shield Wall

This is usually a cylindrical wall surrounding the reactor vessel for radiation shielding purposes. It is supported on the reactor pedestal. The wall may be lined on both surfaces with steel plates which also may act as the main structural components of the wall. The wall may also be utilized as an anchor for pipe restraints.

The general arrangement and principal features of the wall are reviewed with particular emphasis on structure framing and behavior under loads.

g. Other Interior Structures

The other major interior structures constructed of reinforced concrete or structural steel or combinations thereof that are also reviewed in a similar manner are the floors located inside the drywell and in the annulus between the drywell and the containment, and the polar crane supporting elements. The general arrangement and principal features of these structures are reviewed.

2. Applicable Codes, Standards, and Specifications

The information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of the containment internal structures, is reviewed. The specific edition, date, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various load combinations thereof is reviewed. The loads normally applicable to containment internal structures include the following:

a. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as in refueling and pressure suppression pools.
b. Those loads to be sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) specified for the plant site.
c. Those loads to be sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) specified for the plant site.
d. Those loads to be sustained during abnormal plant conditions. The most critical abnormal plant condition during which most of the containment internal structures have to perform their primary function is the design basis LOCA. Ruptures of other high-energy pipes should also be considered. Time-dependent and dynamic loads induced by such accidents include elevated temperatures and differential pressures across compartments, jet impingement, impact forces associated with the postulated ruptures of piping, and loads applicable to some structures such as pool swell loads in the BWR Mark III containment and drag forces in the PWR ice-condenser containment.

The various combinations of the above loads that are normally postulated and reviewed include the following: normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; and normal operating with extreme environmental and abnormal loads.

In addition, the following information is reviewed:

a. The extent to which the applicant's criteria comply with the "Building Code Requirements for Reinforced Concrete," ACI 318-71 (Ref. 2) for concrete, and with the AISC "Specification for Design," Fabrication and Erection of Structural Steel for Buildings" (Ref. 3) for steel, as applicable.
b. For concrete portions of the divider barrier of the PWR ice-condenser containment and for concrete portions of the drywell of the Mark III BWR containment, the extent to which the applicant's loading criteria comply with Article CC-3000 of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359) (Ref. 4). For steel pressure-resisting portions of these two structures, the extent to which the applicant's loading criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Div. 1, (Ref. 5) as augmented by Regulatory Guide 1.57 (Ref. 9).
c. For steel linear supports of the reactor coolant system, the extent to which the applicant's criteria comply with Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).

4. Design and Analysis Procedures

The design and analysis procedures utilized for the containment internal structures are reviewed with emphasis on the extent of compliance with the applicable codes as indicated in Section I.3, including those applicable to the following areas:

For PWR Dry Containment Internal Structures

a. Reactor Coolant System Supports

The support system for the reactor vessel, steam generators, and reactor coolant pumps, as described in Section I of this plan, should be designed to resist various combinations of loadings, including normal operating loads, seismic loads, and loss of coolant and other pipe rupture accident loads.

Analytical procedures for determining normal operating loads and accident loads are reviewed by the Mechanical Engineering Branch (MEB).

Analytical procedures for determining seismic loads are as described in Standard Review Plan 3.7.3.

After the procedures for determining individual loads and combinations thereof are so reviewed, the design and analysis methods utilized for the linear supports are reviewed including the type of analysis (elastic or plastic), the methods of load transfer, and the assumptions on boundary conditions. Specifically, the extent of compliance with design and analysis procedures delineated in Subsection NF of the ASME Section III Code, Division 1 (Ref. 1), is reviewed.

b. Primary Shield Wall and Reactor Cavity

The primary shield wall should withstand all the applicable loads including those transmitted through the reactor supports. It is subjected to most of the loads described in Section I.3 of this plan and should be designed and analyzed for all the applicable load combinations. During normal plant operation, a thermal gradient across the wall is generated by the attenuation heat of gamma and neutron radiation originating from the reactor core. Insulation and cooling systems may be provided to reduce the severity of this gradient by limiting the rise in temperature to an acceptable level.

Procedures for determining seismic loads on the primary shield wall are reviewed in accordance with Standard Review Plan 3.7.2.

Loss of coolant accident loads that are applicable to the primary shield wall include a differential pressure created across the reactor cavity by a pipe break in the vicinity of the reactor nozzles. Such a transient pressure may act on the entire cavity or on portions thereof. Procedures for determining such pressures are reviewed by the Containment Systems Branch (CSB).

Other loss of coolant accident loads that apply are those transmitted to the wall through the reactor supports including pipe rupture reaction forces which may induce simultaneous shear forces, torsional moments, and bending moments at the base of the wall. Further, the elevated temperature within and around the primary shield created by the accident may produce transient thermal gradients across the thick wall. Design and analysis procedures for such accident effects are accordingly reviewed.

c. Secondary Shield Walls

The secondary shield walls surrounding the primary loops and supporting the operating floor should be designed for loads similar to those applicable to the primary shield wall including loads of fluid jets from a postulated break of a primary pipe which can impinge on these walls. The analytical techniques utilized for these walls are reviewed including their structural framing and behavior under loads. Where elasto-plastic behavior is assumed and the ductility of the walls is relied upon to absorb the energy associated with jet loads, the procedures and assumptions are reviewed with particular emphasis on such areas as modeling techniques, boundary conditions, force-time functions, and assumed ductility. For the time-dependent differential pressure, however, elastic behavior is required and the methods of determining an equivalent static load are accordingly reviewed.

d. Other Interior Structures

Most of the other interior structures that are also reviewed are combinations of slabs, walls, beams and columns, classified as Category I structures and subject to most of the loads and combinations described in Section I.3 of this plan. Analytical techniques for these structures are reviewed on the same basis as for the structures described above.

For PWR Ice-Condenser Containment Internal Structures

a. Divider Barrier

Since the divider barrier has to maintain a certain degree of leak-tightness during a LOCA and is thus a critical structure with respect to the proper functioning of the containment, it is treated on the same basis as the containment.

The loads that usually govern the design of the divider barrier are those induced by the LOCA, including the time-dependent differential pressure across the barrier and any concurrent concentrated jet impingement loads. As the divider barrier is typically a combination of walls and slabs framed together, the design and analysis procedures are of the conventional type. They are accordingly reviewed with emphasis on the assumed boundary conditions and behavior under loads. Since the differential pressure and jet impingement loadings are dynamic impulsive loads that vary with time, the techniques utilized to determine their equivalent static loads are reviewed.

b. Ice-Condenser

The design of the ice-condenser and its various components may be based on a combination of analysis and testing. The analytical and testing procedures that are reviewed include those for the ice baskets and brackets (couplings); the lattice frames and columns including attachments; the supporting structures comprising the lower supports; the wall panels and cooling duct and supports of various auxiliary components.

The ice-condenser and its components should be analyzed or tested for various loads and combinations thereof including dead and live loads, thermal loads induced by differential thermal expansion within the various elements, seismic loads and loads induced by the loss-of-coolant accident. Accident loads include pressure differential drag loads and loads induced by the change of momentum of the flowing steam.

Elastic analysis is usually utilized for the ice-condenser and its components. However, plastic analysis may also be used as an alternate. Accordingly, the load factors that are applied to each of the applicable loads and the basis and justification of these load factors are reviewed.

Where experimental verification of the design using simulated load conditions is used, the procedures used to account for similitude relationships which exist between the actual component and the test model are reviewed to assure that the results obtained from the test are a conservative representation of the load carrying capability of the actual component under the postulated loading.

For BWR Containment Internal Structures

a. Drywell

The drywell, which has to maintain a certain degree of leak-tightness during a LOCA, is critical with respect to the proper functioning of the containment. Accordingly, and since it geometrically resembles a containment, the design and analysis procedures utilized for the drywell are reviewed on a basis similar to those of containments as described in Section I.4 of Standard Review Plans 3.8.1 and 3.8.2 for concrete and steel portions, respectively.

b. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. Under such a concentrated load, the weir wall should not deform to an extent that might impair or degrade the pressure-suppression performance. Accordingly, the procedures utilized to analyze the wall for such dynamic time-dependent loads are reviewed with particular emphasis on modeling techniques, assumptions on boundary conditions, and behavior under loads.

c. Refueling Pool and Operating Floor

In the BWR Mark III containments reviewed recently, the refueling pool is continuously filled with water to provide biological shielding above the reactor. The operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, is a combination of reinforced concrete and structural steel. The design and analysis procedures for the refueling pool and the operating floor are of the conventional type and are accordingly reviewed, with particular emphasis on the structural framing and behavior under loads. In cases where the floor beams are supported vertically on the containment shell, they should be laterally isolated to minimize interaction between the containment and its interior.

d. Reactor and Recirculation Pump Supports

The design and analysis procedures utilized for the reactor and recirculation pump supports are reviewed in a similar manner to that for PWR reactor and pump supports, as already described in this plan.

e. Reactor Pedestal

The reactor pedestal supports the reactor and has to withstand the loads transmitted through the reactor supports. It is thus subjected to most of the loads described in Section I.3 of this plan and is designed and analyzed for all the applicable load combinations.

Because of the similarity in geometry and function of the BWR reactor pedestal to the PWR primary shield wall, the design and analysis procedures are similar and are reviewed accordingly as has already been discussed in this plan.

f. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, is also subjected to most of the loads described in Section I.3 of this plan. In most cases, the wall is utilized to anchor pipe restraints placed around the reactor coolant system piping. Moreover, a pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall is usually lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads.

The analytical and design techniques utilized to determine the effect of the design loads on the wall are reviewed with particular emphasis on the assumed boundary conditions and the behavior of the wall under loads.

g. Other Interior Structures

There are several platforms within the BWR Mark III containment some of which are inside the drywell and the others outside in the annulus between the drywell and the containment. Platforms inside the drywell are usually of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete and have to be designed to resist the various applicable loads particularly the effects of pool swell during a loss-of-coolant accident. The analytical procedures for determining pool swell loads are reviewed by the Containment Systems Branch (CSB). Design and analysis procedures for these platforms are reviewed with particular emphasis on the framing and structural behavior under loads.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the various interior structures of the containment are reviewed, specifically with respect to stresses, strains, deformations, and factors of safety against structural failure, with emphasis on the extent of compliance with the applicable codes as indicated in Section I.3 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

Information provided on the materials that are used in the construction of the containment internal structures is reviewed. Among the major materials of construction that are reviewed are the concrete ingredients, reinforcing bars and splices, and structural steel and various supports and anchors.

The quality control program that is proposed for the fabrication and construction of the containment interior structures is reviewed including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special, new, or unique construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed interior structure.

In addition, the following information should be provided:

a. The extent to which the materials and quality control programs comply with the "Building Code Requirements for Reinforced Concrete," ACI 318-71 (Ref. 2), for concrete, and with the AISC "Specifications for Design, Fabrication and Erection of Structural Steel for Buildings,"(Ref. 3), for steel, as applicable.
b. For steel linear supports of the reactor coolant system, the extent to which the material and quality control programs comply with Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).
c. For quality control in general, the extent to which the applicant complies with ANSI N45.2.5 (Ref. 7).
d. If welding of reinforcing bars is proposed, the extent to which the applicant complies with the applicable sections of the proposed "Standard Code for Concrete Reactor Vessels and Containments," ACI-ASME (ACI-359) (Ref. 4), should be described and any exceptions taken should be justified.

7. Testing and Inservice Surveillance Programs

If applicable, any post-construction testing and in-service surveillance programs are reviewed on a case-by-case basis.

The structural test for the drywell of the BWR Mark III containment is reviewed in a similar manner to that of the containment.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. Description of the Internal Structures

The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.3.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 8).

Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated at the application acceptance review. New or unique design features that are not specifically covered in the "Standard Format" may require a more detailed review. The reviewer determines if additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and in-service surveillance, if any, of interior structures of containments are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. The following codes, standards, specifications, and guides are acceptable.

Code, Standard, or Specification	Title
ACI 318-71	Building Code Requirements for Reinforced Concrete
ACI/ASME (ACI-359)	Proposed Standard Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2
ASME	Boiler and Pressure Vessel Code, Section III, Subsections NE and NF
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings
ANSI N45.2.5	Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
Regulatory Guides	
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures
1.15	Testing of Reinforcing Bars for Category I Concrete Structures
1.55	Concrete Placement in Category I Structures

3. Loads and Load Combinations

With the exception of the divider-barrier and ice-condenser elements of the ice-condenser PWR containment, the drywell of the BWR Mark III containment, and the steel linear supports of the reactor coolant system, the loads and load combinations for all other containment interior structures described in Section I.1 of this plan, are acceptable if found in accordance with the following:

Loads, Definitions, and Nomenclature

All the major loads to be encountered or to be postulated are listed below. All the loads listed, however, are not necessarily applicable to all the interior structures. Loads and the applicable load combinations for which each structure has to be designed will depend on the conditions to which that particular structure could be subjected.

Normal loads, which are those loads to be encountered during normal plant operation and shutdown, include:

- D --- Dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads. For equipment supports, it also includes static and dynamic head and fluid flow effects.
- L --- Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence. For equipment supports, it also includes loads due to vibration and any support movement effects.
- T_o --- Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.
- R_o --- Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

Severe environmental loads include:

- E --- Loads generated by the operating basis earthquake.

Extreme environmental loads include:

- E' --- Loads generated by the safe shutdown earthquake.

Abnormal loads, which are those loads generated by a postulated high-energy pipe break accident, include:

- P_a --- Pressure equivalent static load within or across a compartment generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.
- T_a --- Thermal loads under thermal conditions generated by the postulated break and including T_o.
- R_a --- Pipe reactions under thermal conditions generated by the postulated break and including R_o.
- Y_r --- Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.
- Y_j --- Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.
- Y_m --- Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Load Combinations for Concrete Structures

For concrete interior structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the working stress design (WSD) method or the strenght design method may be used.

(i) If the WSD method is used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

If thermal stresses due to T_o and R_o are present, the following combinations should be also considered:

(1a) $D + L + T + R_o$

(2a) $D + L + T_o + R_o + E$

Both cases of L having its full value or being completely absent should be checked.

(ii) If the strength design method is used, the following load combinations should be considered:

(1) $1.4D + 1.7L$

(2) $1.4D + 1.7L + 1.9E$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $(0.75)\ (1.4D + 1.7L + 1.7T_o + 1.7R_o)$

(2b) $(0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o + 1.7R_o)$

b. For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the strength design method should be used and the following load combinations should be considered:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + 1.5\ P_a$

(5) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25E$

(6) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (4), (5), and (6), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5) and (6) and the corresponding structural acceptance criteria of Section II.5 of this plan should first be satisfied without Y_r, Y_j, and Y_m. When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

<u>Load Combinations for Steel Structures</u>

For steel interior structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the elastic working stress design methods for Part 1 of AISC, or the plastic design methods of Part 2 of AISC, may be used.

(i) If the elastic working stress design methods are used:

(1) D + L

(2) D + L + E

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(ii) If the plastic design methods are used:

(1) 1.7D + 1.7L

(2) 1.7D + 1.7L + 1.7E

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $1.3 (D + L + T_o + R_o)$

(2b) $1.3 (D + L + E + T_o + R_o)$

b. For factored load conditions, the following load combinations should be considered:

(i) If the elastic working stress design methods are used:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + P_a$

(5) $D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + E$

(6) $D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + E'$

(ii) If the plastic design methods are used:

(3) $D + L + T_o + R_o + E'$

(4) $D + L + T_a + R_a + 1.5 P_a$

(5) $D + L + T_a + R_a + 1.25 P_a + 1.0 (Y_j + Y_r + Y_m) + 1.25 E$

(6) $D + L + T_a + R_a + 1.0 P_a + 1.0 (Y_j + Y_r + Y_m) + 1.0 E'$

In the above combinations, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature.

In combinations (4), (5), and (6), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5) and (6) and the corresponding structural acceptance criteria of Section II.5 of this plan should first be satisfied without Y_r, Y_j, and Y_m. When considering these loads, however, local section strengths or stresses may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

For the divider barrier, ice-condenser elements, the Mark III containment drywell, and for the steel linear supports of the reactor coolant system, the loading criteria are acceptable if found in accordance with the following:

a. Divider barrier

As the structural integrity of the divider barrier and, to a certain extent, its leak-tight integrity as well, are important to the proper functioning of the ice-condenser containment system, it is treated for design purposes similar to the containment itself.

Accordingly, for concrete pressure-resisting portions of the divider barrier, the loads and load combinations of Article CC-3000 of ACI-359 (Ref. 4) will apply, with the following exceptions.

For Table CC-3200-1

(i) Jet impingement loads, Y_j, and impact loads of missiles associated with the loss-of-coolant accident, Y_m, should be included.

(ii) The 6th combination, representing abnormal conditions, need not include Y_r in combination with 1.5 P_a,

(iii) In the 7th, 8th, and 9th combinations, representing abnormal/severe environmental and abnormal/extreme environmental load conditions, the "and/or" between R_a and Y_r should be deleted and, in addition to R_a and Y_r, the combinations should include Y_j and Y_m.

(iv) It should be indicated that the maximum values of P_a, T_a, R_a, Y_r, Y_j, and Y_m, including an appropriate dynamic load factor, should be applied simultaneously, unless a time-history analysis is performed to justify otherwise.

Steel portions of the divider barrier which resist the design differential pressure and are not backed by concrete, such as penetrations, hatches, locks and guard pipes, should be designed in accordance with the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, (Ref. 5) together with the applicable loads, load combinations, and acceptance criteria of Regulatory Guide 1.57, (Ref. 9). Specifically, the load combinations of Section II.3 of Standard Review Plan 3.8.2 apply.

b. Ice-condenser Elements

In the ice-condenser containment system the structural integrity of the ice baskets, ice bed framing, and their supports, is important to the functional integrity of the containment system. The major loads that are applicable to the ice-condenser elements are: D, L, E, E', and P_a. For this structure, P_a is the LOCA pressure load induced by drag and change in momentum of the flowing air and steam. Load combinations for the ice-condenser elements are acceptable if found in accordance with the following:

(i) For service load conditions, if elastic working stress design methods are used:

(1) D + L

(2) D + L + E

(ii) For service load conditions, if plastic design methods are used:

(1) 1.7 D + 1.7 L

(2) 1.7 D + 1.7 L + 1.7 E

(iii) For service load conditions, if an experimental test verification of the design is used:

(1) 1.9 D + 1.9 L

(2) 1.9 D + 1.9 L + 1.9 E

If thermal stresses are significant and have to be considered, an acceptable procedure for accounting for such thermal loads is contained in item (a) of Subarticle NF-3231.1 of Subsection NF of the ASME Code, Section III, Division 1 (Ref. 1).

(iv) For factored load conditions, if elastic working stress design methods are used:

(3) $D + L + E'$

(4) $D + L + P_a$

(5) $D + L + P_a + E'$

(v) For factored load conditions, if plastic design methods are used:

(3) $1.3\ D + 1.3\ L + 1.3\ E'$

(4) $1.3\ D + 1.3\ L + 1.3\ P_a$

(5) $1.2\ D + 1.2\ L + 1.2\ P_a + 1.2\ E'$

(vi) For factored load conditions, if an experimental test verification of the design is used:

(3) $1.4\ D + 1.4\ L + 1.4\ E'$

(4) $1.4\ D + 1.4\ L + 1.4\ P_a$

(5) $1.3\ D + 1.3\ L + 1.3\ P_a + 1.3\ E'$

c. <u>BWR Mark III Containment Drywell</u>

As the structural integrity of the drywell and, to a certain extent, its leak-tight integrity as well, are critically important to the proper functioning of the Mark III pressure-suppression system, the drywell is treated, for design and testing purposes only, similar to the containment itself.

Accordingly, for concrete pressure-resisting portions of the drywell, the loads and loading combinations of Article CC-3000 of ACI-359 (Ref. 4) will apply, with the exceptions listed for concrete portions of the PWR ice-condenser divider barrier.

For steel components of the drywell that resist pressure and are not backed by concrete, such as the drywell head, the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, (Ref. 5) should be used together with the applicable loads, load combinations, and acceptance criteria of Regulatory Guide 1.57 (Ref. 9). Specifically, the load combinations of Section II.3 of Standard Review Plan 3.8.2 apply.

For the lower vent portion of the drywell:

(i) If the main reinforcement of the drywell is carried down between the vent holes and the reinforced concrete section is relied upon for structural purposes, the criteria that apply to concrete portions of the drywell as described above will apply.

(ii) If the main reinforcement of the drywell is terminated above the vent holes and two steel plates lining both faces of the drywell are alone utilized for structural purposes, the criteria that apply to steel portions of the drywell as described above will apply.

(iii) If other structural systems are used in the vent region, the loads and load combinations are reviewed and judged on a case-by-case basis.

d. <u>Reactor Coolant System Supports</u>

Steel linear supports for the reactor vessel, steam generators, reactor coolant pumps, and recirculation pumps, as described in Section I of this plan, are governed by Subsection NF of the ASME Code, Section III, Division 1. This Code does not explicitly delineate load combinations for the design of these supports. Accordingly, the following combinations should be satisfied as a minimum:

Load Combinations

(1) If the elastic method of analysis of paragraph NF-3231.1 of Subsection NF of the ASME Code, Section III, Division 1, is used, the following combinations should be satisfied as a minimum:

(i) $D + L + E$

(ii) $D + L + E' + P_a + Y_r + Y_j + Y_m$

In addition, the conditions of item (a) of paragraph NF-3231.1 shall be satisfied.

(2) If the limit method of analysis of paragraph NF-3231.2 of Subsection NF is used, the following combinations should be satisfied as a minimum:

(i) $1.7\ (D + L + E)$

(ii) $1.0\ (D + L + E' + P_a + Y_r + Y_j + Y_m)$

4. Design and Analysis Procedures

The design and analysis procedures utilized for the interior structures of the containment are acceptable if found in accordance with the following:

For PWR Dry Containment Internal Structures

a. Reactor Coolant System Supports

The linear support systems for the reactor vessel, steam generators, and reactor coolant pumps, as described in Section I of this plan, should be analyzed for and designed to resist various combinations of loadings as indicated in Section II.3 of this plan. Design and analysis procedures for such supports are acceptable if in accordance with Subsection NF of the ASME Section III Code, Division 1, (Ref. 1), particularly with Appendix XIII.

b. Primary Shield Wall and Reactor Cavity

The design and analysis procedures utilized for the shield wall are acceptable if in accordance with the ACI 318-71 Code (Ref. 2). This code is mostly based on the strength design method. However, the use of Section 8.10 of the Code, which is based on the working stress design method where actual elastic/linear stresses in the concrete and reinforcement are determined and compared with their corresponding allowables, is considered acceptable.

Analyses for loss-of-coolant accident loads applicable to the primary shield wall, such as for the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads whereby either a detailed time-history analysis is performed, or a static analysis utilizing the peak of the forcing function amplified by an appropriately chosen dynamic load factor is utilized. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads such as Y_r and Y_j, elasto-plastic behavior may be assumed as long as the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if in accordance with recognized dynamic analysis methods.

c. Secondary Shield Walls

Design and analysis procedures utilized for the secondary shield walls are acceptable if in accordance with conventional beam/slab design and analysis procedures described in the ACI 318-71 Code.

Similar to the primary shield wall, the secondary shield walls are also subject to dynamic loss-of-coolant accident loads and the same methods described in paragraph b. above are, therefore, applicable and acceptable.

d. Other Interior Structures

Most of the other interior structures that are reviewed are combinations of reinforced concrete and steel slabs, walls, beams, and columns, which are classified as Category I structures subject to the loads and load combinations described in Section II.3 of this plan. Analytical techniques for these structures are acceptable if found in accordance with those described in the ACI 318-71 Code for concrete and with those in the AISC specifications for steel.

For PWR Ice-condenser Containment Internal Structures

a. Divider Barrier

The most important loads that usually govern the design of the divider barrier are those induced by the loss-of-coolant accident, including the differential pressure across the barrier and any concentrated jet impingement loads. As the divider barrier is a combination of walls and slabs framed together, the design and analysis procedures are acceptable if in accordance with those contained in Section 8.10 of the ACI 318-71 Code for the concrete portions of the divider barrier. These methods are based on the elastic/linear working stress design method where actual stresses are determined.

For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1.

b. Ice-condenser Elements

The design and analysis procedures for the ice-condenser and its various components are acceptable if in accordance with either the elastic/linear design method of Part 1 of the AISC Specifications or with the plastic design method of Part 2 of the same Specifications. For components where experimental testing is utilized to verify the design, the testing procedures are acceptable if in accordance with recognized prototype or model testing procedures where the effect of scaling and similitude are taken into consideration.

For BWR Containment Internal Structures

a. Drywell

The design and analysis procedures utilized for concrete portions of the drywell are acceptable if in accordance with Section II.4 of Standard Review Plan 3.8.1. For steel portions of the drywell that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1.

b. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. The deflection of the wall under such a load must be limited so as not to impair the pressure-suppression performance. The procedures utilized to analyze the wall for such a dynamic time-dependent load are acceptable if a detailed time-history dynamic analysis is performed or if an equivalent static analysis is performed utilizing the peak of the jet load amplified by an appropriately chosen dynamic load factor.

c. Refueling Pool and Operating Floor

The refueling pool and the operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, are a combination of reinforced concrete and structural steel. The design and analysis procedures are acceptable if found in accordance with conventional methods described in the ACI 318-71 Code for concrete and in the AISC Specifications for structural steel.

d. Reactor Supports

The linear support system for the reactor vessel, described in Section I of this plan, should be designed to resist various combinations of loadings as indicated in Section II.3 of this plan. Among the major loads that should be considered are normal operating loads, seismic loads, and loss-of-coolant accident loads.

Design and analysis procedures are acceptable if in accordance with those delineated in Subsection NF of the ASME Section III Code, Division 1, particularly with Appendix XVII.

e. Reactor Pedestal

The reactor pedestal, which supports the reactor and has to withstand the loads transmitted through the reactor supports, should be subjected to most of the loads described in Section II.3 and should be designed for all the applicable load combinations.

The design and analysis procedures are acceptable if found to be similar to those referenced for the primary shield wall of PWR containments in paragraph (b) under PWR dry containments.

f. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, should be subjected to most of the loads described in Section II.3 of this plan. In most cases, the wall is utilized to anchor most of the pipe restraints placed around the reactor coolant system piping. A pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall may be lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads.

Similar to the reactor pedestal, the biological shield wall is also subjected to dynamic loss-of-coolant accident loads and the same methods are, therefore, applicable and acceptable.

g. Miscellaneous Platforms

Platforms inside the drywell are usually of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete. The analytical and design procedures for these platforms are acceptable if in accordance with the ACI 318-71 Code for reinforced concrete, and with the AISC Specifications for structural steel. Of particular interest are the dynamic loads induced on these floors by pool swell during a LOCA.

Computer programs used in the design and analysis of containment interior structure should be described and validated by any of the procedures described in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

With the exception of the divider barrier and ice-condenser elements of the ice-condenser PWR containment, the drywell of the BWR Mark III containment, and the steel linear supports of the reactor coolant system, the structural acceptance criteria for all other interior structures of the containment described in Section I.1 of this plan are acceptable if found in accordance with the following:

For each of the loading combinations delineated in the beginning of Section II.3 of this plan, the following defines the allowable limits which constitute the structural acceptance criteria:

In Combinations for Concrete Internal Structures	Limit
(a)(i) 1, 2	S(1)
(a)(i) 1a, 2a	1.3 S
(a)(ii) 1, 2	U(2)
(a)(ii) 1b, 2b	U
(b) 3, 4, 5, 6	U

In Combinations for Steel Internal Structures	Limit
(a)(i) 1, 2	S
(a)(i) 1a, 2a	1.5 S
(a)(ii) 1, 2	Y(3)
(a)(ii) 1b, 2b	Y
(b)(i) 3, 4, 5(4)	1.6 S
(b)(i) 6(4)	1.7 S
(b)(ii) 3, 4, 5, 6	.9 Y

Notes

(1) S --- For concrete structures, S is the required section strength based on the working stress design method and the allowable stresses defined in Section 8.10 of ACI 318-71.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

The 33% increase in allowable stresses for concrete and steel due to seismic loadings is not permitted.

(2) U --- For concrete structures, U is the section strength required to resist design loads based on the strength design methods described in ACI 318-71.

(3) Y --- For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

(4) --- For these two combinations, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

For the divider barrier, ice-condenser elements, the drywell, and the linear steel supports of the reactor coolant system, the structural acceptance criteria are acceptable if found in accordance with the following:

a. Divider barrier

(i) For concrete portions of the divider barrier, the specified limits for stresses and strains are acceptable if found in accordance with Subsection CC-3400 of the ACI-359 Code, but with the following exceptions:

CC-3421.1

- The footnote on page 196 should be revised to indicate that the 33-1/3% increase in allowable stresses is permitted only for temperature loads and not for seismic loads.

CC-3422.1

- Item (c) should be deleted.

CC-3422.2

- The footnote on page 197 should be deleted.

(ii) For steel portions of the divider barrier which resist the design differential pressure and are not backed by concrete, the design should be similar to that of steel containments. Accordingly, the load combinations and stress limits of Section II.3 of Standard Review Plan 3.8.2 apply.

b. Ice-condenser Elements

For load combinations delineated in Section II.3 of this plan for the ice-condenser elements, the stress limits are acceptable if found in accordance with the following:

For Combinations:		Limit
(i)	(1), (2)	S(1)
(ii)	(1), (2)	Y(2)
(iii)	(1)	C(3)
(iv)	(3), (4)	1.3S
(iv)	(5)	1.6S
(v)	(3), (4), (5)	Y
(vi)	(3), (4), (5)	C

Notes

(1) S --- As defined in "Notes" under first tables in II.5 above.

(2) Y --- As defined in "Notes" under first tables in II.5 above.

(3) C --- Where experimental testing is used for verification of the design, C shall be the ultimate load carrying capacity of the member. Size effects and any similitude relationship which may exist between the actual component and the test model shall be accounted for in the evaluation of C.

c. BWR Mark III Containment Drywell

(i) For concrete portions of the drywell, the acceptance criteria of paragraph (a)(i) as described for the divider barrier apply.

(ii) For steel portions of the drywell that resist pressure and are not backed by structural concrete, the acceptance criteria of paragraph (a)(ii) as described for the divider barrier apply.

(iii) For the lower vent portion of the drywell:

- If the main reinforcement of the drywell is carried down between the vent holes and the reinforced concrete section is relied upon for structural purposes, the structural acceptance criteria is the same as for (i) above.
- If the main reinforcement of the drywell is terminated above the vent holes and two steel plates lining both faces of the wall are utilized for structural purposes, the acceptance criteria for (ii) above will apply.
- If other structural systems are used in the vent region, the acceptance criteria are reviewed on a case-by-case basis.

d. Reactor Coolant System Supports

The structural acceptance criteria for the steel linear supports of the reactor coolant system are acceptable if found in accordance with the following:

For load combinations delineated in paragraph (d) of Section II.3 of this plan for the reactor coolant system linear supports, the following acceptance criteria will apply:

(1) If the elastic analysis method is used:

Combination	Allowable Limits
(i)	Limits of XVII-2000 of Appendix XVII of the ASME Code, Section III.
(ii)	Limits of F-1370 of Appendix F of the ASME Code, Section III.

(2) If the limit method of analysis is used:

Combination	Allowable Limits
(i)	Limits of XVII-400 of Appendix XVII of the ASME Code, Section III.
(ii)	Same as above.

6. Materials, Quality Control, and Special Construction Techniques

The specified materials of construction and quality control programs are acceptable if in accordance with the applicable code or standard as indicated in Section I.6 of this plan.

Special construction techniques, if any, are treated on a case-by-case basis.

7. Testing and In-service Surveillance Requirements

Each BWR Mark III containment drywell should be subjected to a structural proof test. Such a test is acceptable if in accordance with the following:

a. The drywell should be subjected to an acceptance test that increases the drywell internal pressure in three or more approximately equal pressure increments from atmospheric pressure to at least the design pressure. The drywell should be depressurized in the same number of increments. Measurements should be recorded at atmospheric pressure and at each pressure level of the pressurization and depressurization cycles. At each level, the pressure should be held constant for at least one hour before the deflections and strains are recorded.

b. So that the overall deflection pattern can be determined in prototype drywells, radial deflections should be measured at least at three points along each of at least three meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections should be measured at the lower vent region, at about mid-height and at near the top of the cylindrical wall. The measurement points may be relocated depending on the distribution of stresses and deformations anticipated in each particular design.

c. In prototype drywells only, strain measurements sufficient to permit an evaluation of strain distribution should be recorded at least at two opposing meridians at the following locations on the wall:

   (1) at the bottom of the wall, and

   (2) at mid-height of the wall.

   These strain measurements should be made at least at three positions within the wall section; one at the center and one each near the inner and outer surfaces.

d. In nonprototype drywells, deflection and strain measurements need not be made if strain levels have been correlated with deflection measurements during the acceptance test of a prototype drywell if measured strains and deflections are within the predefined tolerances of their predicted response.

e. Any reliable system of displacement meters, optical devices, strain gauges, or other suitable apparatus may be used for the measurements.

f. If the test pressure drops due to unexpected conditions to or below the next lower pressure level, the entire test sequence should be repeated. Significant deviations from the previous test should be recorded and evaluated.

g. If any significant modifications or repairs are made to the drywell following and because of the initial test, the test should be repeated.

h. A description of the proposed acceptance test and instrumentation requirements should be included in the preliminary safety analysis report.

i. The following information should be submitted prior to the performance of the test:

(i) The numerical values of the predicted responses of the structure which will be measured.

(ii) The tolerances to be permitted on the predicted responses.

(iii) The bases on which the predicted responses and the tolerances thereon were established.

i. The following information should be included in the final test report:

(i) A description of the actual test and instrumentation.

(ii) A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and strains.

(iii) An evaluation of the accuracy of the measurements.

(iv) An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures.

(v) A discussion of the calculated safety margin provided by the structure as deduced from the test results.

For steel linear supports of the reactor coolant system, testing and in-service surveillance requirements are acceptable if in accordance with Subsection NF of the ASME Section III Code, Division I.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. Description of the Internal Structures

After each structure and its functional characteristics are identified, information on similar structures of previously licensed applications is obtained for reference. Such information, which is available in safety analysis reports and amendments of licensed plants enables identification of differences for the case under review which require additional scrutiny. New or unique features that have not been used in the past are of particular interest. The information furnished in the SAR is reviewed for sufficiency in accordance with the "Standard Format..." Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided in the SAR. Any additional required information is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications is checked against the list in Section II.2 of this plan. The reviewer assures himself that the applicable edition and stated effective addenda are utilized.

3. Loads and loading Combinations

The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Any deviations from the acceptance critieria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant for further consideration.

4. Design and Analysis Procedures
The reviewer familiarizes himself with the design and analysis procedures that are generally utilized for the type of structure being reviewed. Since the assumptions made on the expected behavior of the structure and its various elements under loads may be significant, the reviewer determines that they are conservative. The behavior of the structure under various loads and the manner in which these loads are treated in conjunction with other coexistent loads, are reviewed to establish compliance with procedures delineated in Section II.4 of this plan.

5. Structural Acceptance Criteria
The limits on allowable stresses and strains in the concrete, reinforcement, structural steel, etc., are compared with those specified in Section II.5 of this plan. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification provided to show that the functional integrity of the structure will not be affected is evaluated. If such justification is not acceptable, a request for the required additional justification and bases is made.

6. Materials, Quality Control, and Special Construction Techniques
The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that specified in Section II.6 of this plan. If a new material not used in prior license applications is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the structure.

7. Testing and In-service Surveillance Requirements
Procedures for the structural test of the BWR Mark III containment drywell are reviewed and compared with the procedures described in Section II.7 of this plan. Any other proposed testing andin-service surveillance programs are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, and with codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4."

V. <u>REFERENCES</u>

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, "Requirements for Component Supports," American Society of Mechanical Engineers.

2. ACI 318-1971, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

3. AISC, "Specification for Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction (1969).

4. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard Code for Concrete Reactor Vessels and Containments," issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

6. Regulatory Guide 1.55, "Concrete Placement in Category I Structures."

7. ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Draft 3, Revision 1, January 1974, American National Standards Institute.

8. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

9. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."

10. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena.

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to all seismic Category I structures and other safety-related structures that may not be classified as seismic Category I, other than the containment and its interior structures, are reviewed.

1. Description of the Structures

The descriptive information, including plans and sections of each structure, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon for the structure to perform its safety-related function. Also reviewed is the relationship between adjacent structures including the separation provided or structural ties, if any. Among the major plant structures that are reviewed, together with the descriptive information reviewed for each, are the following:

a. Containment Enclosure Building

The containment enclosure building, which may surround all or part of the primary concrete or steel containment structure, is primarily intended to reduce leakage during and after a loss-of-coolant accident (LOCA) from within the containment. Concrete enclosure buildings also protect the primary containment, which may be of steel or concrete, from outside hazards.

The enclosure building is usually either a concrete structure or a structural steel and metal siding building.

Where it is a concrete structure, it usually has the geometry of the containment and, as applicable, the descriptive information reviewed is similar to that of a concrete containment as contained in Section I.1 of Standard Review Plan 3.8.1.

Where it is a structural steel and metal siding building, the following items are reviewed: general arrangement of the building including its foundations, wall, and roof; any bracing and lateral ties provided for the stability of the building; the roof supports which may bear on the dome of the containment; and major corner and siding joint connections.

b. Auxiliary Building

The auxiliary building, which is usually adjacent to the containment and which may be shared by the two containments in 2-unit plants, is usually of reinforced concrete and structural steel construction. The general arrangement of the structural walls, columns, floors, roof, and any removable sections, is reviewed.

c. Fuel Storage Building

The fuel storage building, which may be independent or part of the auxiliary building, is also of reinforced concrete and structural steel. It houses the new fuel storage area and the spent fuel pool. In addition to the information reviewed for the auxiliary building, the general arrangement of the spent fuel pool is reviewed including its foundations and walls.

d. Control Building

The control room is located in most plants within the auxiliary building. However, where it is located in a separate building, usually called the control building, the building is reviewed as a separate structure. To provide missile protection and shielding, this building is usually of reinforced concrete and the descriptive information reviewed is similar to that reviewed for the auxiliary building.

e. Diesel Generator Building

The emergency diesel generators are, in some plants, located within the auxiliary building. However, they may also be located in a separate building called the diesel generator building. Again, this is usually a reinforced concrete structure and the descriptive information reviewed is similar to that reviewed for the auxiliary building.

f. Other Structures

In most plants, there are several miscellaneous seismic Category I structures and other structures that may be safety-related but, because of other design provision, may not be classified as seismic Category I. These structures are usually either of reinforced concrete or structural steel, or a combination thereof. The descriptive information reviewed for such structures is similar to that reviewed for the auxiliary building. Among such structures are: pipe and electrical conduit tunnels, waste storage facilities, stacks, intake structures, pumping stations, and cooling towers.

Further, the reviewer may encounter special safety-related structures such as emergency cooling water tunnels, embankments, concrete dams, and water wells. Such structures are reviewed on a case-by-case basis. The descriptive information provided is reviewed to understand the structural behavior of these structures, specifically during seismic events and plant process conditions during which such structures are required to remain functional.

2. Applicable Codes, Standards, and Specifications

   Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Cateogory I structures, is reviewed.

3. Loads and Load Combinations

   Information pertaining to the applicable design loads and various combinations thereof, is reviewed. The loads normally applicable to seismic Category I structures include the following:

   a. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those in spent fuel pools.
   b. Those loads to be sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) and the design wind specified for the plant.
   c. Those loads to be sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) and the design tornado specified for the plant.
   d. Those loads to be sustained during abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of high-energy piping. Loads induced by such an accident may include elevated temperatures and pressures within or across compartments, and possibly jet impingement and impact forces associated with such ruptures.

   The various combinations of the above loads that are normally postulated and reviewed include normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; and normal operating loads with extreme environmental and abnormal loads.

   The loads and load combinations described above are generally applicable to all types of structures. However, other site-related loads might also be applicable. Such loads, which are not normally combined with abnormal loads, include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures

   The design and analysis procedures utilized for Category I structures are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code (Ref. 1) for concrete structures and with the AISC Specifications (Ref. 2) for steel structures, including the following areas:

   a. General assumptions on boundary conditions.
   b. The expected behavior under loads and the methods by which vertical and lateral loads and forces are transmitted from the various elements to their supports and eventually to the foundation of the structure.
   c. The computer programs that are utilized.

Any new or unique procedures used in the design and analysis are reviewed on a case-by-case basis.

5. Structural Acceptance Criteria

   The design limits imposed on the various parameters that serve to quantify the structural behavior of each structure and its components are reviewed, specifically with respect to stresses, strains, gross deformations, and factors of safety against structural failure. For each load combination specified, the specified allowable limits are compared with the acceptable limits delineated in Section II.5 of this plan.

6. Materials, Quality Control, and Special Construction Techniques

   Information on the materials that are used in the construction of Category I structures is reviewed. Among the major materials of construction that are reviewed are the concrete ingredients, the reinforcing bars and splices, and the structural steel and anchors.

   The quality control program that is proposed for the fabrication and construction of Category I structures is reviewed including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

   Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed structure.

   In addition, the information contained in items a, c, and d of Section I.6 of Standard Review Plan 3.8.3, is also reviewed.

7. Testing and In-service Surveillance Programs

   If applicable, any post-construction testing and in-service surveillance programs are reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review, described in Section I of this plan are as follows:

1. Description of the Structure

   The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.4.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref 3).

   Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated at the application acceptance review. New or unique design features that are not specifically covered in the "Standard Format...," require a more detailed review. The reviewer determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique features.

2. Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is contained in Section II.2 of Standard Review Plan 3.8.3.

3. Loads and Load Combinations

The specified loads and load combinations are acceptable if found to be in accordance with the following:

Loads, Definitions, and Nomenclature

All the major loads to be encountered or to be postulated in a nuclear power plant are listed below. All the loads listed, however, are not necessarily applicable to all the structures and their elements. Loads and the applicable load combinations for which each structure has to be designed will depend on the conditions to which that particular structure may be subjected.

Normal loads, which are those loads to be encountered during normal plant operation and shutdown, include:

- D --- Dead loads or their related internal moments and forces including any permanent equipment loads and hydrostatic loads.
- L --- Live loads or their related internal moments and forces including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.
- T_o --- Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition,
- R_o --- Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition,

Severe environmental loads include:

- E --- Loads generated by the operating basis earthquake.
- W --- Loads generated by the design wind specified for the plant.

Extreme environmental loads include:

- E' --- Loads generated by the safe shutdown earthquake.
- W_t --- Loads generated by the design tornado specified for the plant. Tornado loads include loads due to the tornado wind pressure, the tornado-created differential pressure, and to tornado-generated missiles.

Abnormal loads, which are those loads generated by a postulated high-energy pipe break accident, include:

- P_a --- Pressure equivalent static load within or across a compartment generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.
- T_a --- Thermal loads under thermal conditions generated by the postulated break and including T_o.

R_a --- Pipe reactions under thermal conditions generated by the postulated break and including R_o.

Y_r --- Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j --- Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m --- Missile impact equivalent static load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Load Combinations for Concrete Structures

For concrete structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the working stress design (WSD) method or the strength design method may be used.

(i) If the WSD method is used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

(3) $D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(3a) $D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent should be checked.

(ii) If the strength design method is used, the following load combinations should be considered:

(1) $1.4\ D + 1.7\ L$

(2) $1.4\ D + 1.7\ L + 1.9\ E$

(3) $1.4\ D + 1.7\ L + 1.7\ W$

If thermal stresses due to T_o and R_o are present the following combinations should also be considered:

(1b) (0.75) $(1.4\ D + 1.7\ L + 1.7\ T_o + 1.7\ R_o)$

(2b) (0.75) $(1.4\ D + 1.7\ L + 1.9\ E + 1.7\ T_o + 1.7\ R_o)$

(3b) (0.75) $(1.4\ D + 1.7\ L + 1.7\ W + 1.7\ T_o + 1.7\ R_o)$

Both cases of L having its full value or being completely absent should be checked. In addition, the following combinations should be considered:

(2b') $1.2\ D + 1.9\ E$

(3b') $1.2\ D + 1.7\ W$

Where soil and hydrostatic pressures are present, in addition to all the above combinations where they have been included in L and D respectively, the requirements of Sections 9.3.4 and 9.3.5 of ACI-318-71 (Ref. 1) should also be satisfied.

b. For factored load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental conditions, the strength design method should be used and the following load combinations should be considered.

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of Section II.5 of this plan should be satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strength capacities may be exceeded provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

Load Combinations for Steel Structures

For steel structures, the load combinations are acceptable if found in accordance with the following:

a. For service load conditions, either the elastic working stress design methods of Part 1 of the AISC specifications, or the plastic design methods of Part 2 of the AISC specifications, may be used.

(i) If the elastic working stress design methods are used, the following load combinations should be considered:

(1) $D + L$

(2) $D + L + E$

(3) $D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1a) $D + L + T_o + R_o$

(2a) $D + L + T_o + R_o + E$

(3a) $D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent should be checked.

(ii) If plastic design methods are used, the following load combinations should be considered:

(1) $1.7\ D + 1.7\ L$

(2) $1.7\ D + 1.7\ L + 1.7\ E$

(3) $1.7\ D + 1.7\ L + 1.7\ W$

If thermal stresses due to T_o and R_o are present, the following combinations should also be considered:

(1b) $1.3\ (D + L + T_o + R_o)$

(2b) $1.3\ (D + L + E + T_o + R_o)$

(3b) $1.3\ (D + L + W + T_o + R_o)$

Both cases of L having its full value or being completely absent should be checked.

b. For factored load conditions, the following load combinations should be considered:

(i) If elastic working stress design methods are used:

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + P_a$

(7) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$

(8) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$

(ii) If plastic design methods are used:

(4) $D + L + T_o + R_o + E'$

(5) $D + L + T_o + R_o + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j + Y_r + Y_m) + E'$

In the above factored load combinations, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of Section II.5 of this plan should be first satisfied without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strengths may be exceeded provided there will be no loss of function of any safety-related system.

4. Design and Analysis Procedures

The design and analysis procedures utilized for Category I structures, including assumptions on boundary conditions and expected behavior under loads, are acceptable if found in accordance with the following:

a. For concrete structures, the procedures are in accordance with ACI-318-71, "Building Code Requirements for Reinforced Concrete," (Ref. 1).

b. For steel structures, the procedures are in accordance with the AISC "Specification...," (Ref. 2).

Computer programs are acceptable if the validation provided is found in accordance with procedures delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

For each of the loading combinations delineated in Section II.3 of this plan, the following defines the allowable limits which constitute the structural acceptance criteria.

In Combinations for Concrete	Limit
a(i)1, 2, 3	S(1)
a(i)1a, 2a, 3a.	1.3 S
a(ii)1, 2, 3.	U(2)
a(ii)1b, 2b, 3b	U
a(ii)2b', 3b'	U
(b)4, 5, 6, 7, 8.	U

In Combinations for Steel	Limit
a(i)1, 2, 3	S
a(i)1a, 2a, 3a.	1.5 S
a(ii)1, 2, 3.	Y(3)
a(ii)1b, 2b, 3b	Y
b(i)4, 5, 6, 7(4)	1.6 S
b(i)8(4).	1.7 S
b(ii)4, 5, 6, 7, 8.	.9 Y

NOTES

(1) S --- For concrete structures, S is the required section strength based on the working stress design method and the allowable stresses defined in Section 8.10 of ACI-318-71.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

The 33% increase in allowable stresses for concrete and steel due to seismic or wind loadings is not permitted.

(2) U --- For concrete structures, U is the section strength required to resist design loads based on the strength design methods described in ACI-318-71.

(3) Y --- For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

(4) --- For these two combinations, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

6. <u>Materials, Quality Control, and Special Construction Techniques</u>
For Category I structures outside the containment, the acceptance criteria for materials, quality control, and any special construction techniques are in accordance with the codes and standards indicated in Section I.6 of Standard Review Plan 3.8.3, as applicable.

7. <u>Testing and Inservice Surveillance Requirements</u>
At present there are no special testing or in-service surveillance requirements for Category I structures outside the containment. However, where some requirements become necessary for special structures, such requirements are reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. <u>Description of the Structures</u>
After the type of structure and its functional characteristics are identified, information on similar and previously licensed plants is obtained for reference. Such information, which is available in safety analysis reports and amendments of previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format...,"Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. <u>Applicable Codes, Standards, and Specifications</u>
The list of codes, standards, guides, and specifications is compared with the list referenced in Section II.2 of this plan. The reviewer assures himself that the appropriate code or guide is utilized and that the applicable edition and stated effective addenda are acceptable.

3. Loads and Load Combinations

The reviewer verifies that the loads and load combinations are as conservative as those specified in Section II.3 of this plan. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant.

4. Design and Analysis Procedures

The reviewer assures himself that for the design and analysis procedures, the applicant is utilizing the ACI-318-71 Code and the AISC Specifications for concrete and steel structures, respectively.

Any computer programs that are utilized in the design and analysis of the structure are reviewed to verify their validity in accordance with the acceptance criteria delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

The limits on allowable stresses and strains in the concrete, reinforcement, and structural steel are compared with the corresponding allowable stresses specified in Section II.5 of this plan. If the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification that the structural integrity of the structure will not be affected is evaluated. If such justification is determined to be inadequate, the proposed deviations are identified and transmitted to the applicant with a request for the required additional justification and bases.

6. Materials, Quality Control, and Special Construction Techniques

The materials, quality control procedures, and any special construction techniques are compared with those referenced in Section II.6 of this plan. If a new material not used in prior licensed cases is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control procedures or construction techniques are evaluated to assure that there will be no degradation of structural quality that might affect the structural integrity of the structure.

7. Testing and In-service Surveillance Programs

Any testing and in-service surveillance programs are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. ACI-318-71, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

2. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," American Institute of Steel Construction (1969).

3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.8.5 FOUNDATIONS

REVIEW RESPONSIBILITIES

Primary - Structural Engineering Branch (SEB)

Secondary - None

I. AREAS OF REVIEW

The following areas relating to the foundations of all seismic Category I structures and other safety-related structures are reviewed.

1. Description of the Foundations

The descriptive information, including plans and sections of each foundation, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the foundation function. Also reviewed is the relationship between adjacent foundations, including the methods of separation provided where such separation is utilized to minimize seismic interaction between the buildings. In particular, the type of foundation is identified and its structural characteristics are examined. Among the various types of foundations that are reviewed are mat-foundations and footings, including individual column footings, combined footings supporting more than one column, and wall footings supporting bearing walls.

Other types of foundations that may also be utilized are pile foundations, caisson foundations, combinations of footings, retaining walls, abutments, and rock anchor systems. These foundation types are reviewed on a case-by-case basis.

The major plant Category I foundations that are reviewed, together with the descriptive information reviewed for each, are listed below:

a. Containment Structure Foundation

The most commonly used type of foundation for both concrete and steel containments is a mat foundation, where a flat thick slab supports the containment, its interior structures, and a shield building surrounding the containment, if any. For some PWR containments the base mat has a central depression forming the reactor cavity. The general arrangement of the containment base slab is reviewed as described in Section I.1 of Standard Review Plan 3.8.1, with particular emphasis on methods of transferring horizontal shears, such as those seismically-induced, to the foundation media. Where shear keys are utilized for such purposes, the general arrangement of the keys is reviewed. Where waterproofing membranes are utilized, their effect on the shear resistance of the foundation is reviewed. In prestressed concrete containments, where a tendon inspection gallery is utilized, arrangement of the gallery and means of either isolating it from the remainder of the base slab or of relying upon it for some function such as resisting shears, are reviewed.

b. Containment Enclosure Building Foundation

Where the containment enclosure building is constructed of reinforced concrete, it is usually supported on the same mat foundation supporting the containment.

Where it is a structural steel and metal siding building, it may surround only the exposed portion of the containment. In such a situation, the enclosure building columns are founded on individual or combined footings at grade level, on the roof of buildings adjacent to or surrounding the containment, on the dome of the containment, and possibly on brackets anchored on the exterior face of the cylindrical wall of the containment. General arrangement of such foundations is reviewed with particular emphasis on methods of isolating the enclosure building from other buildings in a lateral direction, where this is preferable to minimize seismic interaction.

c. Auxiliary Building Foundation

The auxiliary building foundation is typically of a mat type, particularly where the supporting foundation media is soil.

The general arrangement of the foundation is reviewed, again with particular emphasis on methods of transferring loads from the structure to the foundation media.

d. Other Category I Foundations

The foundations for other Category I structures, which may be one or a combination of several foundation types, are reviewed to an extent similar to that of the containment foundation. Among Category I structures the foundations of which are so reviewed, are: fuel storage buildings, control buildings, diesel generator buildings, intake structures, and cooling towers. Also reviewed are foundations of safety-related structures which, because of other design provisions, are not classified as seismic Category I.

2. Applicable Codes, Standards, and Specifications

Information pertaining to design codes, standards, specifications, regulations, general design criteria and regulatory guides, and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Category I foundations is reviewed.

3. Loads and Load Combinations

Information pertaining to the applicable design loads and their various combinations is reviewed. The loads normally applicable to Category I foundations are the same as those applicable to the structures which the foundations support. These loads are described in Section I.3 of Standard Review Plan 3.8.4.

4. Design and Analysis Procedures

The design and analysis procedures utilized for Category I foundations are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code (Ref. 1) for concrete structures, and with the AISC Specifications (Ref. 2) for steel structures, including the following areas:

a. The assumptions made on boundary conditions and the expected behavior of each foundation when subjected to the various design loads.
b. The methods by which lateral loads and forces and overturning moments thereof are transmitted from the structure to the foundation media. Such forces are mainly generated by the environmental and abnormal plant conditions such as wind, tornadoes, earthquakes, and pipe ruptures. Methods of determining overturning moments due to the three components of the earthquake are also reviewed.
c. The computer programs that are utilized in the design and analysis of foundations.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of each foundation are reviewed with emphasis on the extent of compliance with the ACI-318-71 Code for concrete structures, specifically with respect to stresses, strains, deformations, and factors of safety against overturning and sliding, as applicable.

6. Materials, Quality Control, and Special Construction Techniques

Information on the materials that are used in the construction of Category I foundations is reviewed. Among the major materials of construction that are reviewed are the following:

a. The concrete ingredients.
b. The reinforcing bars and mechanical splices.
c. The structural steel.
d. Rock anchors, including any prestressing system.

The quality control program that is proposed for the fabrication and construction of Category I foundations is reviewed, including the following: nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed foundation.

In addition, the information contained in items a, c, and d of Section I.6 of Standard Review Plan 3.8.3, is also reviewed.

7. Testing and In-service Surveillance Programs

If applicable, any post-construction testing and in-service surveillance programs for foundations, such as monitoring potential settlements and displacements, are reviewed on a case-by-case basis.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Description of the Foundation

   The descriptive information in the SAR is considered acceptable if it meets the minimum requirements set forth in Section 3.8.5.1 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (Ref. 4).

   Deficient areas of descriptive information are identified by the reviewer and a request for additional information is initiated, at the application acceptance review, if possible. New or unique design features that are not specifically covered in the "Standard Format...," require a more detailed review. The reviewer determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique foundation features.

2. Applicable Codes, Standards, and Specifications

   The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Category I foundations are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is contained in Section II.2 of the Standard Review Plan 3.8.3. In addition, the documents listed in Section II.2 of Standard Review Plan 3.8.1 are acceptable for the containment foundation.

3. Loads and Load Combinations

   The specified loads and load combinations utilized in the design of Category I foundations are acceptable if found to be in accordance with those combinations referenced in Section II.3 of Standard Review Plan 3.8.1 for the containment foundation, and with those combinations listed in Section II.3 of Standard Review Plan 3.8.4 for all other Category I foundations.

   In addition to the load combinations referenced above, the combinations utilized to check against sliding and overturning due to earthquakes, winds, and tornadoes, and against floatation due to floods, are found acceptable if in accordance with the following:

   a. D + H + E
   b. D + H + W
   c. D + H + E'
   d. D + H + W_t
   e. D + F'

   where D, E, W, E', W_t are as defined in Standard Review Plan 3.8.4, H is the lateral earth pressure, and F' is the bouyant force of the design basis flood. Justification should be provided for including live loads or portions thereof in these combinations.

4. Design and Analysis Procedures

   The design and analysis procedures utilized for Category I foundations are acceptable if found in accordance with the following:

a. For Category I concrete foundations other than the containment foundations, the procedures are in accordance with the ACI-318-71, "Building Code Requirements for Reinforced Concrete," (Ref. 1).
b. For Category I steel foundations, the procedures are in accordance with the AISC "Specifications...," (Ref. 2).
c. For the containment foundation, the design and analysis procedures referenced in Section II.4 of Standard Review Plan 3.8.1 are acceptable.

For determining the overturning moment due to an earthquake, the three components of the earthquake should be combined in accordance with methods described in Standard Review Plan 3.7.2. Computer programs are acceptable if the validation provided is found in accordance with procedures delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. Structural Acceptance Criteria

For each of the loading combinations referenced in Section II.3 of this plan, the allowable limits which constitute the acceptance criteria are referenced in Section II.5 of Standard Review Plan 3.8.1 for the containment foundation, and are listed in Section II.5 of Standard Review Plan 3.8.4 for all other foundations. In addition, for the five additional load combinations delineated in Section II.3 of this plan, the factors of safety against overturning, sliding, and floatation are acceptable if found in accordance with the following:

	Minimum Factors of Safety		
For Combination	Overturning	Sliding	Floatation
a. ----------------------------	1.5	1.5	--
b. ----------------------------	1.5	1.5	--
c. ----------------------------	1.10	1.1	--
d. ----------------------------	1.10	1.1	--
e. ----------------------------	--	--	1.1

6. Materials, Quality Control, and Special Construction Techniques

For the containment foundation, the acceptance criteria for materials, quality control, and any special construction techniques are referenced in Section II.6 of Standard Review Plan 3.8.1. For all other Category I foundations, the acceptance criteria are similar to those referenced in Section II.6 of Standard Review Plan 3.8.4.

7. Testing and In-service Surveillance Requirements

At present there are no special testing or in-service surveillance requirements for Category I foundations other than those required for the containment foundation, which are covered in Section II.7 of Standard Review Plan 3.8.1. However, should some requirements become necessary for special foundations, they will be reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below, as may be appropriate for a particular case.

1. <u>Description of the Foundations</u>

After the type of foundation and its structural characteristics are identified, information on similar and previously licensed plants is obtained for reference. Such information, which is available in safety analysis reports and amendments of license applications enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are examined in greater detail. The information furnished in the SAR is reviewed for sufficiency in accordance with the "Standard Format...," Revision 2. A decision is then made with regard to the sufficiency of the descriptive information provided. Any additional required information is requested from the applicant at an early stage of the review process.

2. <u>Applicable Codes, Standards, and Specifications</u>

The list of codes, standards, guides, and specifications is compared with the list referenced in Section II.2 of this plan. The reviewer assures himself that the appropriate code or guide is utilized and that the applicable edition and stated effective addenda are acceptable.

3. <u>Loads and Load Combinations</u>

The reviewer verifies that the loads and load combinations are as conservative as those referenced and specified in Section II.3 of this plan. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable and transmitted to the applicant.

4. <u>Design and Analysis Procedures</u>

The reviewer assures himself that for the design and analysis procedures, the applicant is utilizing the procedures in the applicable code as delineated in Section II.4 of this plan.

Any computer programs that are utilized in the design and analysis of the foundation are reviewed to verify their validity in accordance with the acceptance criteria delineated in Section II.4.e of Standard Review Plan 3.8.1.

5. <u>Structural Acceptance Criteria</u>

The limits on allowable stresses and strains in the concrete, reinforcement, and structural steel, and on factors of safety for overturning, sliding, and floatation are compared with the corresponding allowable values specified in Section II.5 of this plan. If the applicant proposes to deviate from these limits for some of the load combinations and at some localized points, the justification that the structural integrity of the foundation will not be affected is evaluated. If such justification is determined to be inadequate, a request for the required additional justification and bases is made.

6. <u>Materials, Quality Control, and Special Construction Techniques</u>

The materials, quality control procedures, and any special construction techniques are compared with those referenced in Section II.6 of this plan. If a new material not used in prior licensed cases is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, any new quality control procedures or construction techniques are evaluated in detail to assure that there will be no degradation of structural quality that might affect the structural integrity of the foundation.

7. Testing and In-service Surveillance Programs
For the containment foundation, testing and in-service surveillance programs are reviewed in accordance with Section II.7 of Standard Review Plan 3.8.1 for concrete containments. Any testing and in-service surveillance programs for other foundations are reviewed on a case-by-case basis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report:

"The criteria used in the analysis, design, and construction of all the plant Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff.

"The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4."

V. REFERENCES

1. ACI-318-71, "Building Code Requirements for Reinforced Concrete," American Concrete Institute (1971).

2. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Building," American Institute of Steel Construction (1969).

3. ASME Boiler and Pressure Vessel Code, Section III, Division 2 (ACI-359), "Proposed Standard for Concrete Reactor Vessels and Containments, issued for interim trial use and comment, April 1973, American Society of Mechanical Engineers.

4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2 (in preparation).

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

Information concerning design transients and methods of analysis for seismic Category I components, including both those designated as Class 1, 2, 3, or CS under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code"), and component supports, reactor internals, and other components not covered by the Code, is given in the applicant's safety analysis report (SAR) and is reviewed by the MEB. Certain aspects of dynamic system analysis methods are discussed in Standard Review Plan 3.9.2 as well as in this plan. The following specific subjects are reviewed under this plan:

1. Transients which are used in the design and fatigue analyses of all Code Class 1 and CS components, and of component supports and reactor internals. The Reactor Systems Branch confirms the acceptability of the listed design transients and the number of cycles and events expected over the service lifetime of the plant. The Structural Engineering Branch confirms the number of seismic cyclic loadings acceptable for design. (For design of other non-Code components, see Standard Review Plan 3.9.3.)

2. Descriptions of all computer programs which will be used in analyses of Code and non-Code items listed in this plan.

3. Descriptions of any experimental stress analysis programs which will be used in lieu of theoretical stress analyses.

4. Descriptions of the analysis methods which will be used if the applicant elects to use inelastic stress analysis methods in the design of any of the above-noted components.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are as follows:

1. The applicant shall provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and CS components, and of component supports and reactor internals within the reactor coolant pressure boundary. The number of events for each transient shall be included along with assurance that the number of load and stress cycles per event is properly taken into account. All design transients such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic events, etc., that are contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary shall be specified. All transients or combinations of transients shall be categorized with respect to the plant operating conditions identified as "normal," "upset," "emergency," "faulted," or "testing" and defined in Reference 4.

   The section of the applicant's SAR which pertains to design transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions resulting from those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgement and experience. Some guidance on the selection of these transients can be found in Reference 5. The design transients, plant and component conditions, and loading combinations must provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant to satisfy, in part, the requirements of References 1 and 2.

2. A list of computer programs that will be used in dynamic and static analyses to determine the structural and function integrity of seismic Category I Code and non-Code items and the analyses to determine stresses shall be provided, including a brief description of each program and the extent of its application. The design control measures, as required by Appendix B of 10 CFR Part 50, that will be employed to demonstrate the applicability and validity of these computer programs should meet one of the following criteria:
   a. The computer program is recognized and widely used, with a sufficient history of successful use to justify its applicability and validity without further demonstration by the applicant. The dated program version that will be used, the software or operating system, and the hardware configuration must be specified to be accepted by virtue of its history of use.

b. The computer program solutions to a series of test problems with accepted results have been demonstrated to be substantially identical to those obtained by a similar program which meets the criteria of (a) above. The test problems shall be demonstrated to be similar to or with the range of applicability for the problems analyzed by the computer program to justify acceptance of the program.

c. The program solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results published in technical literature. The test problems shall be demonstrated to be similar to the problems analyzed to justify acceptance of the program.

A summary comparison of the results obtained from the use of each computer program under options (b) or (c) above with either the results derived from a similar program meeting option (a), or a previously approved computer program, or results from the test problems of option (c) shall be provided. Include typical static and dynamic response loading, stress, etc., comparisons, preferably in graphical form.

3. If experimental stress analysis methods are used in lieu of analytical methods, for any seismic Category I Code or non-Code items, the section of the SAR discussing the experimental stress analysis methods will be acceptable if the information provided meets the provisions of Appendix II of Reference 4, and as in the case of analytical methods, if the information provided is sufficiently detailed to show the validity of the design to meet the provisions of the Code-required "Design Specifications."

4. When inelastic stress or deformation design limits are specified by the applicant for Code Class 1 and CS components, and for component supports, reactor internals, and other non-Code items, the methods of analysis used to calculate the stresses and deformations resulting from faulted condition loadings shall conform to the methods outlined in Appendix F of Reference 4, subject to deformation constraints discussed in III.4 below. It is acceptable to apply similar limits to Code Class 2 and 3 components provided the analytical methods, applicable criteria, and fabrication procedures of Code Class 1 components are used. Other applicable limits permitted by the Code are acceptable.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. The list of transients and the number of events estimated for each transient presented in the applicant's SAR is compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in II above. Any deviations from previous accepted practice are noted and the applicant is required to justify these deviations. The MEB verifies that each design

transient has been properly categorized with respect to the component operating conditions of design, i.e., "normal," "upset," "emergency," "faulted" and "testing" as defined in Reference 4.

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

2. The information pertaining to computer programs which is presented in the applicant's SAR is reviewed as follows:
   a. The list of programs is evaluated to determine that the applicant has adequately described each program with respect to the type of analysis that is performed and the specific components to which the program is applied.
   b. The design control measures, which are required by 10 CFR Part 50, Appendix B, are reviewed for each program. The procedures outlined in II.2.a, b, or c of this plan must be met for each program. Verification by the applicant that he has met the requirements of at least one of the above paragraphs is acceptable.
   c. The summary comparison of the results obtained from the use of each program which is not recognized and widely used (See II.2 of this plan) with either the results derived from a similar recognized and widely used program, a previously approved computer program, or results from test problems is reviewed and evaluated. Numerical results so derived should compare favorably enough to provide confidence in the validity of the program.

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

3. If the applicant elects to use experimental stress analysis techniques in lieu of theoretical stress analyses, sufficient information must be presented in the SAR to demonstrate that the requirements of Appendix II to Reference 4, as they apply to the conditions set forth in the "Design Specifications" have been met.

4. If the applicant employs an inelastic method of analysis to evaluate the design of safety-related Code or non-Code items for the faulted plant condition (NB-3225 and Appendix F of Reference 4), the review covers the following points:
   a. The applicant must demonstrate that the stress-strain relationship for component materials that will be used in the analysis is valid. The ultimate strength values at service temperature must be justified.
   b. The analytical procedures to be used in the analysis are reviewed to determine the validity of the analysis. If a computer program is used, the applicable requirements of II.2 above shall be met.

c. If elastic, elastic-inelastic, or limit analysis methods are used for components in conjunction with elastic or inelastic system analyses, the basis upon which these procedures are used are reviewed. The applicant shall provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the methods used for the system analysis are based. (For example, current small deformation methods of analysis typically tend to have acceptable effective strain limits in the range of 1/2 to 1-1/2 percent and large deformation methods 10 to 20 percent.)

Any deviations that have not been justified to the satisfaction of the staff are identified and the finding is transmitted to the applicant with a request that, unless conformance with the MEB acceptance criteria is agreed upon, additional technical justification be submitted.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The criteria used in the methods of analysis that the applicant has employed in the design of all seismic Category I ASME Code Class 1, 2, 3, and CS components, component supports, reactor internals, and other non-Code items are in conformance with established technical positions and criteria which are acceptable to the Regulatory staff.

"The use of these criteria in defining the applicable design transients, computer codes used in analyses, analytical methods, and experimental stress analysis methods provides assurance that the stresses, strains, and displacements calculated for the above-noted items are as accurate as the current state-of-the-art permits and are adequate for the design of these items."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion 14, "Reactor Coolant Pressure Boundary."

2. 10 CFR Part 50, Appendix A, Criterion 15, "Reactor Coolant System Design."

3. 10 CFR Part 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants."

4. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

5. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Reactors."

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.2 DYNAMIC TESTING AND ANALYSIS OF MECHANICAL SYSTEMS AND COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

MEB reviews the criteria, testing procedures, and dynamic analyses employed to assure the structural and functional integrity of piping systems, mechanical equipment, and reactor internals under vibratory loadings, including those due to fluid flow and postulated seismic events. The staff review covers the following specific areas:

1. Preoperational piping vibrational and dynamic effects testing should be conducted during startup functional testing on all safety-related piping systems designated as Class 1, 2, or 3 under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code"), and the supports and restraints for these systems. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under operational transient conditions anticipated during service. The test program description should include a list of different flow modes, a list of selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessvie vibration occurs.

2. Seismic qualification testing of safety-related mechanical equipment is required to assure its ability to function during and after a postulated seismic occurrence. At the construction permit (CP) stage, the staff review covers the following specific areas:
   a. The criteria for seismic qualification such as the deciding factors for choosing test or analysis, the considerations defining the input motion, and the steps to demonstrate adequacy of the seismic qualification program.
   b. The methods and procedures used to assure seismic Category I mechanical equipment operability during and after the safe shutdown earthquake (SSE), and to assure structural and functional integrity of the equipment after several occurrences of

the operating basis earthquake. Included are mechanical equipment such as fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays.

c. The methods and procedures of analysis or testing for the supports for the seismic Category I mechanical equipment listed above, and the procedures used to account for the possible amplification of loads (amplitude and frequency content) under seismic conditions.

At the operating license (OL) stage, the staff reviews the results of tests and analyses to assure the proper implementation of the criteria established in the CP review, and to demonstrate adequate seismic qualification.

3. Dynamic responses of structural components within the reactor vessel caused by operational flow transients should be analyzed for prototype (first of a design) reactors. Generally, this analysis is not required for non-prototypes except that segments of an analysis may be necessary if there are substantial deviations from the prototype internals design. The purpose of this analysis is to predict the vibration behavior of the components, so that the input forcing functions and the level of response can be estimated before conducting the methods of analysis, the specific locations for calculated responses, the considerations in defining the mathematical models, the interpretation of analytical results, the acceptance criteria, and the methods of verifying predictions via tests. If the reactor internal structures are not a prototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

4. Flow-induced preoperational vibration testing of reactor internals should be conducted during the startup functional test program. The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude or structural damage. The test program description should include a list of flow modes, a list of sensor types and locations, a description of test procedures and methods to be used to process and interpret the measured data, a description of the visual inspections to be made, and a comparison of the test results with the analytical predictions. If the reactor internal structures are not a prototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given.

5. Dynamic system analyses should be performed to confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident (LOCA) and the SSE. The staff review covers the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

6. A discussion should be provided which describes the methods to be used to correlate results from the reactor internals preoperational vibration test with the analytical results from dynamic analyses of the reactor internals under operational flow transients.

In addition, test results from previous plants of similar characteristics may be used to verify the mathematical models used for the faulted condition (LOCA and SSE) by comparing such dynamic characteristics as the natural frequencies. The staff review covers the methods to be used for comparison of test and analytical results and for verification of the analytical models.

Computer programs used in the analyses discussed in this plan are reviewed in accordance with Standard Review Plan 3.9.1.

The RSB verifies that (1) the various flow modes to be used to conduct the preoperational vibration test are representative of the operational transients anticipated for the reactor during its service, and (2) the LOCA forcing functions used to conduct the system dynamic analysis are representative of the most adverse LOCA loadings.

II. ACCEPTANCE CRITERIA

To fulfill in part the design requirements for safety-related structures, systems, and components set forth in General Design Criteria 1, 2, 4, 14, and 15, the acceptance criteria for the areas of MEB review are as follows:

1. Preoperational vibrational and dynamic effects testing should be conducted during startup functional testing for safety-related piping classified as Code Class 1, 2, and 3, and for piping and component supports. The purpose of these tests is to confirm that the piping, components, and supports have been designed to withstand the dynamic loadings from operational transient conditions that will be encountered during service, as required by the Code. An acceptable test program to confirm the adequacy of the designs should consist of the following:
   a. A listing of the different flow modes of operation and transients such as pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see Reference 8.) For example, the transients associated with the reactor coolant system heatup tests should include, but not necessarily be limited to:
      (1) Reactor coolant pump start.
      (2) Reactor coolant pump trip.
      (3) Operation of pressure-relieving valves.
   b. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak) criteria that will be used to show that the stress and fatigue limits are within the design levels should be provided.
   c. If vibration is noted beyond the acceptance levels set by the criteria of (b) above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping systems restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level.
2. A test program is required to confirm the ability of all seismic Category I mechanical equipment to function as needed during and after an earthquake of magnitude up to and including the SSE.

a. Analysis without testing is acceptable if structural integrity alone can assure the intended function. When a complete seismic test is impracticable, a combination of test and analysis is acceptable.

b. Equipment should be tested in the operational condition. Loadings simulating those of plant normal operation, such as thermal and flow-induced loadings, if any, should be concurrently superimposed upon the seismic loading. Operability should be verified during and after the test.

c. The characteristics of the seismic input motion should be specified by one of the following:

   (1) Response spectrum.

   (2) Power spectral density function.

   (3) Time history.

   Such characteristics, as derived from the structure or system seismic analysis, should be reprentative of the seismic input motion at the equipment mounting locations.

d. The test input motion should be characterized in the same manner as the seismic input motion, and the conservatism in amplitude and frequency content should be demonstrated.

e. Seismic excitations generally have a broad frequency content. Random vibration input motion should be used in the testing. However, single frequency input, such as sine "beats," may be applicable provided one of the following conditions are met:

   (1) The characteristics of the seismic input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects).

   (2) The anticipated response of the equipment is adequately represented by one mode.

   (3) The test input motion has sufficient intensity and duration to excite all modes to the required amplitudes, such that the testing response spectra will envelope the corresponding response spectra will envelope the corresponding response spectra of the individual modes.

f. The test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

g. Dynamic coupling between the equipment and related sytems, if any, such as connected piping and other mechanical components, should be considered.

h. The fixture design should meet the following requirements:

   (1) Simulate the actual service mounting.

   (2) Cause no extraneous dynamic coupling to the test item.

i. The in situ application of vibratory devices to superimpose the seismic vibratory loadings on a complex active device for operability testing is acceptable if it is shown that a meaningful test can be made in this way.

j. The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc., on a prototype basis.

k. Analyses or tests should be performed for all supports of mechanical equipment to assure their structural capability to withstand seismic excitation. The analytical results must include the following:

   (1) The required input motions to the mounted equipment should be obtained and characterized in the manner as stated in 2.c, above.

   (2) The combined stresses of the support structures should be within the limits of the Code, Subsection NF, "Component Support Structures."

l. Supports should be tested with equipment installed or with an equivalent mass that simulates the equipment dynamic coupling to the support. If the equipment is installed in a nonoperating condition for the support test, the response at the equipment mounting location should be characterized in the manner as stated in 2.c, above. In such a case, the equipment should be tested separately for operability and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response.

m. The requirements of 2.c, 2.d, 2.e, 2.f, and 2.h, above, are applicable when tests are conducted on equipment supports.

3. The following guidelines, in addition to Reference 7, apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs.

   a. The results of vibration calculations for a prototype reactor should consist of the following:

      (1) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, and the response causing the maximum stress amplitude should be calculated.

      (2) The dynamic properties of internal structures, including the natural frequencies, the dominant mode shapes, and the damping factors should be characterized. If analyses are performed on a component structural element basis, the existence of dynamic coupling among component structure elements should be investigated.

      (3) The response characteristics, such as the dependence on hydrodynamic excitation forces, the flow path configuration, coolant recirculation pump frequencies, and the natural frequencies of the internal structures, should be identified.

      (4) Acceptance criteria for allowable responses should be established, as should criteria for the location of vibration sensors. Such criteria should be related to the Code allowable stresses, strains, and limits of deflection that are established to preclude loss of function with respect to the reactor core structures and fuel assemblies.

   b. The forcing functions should account for the effects of transient flow conditions and the frequency content. Acceptable methods for formulating forcing functions for vibration prediction include the following:

(1) Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.

(2) Test-analysis combination method: based on data obtained from plant tests or scaled model tests, (e.g., velocity or pressure distribution data), forcing functions should be formulated which will include the effects of complex flow path configurations and wide variations of pressure distributions.

(3) Response-deduction method: based on a derivation of response characteristics from plant or scaled model test data, forcing functions should be formulated. However, since such functions may not be unique, the computational procedures and the basis for the selection of the representative forcing functions should be described.

c. Acceptable methods of obtaining dynamic responses for vibration predictions are as follows:

(1) Force-response computations are acceptable if the characteristics of the forcing functions are predetermined on a conservative basis and the mathematical model of the reactor internals is appropriately representative of the design.

(2) If the forcing functions are not predetermined, either a special analysis of the response signals measured from reactor internals of similar design may be performed to predict amplitude and modal contributions, or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used.

d. Vibration predictions should be verified by test results. If the test results differ substantially from the predicted response behavior, the vibration analysis should be appropriately modified to improve the agreement with test results and to validate the analytical method as appropriate for predicting responses of the prototype unit, as well as of other units where confirmatory tests are to be conducted.

4. The preoperational vibration test program for the internals of a prototype (first of a design) reactor should conform to the requirements for a prototype test, as specified in Reference 7, including vibration prediction, vibration monitoring, data reduction, and surface inspection. The test program should include, but not necessarily be limited to the following:

a. The vibration testing should be conducted with the fuel elements in the core or with dummy elements which provide equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if it can be demonstrated that testing in this mode is conservative.

b. A brief description of the vibration monitoring instrumentation should be provided, including instrument types and diagrams of locations, which should include the locations having the most severe vibratory motions or having the most effect on safety functions.

c. The planned duration of the test for the normal operation modes to assure that all critical components are subjected to at least 10^7 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 12 days or more will be acceptable.

d. Testing should include all of the different flow modes of normal operation and upset transients.

e. The methods and procedures to be used to process the test data to obtain a meaningful interpretation of the core structure vibration behavior should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and the possible effects on safety functions.

f. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.

g. Visual and nondestructive surface inspections should be performed after the completion of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals, and the equipment to be used for performing such inspections. These inspections should be conducted preferably following the removal of the internals from the reactor vessel. Where removal is not feasible, the inspectons should be performed by means of equipment appropriate for in situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high stress locations, and locations critical to safety functions.

For internals of subsequent reactors that have the same design, size, configuration, and operating conditions as the prototype reactor internals, the preoperational vibration test program should conform to the requirements of a confirmatory test, as specified in Reference 7, which provides an option to choose either monitoring the vibration or conducting a visual inspection after testing.

5. Dynamic system analyses should be performed to confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA and the SSE. Where a substantial separation between the frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings separately.

The most severe dynamic effects from LOCA loadings are generally found to result from a postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating mode.

Mathematical models used for dynamic system analysis for LOCA and SSE effects should include the following:

a. Modeling should include reactor internals and dynamically related piping, pipe supports, and components. Typical diagrams and the basis of modeling should be developed and described.

b. Mathematical models should be representative of system characteristics, such as the flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

c. Any system partitioning and directional decoupling employed in the dynamic system modeling should be justified.
d. The effects of flow upon the mass and flexibility properties of the system should be discussed.

Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions used for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semiempirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions.

The methods and procedures used for dynamic system analyses should be described, including the governing equations of motion and the computational scheme used to derive results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal analysis method may be used for SSE analysis.

The stability of elements in compression, such as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be investigated.

Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports.

The criteria for acceptance of the analytical results are as follows:
a. Deformations should not exceed the allowable limits to assure shutdown functions and adequate passage of core cooling water.
b. Stresses should not exceed the allowable limits of the Code, Subsection NG, "Core Support Structures." The applicable stress limits used should be consistent with those permitted for system components in the analytical stress analyses.
c. The loading combinations should be based on the loads of the faulted condition.

6. Regarding the correlation to be made of tests and analyses of reactor internals, a discussion covering the following items should be provided:
   a. Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for possible verification of the mathematical model used in the analysis.
   b. Comparison of the analytically obtained mode shapes with the shape of measured motion for possible identification of the modal combination or verification of a specific mode.
   c. Comparison of the response amplitude time variation and the frequency content obtained from test and analysis for possible verification of the postulated forcing function.
   d. Comparison of the maximum responses obtained from test and analysis for possible verification of stress levels.
   e. Comparison of the mathematical model used for dynamic system analysis under operational flow transients and under the LOCA or SSE loadings, to note similarities.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

General Design Criteria 1, 2, 4, 14, and 15 state that all structures, system and components important to safety should be designed and tested to assure that safety functions can be performed in the event of operational transients, earthquakes, and LOCA loadings.

Under these guidelines, the staff reviews the treatment of dynamic responses of safety-related piping systems and reactor internal structures by the following procedures:

1. During the CP stage, the staff obtains a commitment from the applicant to conduct a preoperational piping vibrational and dynamic effects test program in accordance with II.1, above.

   During the OL stage, the staff reviews the program and verifies that the acceptance criteria have been met.

2. At the CP stage, the staff reviews the program which the applicant has described in the preliminary safety analysis report (PSAR) for the seismic qualification of all seismic Category I mechanical equipment. The program is measured against the requirements listed in the acceptance criteria section of this review plan. Of particular interest are the proper use of test and analytical procedures. Equipment which is too complex for reliable mathematical modeling should be tested unless the analytical procedures and corresponding design are convincingly conservative. Both the test and the analysis methods are reviewed for assurance that all important modes of response have been excited in tests or considered in analyses. Proper application of test input motions so as to envelop the required input, whether in terms of response spectra, power spectral density, or time history, and in all necessary directions, is verified. The use or treatment of supports is also reviewed.

   At the OL stage, the staff reviews the program again as described by the applicant in the final safety analysis report (FSAR). In addition, the FSAR is reviewed for documentation of successful implementation of the seismic qualification program, including test and analysis results. Also, the acceleration levels used in the tests and in the analyses are reviewed for assurance that they equal or exceed the acceleration at the equipment mounting locations derived from structural response studies of the plant structure as built or as designed.

3. At the CP stage, the applicant should commit to performing an analysis of the vibration of the reactor internal structures if they are designated as a prototype design. A brief description of the methods and procedures to be used for the analysis should be provided.

   At the OL stage, a detailed dynamic analysis should be provided for a prototype design, to be used for vibration prediction prior to the performance of preoperational vibration tests. Acceptance of the analysis is based on the technical soundness of the analytical methods and procedures used and the degree of conformance to the acceptance criteria listed above. In addition, the analysis is verified by correlation with the test results when these are available.

For both CP and OL stages, if the reactor internal structures are not a prototype design, then reference should be made to the reactor which is prototypical of the reactor being reviewed. A brief summary of test and analysis results for the prototype should be given. Alternatively, the information may be contained in another applicable document, such as a topical report, to which reference should be made.

4. At the CP stage, the staff review of the program for preoperational vibration testing of reactor internals for flow-induced vibrations includes the following matters:
   a. The applicant should clarify his intention to perform either a prototype test or a confirmatory test.
   b. If the plant is designated as a prototype, a brief description of the preoperational vibration test program should be provided. The staff review will be based on the conformance of this program to the requirements as listed in II.4, above.
   c. If the plant is not a prototype, the applicant should identify the existing plant of similar design that is the prototype plant. The staff reviews the validity of the designated prototype, including any design difference of reactor internal structures from the prototype plant to verify that any design modifications do not substantially alter the behavior of the flow transients and the response of the reactor internals. Additional detailed analysis, scaled model tests, or installation of some instrumentation during the confirmatory test may be required in order to complete the review. In addition, the applicant should commit to performing the prototype test if adequate test results are not obtained on a timely basis for the designated prototype.

   At the OL stage, the staff review includes the following procedures:
   a. A detailed preoperational vibration test program and the tentative schedule to perform the test are reviewed. If elements of the program differ substantially from the guidelines specified in Reference 7, discussion of the need and justification for the differences should be given.
   b. For a prototype plant, the review covers the acceptability of vibration prediction, the visual surface inspection procedures, the details of instrumentation for vibration monitoring, the methods and procedures to process the test results, and possible supplementary tests, such as component vibration tests, flow tests, and scaled model tests.
   c. For a non-prototype plant, the staff verifies the applicability of the designated prototype, including the design similarity of the reactor internal structures to the prototype. Additional detailed analysis, scaled model tests, or vibration monitoring in the confirmatory tests may be needed in order to complete the review.

5. In the CP stage review of the dynamic analysis of the reactor internals and unbroken loops of the reactor coolant piping under faulted condition loadings, the applicant commits to perform this analysis or identifies the applicable document, generally in form of a topical report, containing the required information. A brief description of the scope and methods of analysis should be provided.

   In the OL review, the staff reviews the detailed information to confirm that an adequate analysis has been made of the capability of reactor internal structures and unbroken loops to withstand dynamic loads from the most severe LOCA and the safe shutdown earthquake. The staff review covers the analytical methods and procedures, the basis of the forcing functions, the mathematical models to represent the dynamic system, and the stability investigations for the core barrel and essential compressive elements. Acceptance of the analysis is based on (1) the technical soundness of the analytical methods used, (2) the degree of conformance to the acceptance criteria listed above, and (3) verification that stresses under the combined loads are within allowable limits of the applicable code and deformations are within the limits set to assure the ability of reactor internal structures and piping to perform needed safety functions.

6. MEB reviews the program which the applicant has committed to implement as part of the preoperational test procedure, principally to correlate the test measurements with the anaytically predicted flow-induced dynamic response of the reactor internals. MEB reviews the applicant's statements in this areas to assure that there is a commitment to submit a report on a timely basis. The report should summarize the analyses and test results so that MEB can review the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The preoperational vibration test program which will be conducted during startup and initial operation on all safety-related piping systems, restraints, components, and component supports classified as ASME Class 1, 2, and 3 is an acceptable program. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis operational transients. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for fulfilling, in part, the requirements of General Design Criterion 15.

"The capability of safety-related mechanical equipment to perform necessary protective actions in the event of a safe shutdown earthquake (SSE) is essential for plant safety. The qualification testing program which will be implemented for seismic Category I mechanical equipment provides adequate assurance that such equipment will function properly under the loads from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-earthquake operation. This program constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2.

"The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the pre-operational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and constitutes an acceptable basis for demonstrating design adequacy of the reactor internals, and satisfies the applicable requirements of General Design Criteria 1 and 4.

"The dynamic system analysis to be performed provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of postulated loss-of-coolant accidents (LOCA) and the safe shutdown earthquake (SSE) and the combined loads of a postulated main steam line rupture and SSE (for a BWR). The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 2 and 4."

For the FSAR, the review should provide justification for a finding similar to that stated above with the phrase "will be implemented" modified to read "has been implemented."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

5. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

7. Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals."

8. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.3 ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

Information is presented in the applicant's safety analysis report (SAR) and is reviewed by the MEB concerning the structural integrity and operability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter "the Code").

The staff review covers the following specific areas:

1. Loading Combinations, Design Transients, and Stress Limits
The design loading combinations (e.g., design loads or anticipated operational loads including design transients in combination with loads calculated to result from postulated accidents and seismic events) specified for Code constructed items designated as Code Class 1, 2, 3 and CS are reviewed to determine that they have been appropriately categorized with respect to "normal," "upset," "emergency," or "faulted" plant conditions. In addition, the staff review determines that the design stress limits and deformation criteria associated with each of the plant operating conditions and appropriate component operating conditions comply with the applicable limits specified in the Code and other criteria. Design stress limits which allow inelastic deformation of Code Class 1, 2, 3 and CS items are evaluated as are the justifications for the proposed design procedures. Piping which is "field run" should be included. Internal parts of components such as valve discs and seats and pump shafting subjected to dynamic loading during operation of the component should be included.

2. Pump and Valve Operability Assurance Programs
The component operability assurance program is intended to assure the operability of Code Class 1, 2, and 3 active valves, 2 inches and greater in nominal pipe size, and the ability of active pumps to function under plant conditions where their operation is relied upon for plant shutdown or for mitigating the consequences of an accident. The program is evaluated with respect to test and analytical methods and combinations thereof. The test program may include prototype testing, either individually under

simulated test conditions in the shop, or in situ after installation. The staff review covers the following specific information and provisions of the component operability assurance program:

a. A listing of active Class 1, 2, and 3 valves and pumps identified by system and "active" function. The Auxiliary and Power Conversion Systems Branch and the Reactor Systems Branch confirm the acceptability of the listing for Class 1, 2 and 3 pumps and valves.

b. The components, in terms of size, type, design, and manufacturer, for which one prototype test is proposed to confirm operability.

c. The components for which prototype test results are available, from applications for other plants or other sources, and the comparisons that show that the test conditions are equivalent to the plant design conditions.

d. The identification of combinations of plant conditions and loads which the active component is expected to withstand during the "active" function (such conditions are generally specified in the component design specification, as required by Code rules).

e. The test conditions and loads that will be imposed on components to confirm operability, and the comparisons to show that these are representative of plant conditions and loads (where more than one set of conditions may be applicable, the most adverse or bounding combinations should be evaluated).

f. The extent to which analytical methods will be used in lieu or in partial fulfillment of the provisions of the component operability assurance program.

3. Design and Installation of Pressure Relief Devices

The design and installation criteria applicable to the mounting of pressure relief devices (safety valves and relief valves) for the overpressure protection of Code Class 1 and Class 2 components are reviewed. The review includes evaluation of the applicable loading conditions and design stress criteria as related to the normal, upset, and emergency plant operating conditions. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens, and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system.

The design of safety and relief valve systems is reviewed with respect to the load combinations imposed on the safety or relief valves, upstream piping or header, downstream or vent piping, and system supports.

The loading combinations should identify the most severe combination of the applicable loads due to internal fluid pressure, dead weight of valves and piping, thermal load under heatup, steady state and transient valve operation, reaction forces when valves are discharging (thrust, bending, and torsion), and seismic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE).

The structural response of the piping and support system is reviewed with particular attention to the dynamic or time-history analyses employed in evaluating the appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging.

Where the use of hydraulic snubbers is proposed, the snubber performance characteristics are reviewed to assure that their effects have been considered in the analyses under steadystate valve operation and repetitive load applications caused by cyclic valve opening and closing during the course of a pressure transient.

The Auxiliary and Power Conversion Systems Branch verifies that the number and size of valves specified for the steam and feedwater systems have adequate pressure relieving capacity as confirmed by their review and evaluation of the "Overpressure Protection Analysis" that has been prepared in accordance with the requirements of the Code.

The Reactor Systems Branch verifies that the number and size of valves specified for the reactor coolant pressure boundary have adequate pressure relieving capacity as confirmed by their review and evaluation of the "Report on Overpressure Protection" that has been prepared in accordance with the requirements of the Code. The design criteria for pressure-relieving devices which may have an active function during and after a faulted plant condition are judged also against the requirements of the component operability assurance program.

4. Component Supports

The review of information submitted by the applicant includes an evaluation of Code Class 1, 2, and 3 component supports. The review includes an assessment of the design and structural integrity of the supports and their effect on the operability of active components. The review addresses three types of supports: plate and shell, linear, and component standard types, and their function.

Nuclear power plant component supports are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component.

Linear supports covered in this plan are those which are not included in Standard Review Plan 3.8.3.

II. ACCEPTANCE CRITERIA

The criteria by which the areas of review defined in Section I are judged to be acceptable are as follows:

1. Loading Combinations, Design Transients, and Stress Limits

The plant and component operating conditions, design transients, and design loading combinations considered for each system should be sufficiently defined to provide the basis for design of Code Class 1, 2, 3 and CS items for all conditions and events expected over the service lifetime of the plant and should satisfy the requirements of General Design Criteria 1, 2, and 4.

The acceptability of the combination of loading conditions and design transients applicable to the design of Code constructed items within a system, including the categorization of the appropriate plant and component operating condition for each initiating event (i.e., LOCA, SSE, pipebreak, etc.) which may be used with each loading combination, is judged by comparison with the positions stated in Reference 5, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. When these combinations have been established, the corresponding stress limits which may be applied to the design of Code constructed items are as specified in the appropriate subsections of Division 1 of Section III of the ASME Code. The need for more conservative stress limits for active components and their supports should be considered in the context and with the other features of the operability assurance program.

2. Pump and Valve Operability Assurance Program

The operation of certain pumps and valves is relied upon to shut down the plant or mitigate the consequences of an accident. These are termed "active" pumps and valves. Certain of these active pumps and valves may be required to function coincidentally with the postulated accident or event. Other active pumps and valves may be required to function only after a postulated accident or event has occurred. Acceptable procedures for demonstrating the operability of active pumps and valves during or after postulated accidents or natural events follow:

a. Pumps and Valves Whose Operability is Required During an Accident or Event

This section presents acceptable procedures for demonstrating the operability of active pumps and valves during accident or event conditions. The pump or valve includes the pressureretaining body, all internal structures, and all appurtenances necessary for component operation. The most desirable operability assurance program consists of testing the pump or valve under simulated accident or event loadings (pressure, external loads due to SSE, etc.) and environmental conditions (temperature, humidity, etc.). When this approach is not practicable, other conservative procedures may be employed. These include more elementary testing or a combination of testing and analysis. In addition, design of the pump and valve supports must be considered and accounted for in the testing and analysis to demonstrate operability. The design specification must be written to include the requirements for operability under the accident conditions; assurance of this must be provided in the SAR. Design stress limits discussed in II.1 are acceptable for active components and their supports if considered in the operability assurance program. The following programs provide an acceptable approach to demonstrate the operability of active pumps and valves requires to operate during an accident or event.

(1) Testing

The following features should be incorporated into a test program:

(a) An individual pump or valve is tested in the manufacturer's shop, or in situ following installation in the system provided the test conditions simulate those conditions under which the "active" function is required.

(b) The pump or valve is tested in the operational mode.

(c) The test program is based upon selectively testing a representative number of pumps or valves according to type, load level, size, etc. on a prototype basis. Pumps or valves that can be demonstrated to be equivalent (e.g., similar nondestructive examination program, materials, weldments, pressure, and temperature) to a prototype pump or valve, may be exempted from testing provided the test results of the prototype pump or valve are documented and available, and the loading conditions for the exempted pump or valve are equivalent to or less severe than those imposed during testing of the prototype pump or valve.

(d) The characteristics of the required seismic or accident input motion are properly specified as obtained from the system dynamic analysis and are representative of the input motion at the component mounting locations. The characteristics of the required input motion are specified by response spectrum, power spectral density function, or time history. Such characteristics, as derived from the structures or systems analysis, are representative of the input motion at the equipment mounting locations. Seismic excitation generally has a broad frequency content. Random vibration input motion should be used. However, single frequency input motions, such as sine "beats," are acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), the anticipated response of the equipment is adequately represented by one mode, or the input has sufficient intensity and duration to excite all modes to the required amplitudes such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

(e) Seismic or accident input motion is applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to vibratory motion in the horizontal direction, and vice versa. In the case of a single frequency input motion, the time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided.

(f) The characteristics of applicable environments such as temperature (at the accident condition) are taken into account.

(g) The fixture design simulates the actual service mounting (same stiffness characteristics) and causes no extraneous dynamic coupling to the test item.

(h) End loads are properly taken into account.

(i) Dynamic coupling to other related systems, if any, such as connected piping and other mechanical components, is considered.

The in situ application of vibratory devices to superimpose vibratory loadings on a complex active device is acceptable for operability assurance when it is shown that a meaningful test can be made in this way, with due regard being given to the effects on other parts of the system.

If the dynamic testing of a pump or valve assembly proves to be impracticable, static testing (static application of loads) of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than accident loads, all dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads.

(2) <u>A Combination of Test and Analysis</u>

(a) When complete testing is not practicable, a combination of test and analysis is acceptable. Simple and passive elements, such as valve and pump bodies and their related piping and supports may be analyzed to confirm structural integrity under accident loadings. However, complex active devices such as pump motors, valve operator and gate or disk assemblies, and other electrical, mechanical, pneumatic, or hydraulic appurtenances which are vital to the pump or valve operation must be tested for operability in accordance with the section above, or the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE 344-1975, as appropriate.

(b) The following analyses are acceptable provided they are correlated to classical problems, elementary laboratory tests, or in situ tests:

i. An analysis is performed to determine the seismic input to the valve or pump;

ii. An analysis is performed to determine the system natural frequencies and the movement of the pump or valve during the SSE.

iii. An analysis is performed to determine the pressure differential and the impact energy of a valve disc during a LOCA, and to verify the design adequacy of the disc.

iv. An analysis is performed to determine the forcing functions of the axial and radial loads imposed on a pump rotor due to a LOCA, such that combined LOCA and SSE effects on the shaft and rotor assembly can be evaluated.

v. An analysis is performed to determine the speed of the pump shaft as a result of postulated accidents and to compare it with the design critical speed.

vi. An analysis is performed to verify the design adequacy of the wall thickness of valve and pump pressurecontaining bodies.

vii. An analysis is performed to determine the natural frequencies of a pump shaft and rotor assembly to ascertain whether they are within the frequency range of the seismic excitations. If the minimum natural frequency of the assembly is beyond the excitation frequencies, a static deflection analysis for the shaft is acceptable to account for SSE effects. If the assembly natural frequencies are close to the excitation frequencies, an acceptable dynamic analysis must be performed to determine the structural response of the assembly to the excitation frequencies.

(3) Design Adequacy of Pump and Valve Supports

(a) Analyses or tests are performed for all supports of pumps and valves to ensure their structural capability to withstand seismic excitation.

(b) The analytical results must include the required input motions to the mounted equipment which should be obtained and characterized in the manner specified by one of the following:

i. Response spectrum.
ii. Power spectral density function.
iii. Time history.

Such characteristics, as derived from the structures or systems seismic analysis, should be representative of the input motion at the equipment mounting locations. The analytical results must also show that the combined stresses of the support structures are within the limits of the Code, Subsection NF, "Component Supports."

(c) The support is tested with the pump or valve installed or with equivalent mass inertia effects. If the equipment is inoperative during the support test, the response at the equipment mounting locations is monitored and characterized in the manner stated in (b) above. In such a case the equipment is tested separately and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response in the support test.

b. Pumps and Valves Whose Operability is Required After an Accident or Event

This section presents acceptable procedures for demonstrating the operability of active pumps and valves that are not required to operate coincident with an accident or event, but are required to operate following the accident or event. The applicant must identify those active pumps and valves considered to meet this description and justify such classification. Components that may operate or may inadvertently be operated coincident with an accident or event should meet the requirements of pumps and valves whose operability is required during an accident or event, unless the applicant can demonstrate by test or analysis that operation coincident with an accident or event will not impair the ability of the component to perform its required operation following an accident or event.

An acceptable operability assurance program for active pumps and valves whose operability is required only after an accident or event consists of design integrity and testing phases.

(1) Design Integrity

The integrity of active pumps and valves, whose operability is required only after an accident or event, is established by including in the design specification the requirement that the loads due to the accident (emergency or faulted plant conditions) shall be considered as normal loads for the active pump or valve. Design stress limits discussed in II.1 above are acceptable for active components and their supports if considered in the operability assurance program.

(2) Testing

Operability assurance testing of active pumps and valves, whose operability is required after an accident or event, is required only for the component appurtenances vital to the operation of the component, such as operators, motors, switches, relays, etc. The testing of such items may be accomplished independently of the component provided all coupling effects are identified and properly factored into the tests as boundary conditions. Such qualification testing should be in accordance with the requirements of II.2.a(1), above, or IEEE Std 344-1975, as appropriate.

c. <u>Design Specifications</u>

The design specification is the document by which the component (pump or valve) designer is guided relative to the parameters employed to describe the environment in which the component must perform its function. Consequently, it is essential that for "active" pumps and valves, the environment in which the component must perform its function to shut the plant down or mitigate the effects of an accident is adequately specified as a design requirement. Therefore, the applicant shall provide assurance that the following items are included in the design specifications of "active" pumps and valves:

(1) External loads expressed as flange end loadings associated with the accident condition for which the pump or valve must operate; i.e., the loading combinations associated with the faulted plant condition, and with due regard for the proper representation of the supports, if any, become the design loads for the active component. The design loads must be equal to or less than the end loads specified by the component manufacturer as permitted for normal operation.

(2) All other relevant environmental conditions, such as temperature, humidity, etc., associated with the accident condition are specified as a normal design condition.

(3) Operating clearances or deformation limits necessary to assure operation are specified and maintained for the accident condition in which the component must operate. Excessive rubbing (other than ordinary seal rub) on rotating parts is not acceptable for active pumps under the accident conditions.

(4) All test conditions, including loadings and environmental conditions, are specified and operability requirements stated.

(5) The operability requirements during or after the accident or event are clearly stated.

3. Design and Installation of Pressure Relief Devices

Acceptable design criteria for pressure relief stations in open discharge systems are specified in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

As indicated in Code Case 1569, the rules for acceptable design procedures for systems where the pressure-relieving devices discharge into closed systems or systems with long discharge pipes have not reached the stage of final codification. However, for these closed or quasi-closed systems, the safety analysis report must include a commitment to perform a conservative dynamic analysis of the system, including mounting pipe runs or headers where applicable, relief device mountings, and discharge piping systems. The SAR must also include a description of the calculational procedures, computer programs, and other methods to be used in the analysis. The analysis must include the time history or equivalent effects of changes of momentum due to fluid flow changes of direction. The fluid states considered must include postulated water slugs where water seals are used. Stress computations and stress limits must be in accord with applicable rules of the Code.

4. Component Supports

To be acceptable, the component support designs should provide adequate margins of safety under all plant operating conditions.

The acceptability of the combinations of loading conditions and design transients applicable to the design of component supports within a system, including the categorization of the appropriate plant and component support operating condition for each initiating event, (i.e., LOCA, SSE, pipe break, etc.) which may be used with each loading combination, is judged by comparison with the positions stated in Reference 5, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. When these conditions have been established, the corresponding stress limits which may be applied to the design of component supports are as specified in Subsection NF of Division 1 of Section III of the ASME Code. The need for more conservative stress limits for active component supports should be considered in context with the other features of the operability assurance program.

In addition, if the component support affects the operability requirements of the supported component, then deformation limits should also be specified. The deformation limits for active component supports should be compatible with the operability requirements of the components supported. In establishing allowable deformations, the possible movements of the support base structures must be taken into account.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedures apply:

1. Loading Combinations, Design Transients, and Stress Limits

The objectives in reviewing the loading combinations and stress limits employed by the applicant in the design of Code Class 1, 2, 3, and CS items are to confirm that each of the plant operating conditions have been included, that the loading combinations and design transients applicable to the design of Code constructed items and the categorization of proposed operating conditions are appropriate, that the design stress levels associated with each imposed loading combination are low enough to provide adequate margins with respect to the structural integrity of the item, and that for active components and their supports, stress levels are considered in the operability assurance program. The review conducted during the CP stage determines that the objectives have been addressed and are being implemented in the design in the form of a commitment by the applicant that specific design criteria will be utilized by checking actual summary analysis results, the OL stage review verifies that the design criteria have been utilized and that components have been designed to meet the objectives. To assure that these objectives are met, the review is performed as follows:

a. The applicant's proposed combination of plant operating conditions and appropriate compensating conditions in terms of anticipated transients and design basis events is reviewed for completeness and for categorization as normal, upset, emergency, or faulted.

b. The combination of design loading conditions, including procedures for combination, proposed by the applicant for each Code constructed item are reviewed to determine if they are adequate. This aspect of the review is made by comparison with the loading combinations set forth in Regulatory Guide 1.48. Deviations from the guide are evaluated on a case-by-case basis by questions addressed to the applicant to determine the rationale and justification for exceptions. Final determination is based on engineering judgment and past experience with prior applications.

c. The design stress limits selected by the applicant for each plant and item operating condition as established in (b) are reviewed to determine if they meet those specified in the appropriate subsection of Division 1 of the Code, and in Regulatory Guide 1.48. Deviations from Regulatory Guide 1.48 may be permitted provided justification is presented by the applicant. The acceptability determination is based on considerations of adequate margins of safety.

d. Analytical methods for components including their internal parts subjected to the faulted component operating condition dynamic loading should meet the criteria set forth in Section 5 of Standard Review Plan 3.9.2 as prescribed for reactor internals.

2. Pump and Valve Operability Assurance Program

The objective of the review of the pump and valve operability assurance program is to determine whether the program submitted will assure the operability of a component which is required to function to shut down the plant or mitigate the consequences of an accident. During the CP stage, a commitment to adopt a program which satisfactorily meets the acceptance criteria is required. At the OL stage, it is verified that the detailed procedures actually meet this objective. To assure the achievement of the objective, the review is performed as follows:

a. The applicant's program is reviewed to determine if it consists of the proper combination of test and analysis.

b. The test and analysis methods and programs are reviewed by comparing the information submitted in the SAR with the acceptance criteria delineated in Section II.2 of this review plan. In those cases that are not directly comparable, the reviewer determines whether an acceptable level of assurance of operability has been reached.

3. Design and Installation of Pressure Relief Devices

The objective of the review of the design and installation of pressure relief devices is to assure the adequacy of the design and installation, so that there is assurance of the integrity of the pressure relieving devices and associated piping during the functioning of one or more of the relief devices. In the CP review, it is determined whether there is reasonable assurance that the final design will meet these objectives. At the OL stage, the final design is reviewed to determine that the objectives have been met.

The review is performed as follows:

a. The design of the pressure-retaining boundary of the device is reviewed by comparison with the Code. Since explicit rules are not yet available within the Code for the design of safety and pressure relief valves, the design is reviewed on the basis of reference to sections of the Code on vessels, piping, and line valves, and on experience with similar installations and good engineering design practice.

   Allowable stress limits are compared with those for the appropriate class of construction in the Code. Deviations are identified and the applicant is requested to provide justification. Stress limits and loading combinations for the various plant operating conditions are covered under the subsections entitled "Loading Combinations, Design Transients, and Stress Limits" in this plan.

b. The design of the installation is reviewed for structural adequacy to withstand the dynamic effects of relief valve operation. The applicant should include and discuss: reaction force, valve opening sequence, valve opening time, method of analysis, and magnitude of a dynamic load factor (if used). In reaching an acceptance determination, the reviewer compares the submission with the requirements in II.3, above.

   Where deviations occur, they are identified and the justification is evaluated. Valve opening sequence effects must consider the worst combination possible and forcing functions must be justified with valve opening time data. The review is based in part on comparisons with prior acceptable designs tested in operating plants.

4. Component Supports

The objective in the review of component supports is to determine that adequate attention has been given the various aspects of design and analysis, so that there is assurance as to support structural integrity and as to operability of active components that interact with component supports.

The structural integrity and the effects on operability of the three types of component supports described in I.4 are reviewed against the criteria and guidelines of II.1 and II.4 of this plan.

Also, the ASME Code provides rules for the construction requirements for metal supports which are intended to transmit loads from the pressure-retaining barrier of the component, as defined in Subsection NF of the Code, to the load-carrying structural member, whether concrete or structural steel.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The specified design basis combinations of loadings as applied to safety-related ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site, or an upset, emergency, or faulted plant transient occurring during normal plant operation, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components constitute an acceptable basis for design in satisfying applicable portions of General Design Criteria 1, 2, and 4.

"The component operability assurance program for ASME Code Class 1, 2, and 3 active valves and pumps provides adequate assurance of the capability of such active components (a) to withstand the imposed loads associated with normal, upset, emergency, and faulted plant and component operating conditions without loss of structural integrity, and (b) to perform necessary "active" functions (e.g., valve closure or opening, pump operation) under accident conditions and conditions expected when plant shutdown is required. The specified component operability assurance test program constitutes an acceptable basis for satisfying applicable portions of General Design Criteria 1, 2, and 4 and is acceptable to the staff.

"The criteria used in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67.

"The specified design basis loading combinations used for the design of safety-related ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or an upset, emergency, or faulted plant transient, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports constitute an acceptable basis for satisfying applicable portions of General Design Criteria 1, 2, and 4."

Class CS component evaluation findings are covered in Standard Review Plan 3.9.5 in connection with reactor internals.

V. REFERENCES

1. 10 CFR § 50.55a, "Codes and Standards."

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

4. IEEE Std 344-1975, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

5. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

6. Regulatory Guide 1.67, "Installation of Overpressure Protection Devices."

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.4 CONTROL ROD DRIVE SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

Information in the areas noted below is provided in the applicant's safety analysis report and is reviewed by the MEB in accordance with this plan. This information pertains to the reactor control rod drive system (CRDS), which is considered to extend to the coupling interface with the reactivity control elements in the reactor pressure vessel. For electromagnetic systems, the review under this plan is limited to just the control rod drive mechanism (CRDM) portion of the CRDS. For hydraulic systems, the review covers the CRDM and also the hydraulic control unit, the condensate supply system, and the scram discharge volume. For both types of systems, the CRDM housing should be treated as part of the reactor coolant pressure boundary (RCPB); the relevant mechanical engineering information may be presented in this section or by reference to the sections on the RCPB.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated in CRDS of current types, information should be supplied for the new systems or new features similar to that described below.

1. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly.

2. A review is performed of information pertaining to design codes, standards, specifications, and standard practices, as well as to General Design Criteria, Regulatory Guides, and branch positions that are applied in the design, fabrication, construction, and operation of the CRDS.

The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which they apply. Pressurized parts of the system are reviewed to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code") for those portions which are not part of the reactor coolant pressure boundary, and with other specified parts of Section III, or other sections of the Code for pressurized portions which are not part of the reactor coolant pressure boundary. The MEB reviews the non-pressurized portions of the control rod drive system to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately covers the areas of concern in stress, deformation, and fatigue.

3. Information is reviewed which pertains to the applicable design loads and their appropriate combinations, to the corresponding design stress limits, and to the corresponding allowable deformations. The deformations are of interest in the present context only in those instances where a failure of movement could be postulated due to excessive deformation and such movement would be necessary for a safety-related function.

   If the applicant selects an experimental testing option in lieu of establishing a set of stress and deformation allowables, a detailed description of the testing program must be provided for review.

   In the preliminary safety analysis report (PSAR), the load combinations, design stress limits, and allowable deformations criteria should be provided for review.

   In the final safety analysis report (FSAR), the actual design should be compared with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

   Loadings imposed during normal plant operation and startup and shutdown transients include but are not limited to pressure, deadweight, temperature effects, and anticipated operational occurrences. Loadings associated with specific seismic and other dynamic events are then combined with the above plant-type loads. Each set of combined loads has a selected stress or deformation limit. The selection of a specific limit is influenced by the probability of the postulated event occurring and the need to assure operation during and after the event.

4. The portion of the SAR is reviewed that describes plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus. For example, the life cycle test program for the CRDS is reviewed. The operability assurance program is reviewed to ascertain coverage of the following:
   a. Life cycle test program.
   b. Proper service environment imposed during test.

c. Mechanism functional tests.

d. Program results.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review are the following:

1. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet Section 3.9.4 of Reference 14.

2. Construction (as defined in NA-1110 of Section III of the ASME Code, Reference 10) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

   a. Pressurized Portions of Equipment Classified as Quality Group A, B, C (Regulatory Guide 1.26)

      Section III of the ASME Code, Class 1, 2, or 3 as appropriate (Ref. 10).

   b. Pressurized Portions of Equipment Classified as Quality Group D (Regulatory Guide 1.26)

      (1) Section VIII, Division 1 of the ASME Code for vessels and pump casings (Ref. 10).

      (2) Applicable to Piping Systems (American National Standards Institute, ANSI)[1/]:

      B16.5 Steel Pipe Flanges and Flanged Fittings (Ref. 16).
      B16.9 Steel Butt Welding Fittings (Ref. 17).
      B16.11 Steel Socket Welding Fittings (Ref. 18).
      B16.25 Butt Welding Ends (Ref. 19).
      B31.1 Piping (Ref. 20).
      SP-25 Standards (Ref. 21).
      SP-66 Valves (Ref. 22).

   c. Non-Pressurized Equipment (Non-ASME Code)

      Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than those for other plants of similar design having a period of successful operation. Justification of any decreases should be provided.

3. For the various plant operating conditions defined in NB-3113 of Section III of the ASME Code (Ref. 10), load combination sets are as given in Standard Review Plan 3.9.3 (Ref. 15). The stress limits applicable to pressurized and non-pressurized portions of the control rod drive systems should be as given in Reference 15 for each loading set.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and overcoming a stuck rod meet system design requirements.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

1/ This list can be extended by a staff review and acceptance of other ANSI & MSS Standards in the piping system area.

1. The objectives of the review are to determine that design, fabrication, and construction of the control rod drive mechanisms provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

   In the construction permit (CP) review, it should be determined that the design criteria utilize proper load combinations, stress and deformation limits, and that operability assurance is provided by reference to a previously accepted testing program or that a commitment is made to perform a testing program which includes the essential elements listed below. In the operating license (OL) review, the results of any testing program not previously reviewed should be evaluated.

2. The design criteria presented should be evaluated for both the internal pressure-containing portions and other portions of the CRDS. These include the CRDM housing, hydraulic control unit, condensate supply system and scram discharge volume, and portions such as the cylinder, tube, piston, and collect assembly.

   Of particular concern are any new and unique features which have not been used in the past. Pressure-containing components are checked to ensure that they meet the design requirements of the codes and criteria which have been accepted by the Reactor Systems Branch, and are identified in Standard Review Plan 3.2.2. The review of the functional design of reactivity control systems, including control rod drive systems, is the responsibility of RSB (See SRP 4.5). The loading combinations for the various plant operating conditions are checked for consistency with Reference 15; given these loading combinations, the stress limits of the appropriate code should not be exceeded, or the limits in Reference 15 should not be exceeded if not specified in the listed design code. Exceptions taken by the applicant to any of the accepted codes, standards, or AEC criteria must be identified and the basis clearly justified so that evaluation is possible. Engineering judgment, experience, comparisons with earlier cases and design margins, and consultation with supervisors permit the reviewer to reach a decision on the acceptability of any exceptions posed by the applicant.

   The choice of materials of construction for unpressurized equipment that is not governed by accepted codes or standards is reviewed by the MTEB.

3. Loading combinations are defined as those loadings associated with plant operations which are expected to occur one or more times during the lifetime of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events. The load combinations which are postulated to occur are specified for each of the plant operating conditions as defined in Paragraph NB-3113 of the ASME Code (Ref. 10). These load combinations are defined in Reference 15 and are compared by the reviewer with those provided by the applicant.

The design stress limits, including fatigue limits, and deformation limits as appropriate to the components of the control rod drive mechanism are compared by the reviewer with those of specified codes, previously designed and successfully operating systems, or with the results of scale model and prototype testing programs.

4. The control rod drive mechanisms of a new design or configuration should be subjected to a life cycle test program to determine the ability of the drives to function over the full range of temperatures, pressures, loadings, and misalignment expected in service. The tests should include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear. Rod travel and number of trips expected during the mechanism operational life should be duplicated in the tests.

   The reviewer checks the elements of the test program to be sure all required parameters have been included and finally reviews the test results to determine acceptability. Excessive wear, malfunction of components, operating times beyond determined limits, scram accumulator leakage, etc., all would be cause for retesting.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and that his evaluation is sufficiently complete and adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the reactivity control system are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

3. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

4. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

5. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

7. 10 CFR Part 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

8. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

9. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

10. ASME Boiler and Pressure Vessel Code, Sections III and VIII, American Society of Mechanical Engineers.

11. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

12. Regulatory Guide 1.29, "Seismic Design Classification."

13. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components."

14. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

15. Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

16. ANSI B 16.5, "Steel Pipe Flanges and Flanged Fittings," American National Standard Institute.

17. ANSI B 16.9, "Wrought Steel Butt Welding Fittings," American National Standard Institute.

18. ANSI B 16.11, "Steel Fittings Steel Welding and Threaded," American National Standard Institute.

19. ANSI B 16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings," American National Standard Institute.

20. ANSI B 31.1, "Power Piping," American National Standard Institute.

21. MSS-SP-25, "Marking for Valves, Fittings, Flanges, and Unions," Manufacturers Standardization Society.

22. MSS-SP-66, "Pressure-Temperature Ratings for Steel Butt Welding End Valves," Manufacturers Standardization Society.

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SECTION 3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Core Performance Branch (CPB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

For the purpose of this standard review plan, the term "reactor internals" refers to all structural and mechanical elements inside the reactor pressure vessel with the exception of the following:

Reactor core (fuel), including the reactivity control elements out to the coupling interfaces with the drive units, as well as the drive elements inside the guide tubes (guide tubes are considered to be a part of reactor internals) and inside the control rod drive mechanism assemblies (drive elements are covered in Standard Review Plan 3.9.4).

In-core instrumentation (in-core instrumentation support structures are considered part of the reactor internals).

The staff review includes the following specific areas:

1. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems should be presented, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

2. The design loading conditions that provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events should be specified. All combinations of design loadings should be listed (e.g., operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the core support structure.

3. Each combination of design loadings should be categorized with respect to the "normal," "upset," "emergency," or "faulted" condition (defined in the ASME Code, Reference 5) and the associated design stress intensity or deformation limits should be stipulated. Design loadings should include safe shutdown earthquake (SSE) and operating basis earthquake (OBE) loads if applicable.

4. The design bases for the mechanical design of the reactor vessel internals should be presented including limits such as maximum allowable stresses; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and holddown). Details of dynamic analyses, input forcing functions, and response loadings are discussed in Standard Review Plan (SRP) 3.9.2.

II. ACCEPTANCE CRITERIA

A discussion of loading combinations applicable to reactor internals is presented in SRP 3.9.3 (Ref. 7).

The design and construction of the core support structures should conform to the requirements of Subsection NG, "Core Support Structures," of the ASME Code (Ref. 5).

The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures should be consistent with the same requirements as listed above for core support structures.

Deformation limits for reactor internals should be established by the applicant and presented in his safety analysis report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified design limits. The requirements for dynamic analysis of these components are discussed in SRP 3.9.2.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

The configuration and general arrangement of all mechanical and structural internal elements covered by this plan are reviewed and compared to those of previously licensed similar plants. Any significant changes in design are noted and the applicant is asked to verify that these changes do not affect the flow-induced vibration test results required by SRP 3.9.2.

With respect to the design and analysis of these components, a statement by the applicant that they are designed in accordance with Subsection NG, "Core Support Structures," of Reference 5 is acceptable. In lieu of such a commitment, the reviewer must determine that the design and analysis of these components are consistent with the requirements discussed in II, above. This is accomplished by requiring that the applicant describe the design procedures and criteria used in the design of these components. This includes a list of the design limits used for all of the applicable loading conditions.

The deformation limits specified for these components are reviewed to verify that the applicant has stated that these deflections will not interfere with the functioning of related components, e.g., control rods and standby cooling systems, and that the stresses associated with these displacements are less than the design limits for the core support structures.

At the operating license stage, the calculated stresses and deformations are reviewed to determine that they do not exceed the specified design limits.

Any deviations that have not been adequately justified are identified and findings to that effect are transmitted to the applicant with a request for conformance with the requirements discussed in II above or additional technical justification.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The design procedures and criteria that the applicant has used for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria which are acceptable to the staff.

"The specified design transients, design loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system upset or faulted condition transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function. In addition, the design procedures and criteria used by the applicant in the design of the reactor internals constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4, and 10."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, Criterion 2, "Design Basis for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, Criterion 10, "Reactor Design."

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

6. Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."

7. Standard Review Plan 3.9.3, "Pressure Retaining Components and Component Supports."

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SECTION 3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The MEB reviews the following areas of the applicant's safety analysis report (SAR) that cover the inservice testing of pumps and valves designated as Class 1, 2, or 3 under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code"), Section III:

1. Inservice Testing of Pumps
   a. The descriptive information in the SAR covering the inservice test program for all Code Class 1, 2, and 3 system pumps is reviewed. The Reactor Systems Branch verifies the code class designations for each listed pump and the completeness of the list.
   b. Reference values for testing for speed, pressure, flow rate, vibration, and bearing temperature at normal pump operating conditions are reviewed.
   c. The pump test schedule, included in the plant technical specifications, is reviewed.
   d. The methods described in the SAR for measuring the reference values and inservice values for the pump parameters listed in I.1.b above are reviewed.

2. Inservice Testing of Valves
   a. The descriptive information in the SAR covering the inservice test program of all Code Class 1, 2, and 3 valves is reviewed. This review does not include those valves defined in IWV-1300 of Section XI of the Code. The Reactor Systems Branch verifies the code class designations for each listed valve and the completeness of the list.
   b. The SAR test program, which includes preservice tests, valve replacement, valve repair and maintenance, indication of valve position, and inservice tests for all valve categories, is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review, described in I of this plan are as follows:

1. Inservice Testing of Pumps
   a. The scope of the applicant's test program is acceptable if it is in agreement with IWP-1000 of Section XI of the Code. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in IWP-3000, the establishment of a reference set of parameters and a consistent test method is a basic criterion of the program.
   b. The pump test program is acceptable if it meets the requirements for establishing reference values and the periodic testing schedule of IWP-3000 of Section XI of the Code. The allowable ranges of inservice test quantities, corrective actions, and bearing temperature tests are established by IWP-3200 and IWP-4300. The pump test schedule in the plant technical specifications is required to comply with these rules.
   c. The test frequencies and durations in the plant technical specifications are acceptable if the provisions of IWP-3300 and IWP-3400 of Section XI of the Code are met. If a pump is normally operated more frequently than once a month, and at the reference conditions, it need not be specially tested. Otherwise, pumps must be tested each month during plant operation, and during shutdown periods if practical. The pumps must be run for at least five minutes under conditions as stable as the system permits. Bearing temperatures must be measured once a year for the duration specified in IWP-3410.
   d. The methods of measurement are acceptable if the test program meets the requirements of IWP-4100, 4200, 4300, 4400, and 4500 of Section XI of the Code with regard to instruments, pressure measurements, temperature measurements, rotational speed, and vibration measurements.

2. Inservice Testing of Valves
   a. To be acceptable, the SAR valve test list must contain all Code Class 1, 2, and 3 valves except those used for operating convenience only, such as manual vent, drain, and test valves, and valves used for maintenance only. The SAR valve list must include a valve categorization which complies with the provisions of IWV-2110 of Section XI of the Code. Each specific valve to be tested by the rules of Subsection IWV is listed in the SAR by type, valve identification number, code class, and IWV-2110 valve category.
   b. The valve test procedures in the plant technical specifications are acceptable if the provisions of IWV-3000 of Section XI of the Code are met with regard to pre-service and periodic inservice valve testing.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case. For each area of review, the following review procedures are followed:

1. Inservice Testing of Pumps

   a. The scope of the applicant's program is reviewed for agreement with II.1.a of this plan. The program is acceptable if a preservice test program is used to establish reference values. The periodic inservice program must verify the reference values within acceptable limits.

   b. The pump test program procedures must agree with the requirements of II.1.b of this plan. The applicant must justify any exception to II.1.b. The program is best presented in tabular form in the plant technical specifications.

   c. The inservice test frequencies and test durations in the plant technical specifications are reviewed for agreement with II.1.c of this plan.

   d. The test results described in the SAR are reviewed for agreement with II.1.d of this plan. The SAR need only provide the necessary information to permit a conclusion that the methods of measurement and the data acquisition system will provide the needed data. The reviewer does not approve or disapprove the instruments or methods proposed or used.

2. Inservice Testing of Valves

   a. The SAR valve test list and valve category description are reviewed for agreement with II.2.a of this plan.

   b. The valve test program is acceptable if the procedures follow the rules of Section II.2.b of this plan for preservice and periodic inservice testing.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided in accordance with the requirements of this review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"To ensure that all ASME Code Class 1, 2, and 3 pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant, a test program is provided which includes baseline preservice testing and periodic inservice testing. The program provides for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress.

"The applicant has stated that the inservice test program for all Code Class 1,.2, and 3 pumps and valves meets the requirements of the ASME Code, Section XI, Subsections IWP and IWV, respectively.

"Compliance with these code requirements constitutes an acceptable basis for satisfying the applicable portions of General Design Criteria 37, 40, 43, and 46."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

2. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

4. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

5. ASME Boiler and Pressure Vessel Code, Section III and Section XI, Subsections IWP and IWV, American Society of Mechanical Engineers.

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SECTION 3.10 SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

Information concerning the methods of test and analysis employed to assure the operability of essential instrumentation and electrical equipment in the event of an earthquake should be provided in the applicant's safety analysis report (SAR) and is reviewed by the MEB in accordance with this plan. Systems and components that must retain structural integrity, remain leaktight, or continue to function in the event of an earthquake, in order to assure safe operation or shutdown of the plant, are designated seismic Category I systems and components.

At the construction permit (CP) stage, the staff review covers the following specific areas:

1. The criteria for seismic qualification, such as the deciding factors for choosing between tests or analyses, the considerations in defining the seismic input motion, and the demonstration of adequacy of the seismic qualification program.

2. The methods and procedures, including tests and analyses, used to assure the operability of seismic Category I instrumentation and electrical equipment in the event of a safe shutdown earthquake (SSE) or less severe earthquakes such as the operating basis earthquake (OBE). Instrumentation and electrical equipment designated as seismic Category I include the reactor protection system, engineered safety feature circuits, emergency power systems, and all auxiliary safety-related electrical systems.

3. The analysis or testing of supports for seismic Category I instrumentation and electrical equipment, and the procedures used to account for possible amplification of vibratory motion (amplitude and frequency content) under seismic conditions. Supports include items such as battery racks, instrument racks, control consoles, cabinets, panels, and cable trays.

At the operating license (OL) stage, the staff reviews the results of tests and analyses to assure the proper implementation of criteria accepted in the CP review, and to demonstrate adequate seismic qualification.

The EICSB verifies that all of the seismic Category I instrumentation and electrical equipment and supports are included in the seismic qualification program, that the electrical performance aspects of the seismic qualification testing meet safety requirements, and that the equipment mounting during the test adequately simulates the actual service mounting. The EICSB also verifies, at the OL stage, that the equipment and instrumentation used in the plant have been appropriately qualified.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review of this plan are as follows:

1. For plants for which the CP application, including the preliminary safety analysis report (PSAR), was docketed before October 27, 1972, the seismic qualification of Category I instrumentation, electrical equipment, and supports should meet the requirements of IEEE Std 344-1971 (Ref. 3). In addition, the following requirements should be met:

   a. Seismic Qualification for Equipment Operability

      (1) Tests or analyses are required to confirm the functional operability of all seismic Category I electrical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE. (The analysis method is not recommended for complex equipment that cannot be modeled accurately enough to predict its response correctly.)

      Designs and equipment that have been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses is submitted.

      (2) Input excitations such as continuous single frequency sinusoidal motions or sine beat motions should be used. The maximum input motion acceleration should equal or exceed the maximum seismic acceleration expected at the equipment mounting location. See II.1.b(3) below for a discussion of the participation of the equipment supports.

      (3) The discrete frequencies at which the test input motion is applied should cover the range 1-33 Hz. If resonant frequencies of the equipment and equipment supports are identified by prior analysis or "sweep" testing or both, tests conducted only at the resonant frequencies are acceptable.

      (4) Equipment should be tested in the operational condition. Procedures for monitoring the equipment under test are reviewed by EICSB.

      (5) The test motion should be applied to one vertical and two orthogonal horizontal axes separately.

      (6) The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc. on a prototype basis.

b. Seismic Design Adequacy of Supports

(1) Analyses or tests should be performed for all supports of seismic Category I electrical equipment and instrumentation to assure their structural capability to withstand seismic excitation.

(2) The analytical results should include the maximum accelerations and associated frequencies at the equipment mounting location, and the combined stresses of the support structures should be within the limits of the ASME Code, Section III, Subsection NF, "Component Support Structures" (Ref. 2).

(3) Supports should be tested with equipment installed. If the equipment is installed in a nonoperational mode for the support test, the response at the equipment mounting location should be monitored such that the maximum accelerations and associated frequencies can be defined. In such a case, equipment should be tested separately for operability and the actual input motion to the equipment should be more conservative in amplitude and frequency content than the monitored response.

(4) The requirements of II.1.a(2), (3), and (5), above, are applicable when tests are conducted on the equipment supports.

2. For plants for which the CP application was docketed after October 27, 1972, the seismic qualification of Category I and instrumentation, electrical equipment, and supports should conform to the following (also see Ref. 4):

a. Seismic Qualification for Equipment Operability

(1) Tests and analyses are required to confirm the functional operability of all seismic Category I electrical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE. Analyses alone, without testing, are acceptable as a basis for seismic qualification only if the necessary functional operability of the instrumentation or equipment is assured by its structural integrity alone. When complete seismic testing is impractical, a combination of tests and analyses is acceptable.

Designs and equipment that have been previously qualified by means of tests and analyses equivalent to those described here are acceptable provided that proper documentation of such tests and analyses is submitted.

(2) The characteristics of the required (seismic) input motion should be specified by response spectrum, power spectral density function, or time history methods. These characteristics, derived from the structures or systems seismic analysis, should be representative of the seismic input motion at the equipment mounting locations.

(3) Equipment should be tested in the operational condition. Operability should be verified during and after the testing.

(4) The actual (test) input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated.

(5) Seismic excitation generally has a broad frequency content. Random vibration input motion should be used. However, single frequency input motions, such as sine beats, are acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), the anticipated response of the equipment is adequately represented by one mode, or the input has sufficient intensity and duration to excite all modes to the required amplitudes such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

(6) The test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

(7) The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item.

(8) The in situ application of vibratory devices to superimpose the seismic vibrator motions on a complex active device for operability testing is acceptable when it is shown that a meaningful test can be made in this way.

(9) The test program may be based upon selectively testing a representative number of components according to type, load level, size, etc., on a prototype basis.

b. Seismic Design Adequacy of Supports

(1) Analyses or tests should be performed for all supports of seismic Category I electrical equipment and instrumentation to assure their structural capability to withstand seismic excitation.

(2) The analytical results should include the required input motions to the mounted equipment as obtained and characterized in the manner stated in 11.2.a(2), above and the combined stresses of the support structures should be within the limits of the ASME Code, Section III, Subsection NF, "Component Support Structures" (Ref. 2).

(3) Supports should be tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location should be monitored and characterized in the manner as stated in II.2.a(2), above. In such a case, equipment should be tested separately for operability and the actual input motion to the equipment in this test should be more conservative in amplitude and frequency content than the monitored response from the support test.

(4) The requirements of 11.2.a(2), (4), (5), (6), and (7), above, are applicable when tests are conducted on the equipment supports.

3. In documenting the implementation of the seismic qualification program described above, the SAR should:
    a. Describe briefly the testing facilities, including the capability of the facilities to test the functioning of the equipment being tested and to provide the test input.
    b. Provide a list of equipment (devices or assemblies) and support structures tested.
    c. Identify the type of testing input motion, including intensity level, frequency content, number of axes, input duration, and time history sketches of the typical input. The validity of such testing input motion should be demonstrated.
    d. Describe the number, type, and location of monitoring sensors used.
    e. Identify whether devices are tested in the operating condition.
    f. Identify whether devices are mounted during the testing of assemblies or supporting structures (i.e., panels, racks, etc.) and demonstrate the validity of any tests conducted without the devices (or suitable substitutes) or with the mounted devices in inoperative condition.
    g. In the event testing is replaced by analysis, provide justification that the analysis assures the proper functioning of the equipment during the SSE.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below as may be appropriate for a particular case.

For each area of review the following review procedures are used:

1. At the CP stage, the staff reviews the program which the applicant has described in the PSAR for the seismic qualification of all Category I instrumentation and electrical equipment. The program is measured against the requirements listed in Section II of this plan. Of particular interest are the proper use of test and analytical procedures. Equipment which is too complex for reliable mathematical modeling should be tested unless the analytical procedures and corresponding design are convincingly conservative. Both the test and the analysis methods are reviewed for assurance that all important modes of response have been excited in tests or considered in anlayses. Proper application of input motions so as to bound the required input, whether in terms of response spectra, power spectral density, or time history in all necessary directions is verified. The use or treatment of supports is also reviewed.

2. At the OL stage, the staff reviews the program again as described by the applicant in the FSAR. In addition, the FSAR is reviewed for documentation of the successful implementation of the seismic qualification program including test and analysis results. Also, the acceleration levels used in the tests and in the analyses are reviewed for assurance that they equal or exceed the levels at the equipment mounting locations derived from structural response studies of the plant structure as built or as designed.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information has been provided and that the review supports conclusions of the following type (for a CP review), to be included in the staff's safety evaulation report:

"The proper functioning of essential instrumentation and electrical equipment in the event of the safe shutdown earthquake (SSE) is necessary to initiate protective actions including, for example, operation of the reactor protection system, engineered safety features, and standby power systems.

"The seismic qualification testing program which will be implemented for seismic Category I instrumentation and electrical equipment provides adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. This program constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 2."

At the OL stage, the review should provide justification for a finding similar to that above with the phrase "will be implemented" modified to read "has been implemented."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. IEEE Std 344-1971, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

4. IEEE Std 344-1975, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

5. K. M. Skreiner, E. G. Fischer, S. N. Hou, and G. Shipway, "New Seismic Requirements for Class I Electrical Equipment," IEEE Paper T 74 048-5, 1974 Winter Meeting of IEEE Power Engineering Society, Institute of Electrical and Electronics Engineers.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The information presented in Section 3.11 of the applicant's safety analysis report (SAR) should be sufficient to support the conclusion that all items of safety-related mechanical and electrical equipment are capable of performing their design safety functions under all normal and accident environmental conditions. The "normal and accident environmental conditions" are deemed to include all environmental conditions which may result from any normal or abnormal mode of plant operation, design basis events, post-design basis events, and containment tests. The information presented should include identification of the safety-related equipment, and for each item of equipment, the environmental design bases, definition of normal and postulated environments, and documentation of the qualification tests and analyses performed to demonstrate the required environmental capability. In the preliminary safety analysis report (PSAR), this documentation may consist of a description of the tests and analyses that have been or will be performed. In the final safety analysis report (FSAR), the results of the qualification tests and analyses for each type of equipment should be provided. Seismic qualification is addressed in Standard Review Plan 3.10.

Section 3.11 of the SAR is reviewed to determine whether the required environmental capability of all safety-related equipment, i.e., the capability to perform design safety functions under normal and accident environments, will be or has been adequately demonstrated.

The EICSB makes a completeness check of the information provided by the secondary review branches as detailed below.

When requested, the secondary review branches (APCSB, CSB, RSB, QAB) will provide information to the EICSB with regard to mechanical and electrical equipment of safety-related systems within their respective primary review responsibilities, but exclusive of any electrical equipment located in the control room or other designated electrical equipment rooms or areas (this equipment is an EICSB responsibility). The SAR sections reviewed by the branches in performance of their secondary review functions are as follows: APCSB reviews Section 3.4.1 and applicable sections of Chapters 9 and 10; CSB reviews Section 6.2; RSB reviews Sections 3.2.1, 3.2.2, 4.4, 6.3, and applicable sections of Chapter 15; and QAB reviews Chapter 17. Guidance with regard to the definition of "safety-related systems" for the purposes of this plan is contained in Standard Review Plan 7.1, and the assignments of primary review responsibility for these systems are contained in the applicable review plans.

The APCSB, CSB, and RSB confirm that the SAR identifies all safety-related equipment.

The APCSB and CSB confirm the location of each item if equipment, both inside and outside the containment. Inside the containment, the location must specify whether inside or outside of the missile shield, for pressurized water reactor (PWR) plants, or whether inside or outside of the drywell, for boiling water reactor (BWR) plants with Mark III containment designs.

The APCSB, CSB, and RSB confirm the validity of the descriptions of both the normal and accident environments provided in the SAR. They will also confirm the acceptability of the values provided in the SAR for the length of time that equipment is required to operate in accident environments.

With regard to the environments resulting from loss of environmental control systems (ventilation, heating, air conditioning), the APCSB will confirm the description of these environments as provided in the SAR for those areas which contain safety-related equipment, including electrical control and instrumentation equipment.

The QAB reviews the environmental design and qualification program described in Section 3.11 of the SAR to ascertain that it is being implemented in accordance with the requirements of the quality assurance program described in Chapter 17 of the SAR.

Specific information may be requested from the MEB as needed.

II. ACCEPTANCE CRITERIA

The general requirements for environmental design and qualification of all equipment important to safety are embodied in General Design Criteria 1, 4, and 23 of Appendix A to 10 CFR Part 50, and in Section XI of Appendix B to 10 CFR Part 50. In addition, the requirement for environmental qualification is included in IEEE Std 279 (Ref. 3) and in IEEE Std 308 (Ref. 4). However, none of the above documents provide specific criteria for assessing the acceptability of an environmental design and qualification program.

Simply stated, the general requirements for environmental design and qualification are as follows. (1) The equipment shall be designed to have the capability of performing design safety functions under all normal and accident environments. (2) The equipment environmental capability shall be demonstrated by appropriate testing and analyses. (3) A quality assurance program shall be established and implemented to provide assurance that these requirements are met. The environmental design of safety-related mechanical and electrical equipment is acceptable when it can be ascertained that all three requirements are met.

Section V of this review plan lists the documents which provide both acceptance criteria and evaluation guidance used in the review. The most important of these documents is IEEE Std 323-1974, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations" (Ref. 6). This document, although specifically written for Class I electric equipment, contains a clear presentation of the principles and criteria that are generic to the environmental qualification process itself; therefore, IEEE Std 323-1974 is considered applicable to the environmental qualification of other types of equipment. This document contains detailed criteria applicable to whatever method of qualification is used, i.e., type testing, analyses, operating experience, on-going qualification, or combined qualification. The environmental design and qualification of safety-related equipment is acceptable when it is ascertained that the criteria of IEEE Std 323-1974 have been met.

IEEE Std 334-1971, "Guide for Type Tests of Continuous-Duty Class I motors Installed Inside the Containment of Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.40); IEEE Std 382-1972, "Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.73); and IEEE Std 383-1974, "Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," are specific with regard to type test qualification of the equipment identified in their titles. The detailed criteria contained in these documents should be used in conjunction with the more comprehensive criteria of IEEE Std 323-1974 for evaluating the respective equipment environmental qualifications.

IEEE Std 317-1972, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.63), contains general guidance for qualification of penetration assemblies. Therefore, this document should be used in conjunction with IEEE Std 323-1974 for evaluating the environmental qualification of this equipment.

The criteria in IEEE Std 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of the Nuclear Power Generating Stations" (augmented by Regulatory Guide 1.30), are used by QAB to evaluate the quality assurance program described by the applicant. The quality assurance program is acceptable if it can be ascertained that the criteria of this standard and guide are met.

III. REVIEW PROCEDURES

This section of the review plan describes the essential elements of the review process including the use of the criteria and evaluation guides.

The review objective is to determine from the information presented in the SAR whether there is reasonable assurance that all items of safety-related electrical and mechanical equipment are capable of performing design safety functions under all normal and accident environmental conditions.

To achieve the objective, the review is divided into two distinct phases; the information audit phase and the evaluation phase. The audit phase is concerned with the completeness of the information presented. The evaluation phase is concerned with whether the required environmental capability will be or has been adequately demonstrated for each item of equipment. The two phases of the review process are performed as follows:

1. Information Audit Phase

   The review should determine that the following information is included:

   a. Equipment Identification

      All safety-related mechanical and electrical equipment must be identified. The equipment tabulations provided should be checked for completeness against the descriptions of safety-related systems contained in SAR Chapters 4, 5, 6, 7, 8, 9, 10, and 11. Definitions of the three categories of safety-related systems are contained in Standard Review Plan 7.1.

      The EICSB is responsible for verifying the completeness of the identification of all the electrical power, control, and instrumentation equipment. In addition, the EICSB confirms the equipment identification inputs of the secondary review branches.

      The secondary review branches are responsible for verifying the completeness of the identification of all mechanical equipment, and all electro-mechanical equipment located outside of the control room or other designated electrical equipment areas which pertain to the safety systems within their primary review responsibilities.

   b. Equipment Location

      The location of each item of safety-related equipment must be identified, both inside and outside the containment. Inside the containment, the location must specify whether inside or outside of the missile shield (for PWR's) or whether inside or outside of the drywell (for BWR Mark III's). Location of equipment is required in order to establish accurate definitions of both the normal and accident environments.

      The EICSB and the secondary review branches are responsible for verifying the location of the items of equipment identified by these branches in accordance with Section III.1.a above. The equipment locations are verified by review of the descriptions of the safety-related systems and the plant layout drawings in applicable sections of the SAR.

c. Normal and Accident Environmental Conditions

Both the normal and accident environmental conditions must be explicitly defined for each item of equipment. These definitions must include the following parameters: temperature, pressure, relative humidity, radiation, chemicals, and vibration (non-seismic).

For the normal environment, specific values should be provided. For the accident environment, these parameters should be presented as functions of time and the cause of the postulated environment (loss-of-coolant accident, steam line break, or other) should be identified.

The EICSB will verify that the normal and accident environments have been defined as indicated above for each item of equipment.

d. Time Required to Operate

The length of time that each item of equipment is required to operate in the accident environment must be provided. EICSB will verify the inclusion of this information. The secondary review branches will confirm the adequacy of the specified time interval for the equipment in their respective areas of primary review responsibility.

e. Environmental Qualification

The SAR should contain a complete description of the design bases and environmental qualification tests and analyses that have been (FSAR) or will be (PSAR) performed on each item of safety-related equipment. This should include qualification for the accident environments, qualification for extreme normal operating environments, and qualification to assure that loss of environmental control systems that are not classified as safety-related will not adversely affect the operability of safety-related equipment, particularly electrical equipment located in the control room and other control equipment rooms. The EICSB will confirm that this information is provided. The evaluation of the adequacy of the information is addressed in the following section of this review plan.

2. Evaluation Phase

The evaluation phase of the review involves the exercise of engineering judgement to determine from the information presented, particularly that regarding environmental qualification, whether an adequate demonstration of the required environmental capabilities of safety-related equipment will be or has been made. This phase of the review is performed after it has been established (by means of the information audit phase of the review previously described) that the information content requirements for Section 3.11 of the SAR have been satisfied. Although specifically written for use in evaluating the environmental qualification of Class

I electric equipment, IEEE Std 323-1974 contains principles and criteria that are comprehensive and generic to the qualification process itself; therefore, it is considered applicable to the environmental qualification of other types of equipment.

This phase of the review is performed as follows:

a. EICSB verifies that for each item of safety-related equipment, the environmental qualification program performed (FSAR) or proposed (PSAR) meets the detailed requirements of IEEE Std 323-1974, with particular emphasis on the following:
   (1) The accuracy and validity of the definitions of the normal and accident environments are verified by checking against the appropriate environmental control system design requirements for normal environments, and against the accident analyses with regard to accident environments resulting from loss-of-coolant accidents (LOCA) or steam or feedwater line breaks.
   (2) Type testing, or partial type testing in conjunction with one or more of the other methods, as defined in IEEE Std 323-1974, must be used for qualifying equipment for postulated accident environments. The qualification method used (type test, operating experience, analysis, combined qualification, or on-going qualification) should be identified. The corresponding requirements of IEEE Std 323-1974 then apply.
   (3) The type test must be designed to demonstrate that the equipment performance meets or exceeds the requirements of the equipment specifications for the plant, i.e., some margin must be demonstrated as indicated in IEEE Std 323-1974. Margin is demonstrated by increasing the levels of testing, the number of test cycles, and the test duration.
   (4) The test sequence, i.e., the order of application of the simulated environmental conditions (aging, radiation, vibration, etc.) during testing, must constitute the most severe sequence for the item being tested.
   (5) The equipment being type tested should be operated under design operating conditions and adequately monitored during testing to determine performance characteristics.
   (6) The equipment qualified by type testing must be prototypical of the actual equipment to be used in the plant. If this is not the case, a detailed analysis must be provided to justify the qualification.

   The criteria of IEEE Stds 317, 334, 382, and 383, and Regulatory Guides 1.40, 1.63, and 1.73 should be used, as applicable, in conjunction with IEEE Std 323 in evaluating the environmental qualification program.

b. The APCSB, CSB, and RSB evaluate the validity of the descriptions of both the normal and accident environments in those areas of the plant for which they have primary review responsibility. The normal environments are evaluated by means of a review of the design of the environmental control systems (ventilation, heating, cooling, air-conditioning); the accident environments by checking against the environmental conditions described in the accident analyses. The accident environments resulting from LOCA and from steam and feedwater line breaks are the responsibility of the RSB. The secondary review branches will advise EICSB of any inadequacy in the descriptions of the normal and accident environments.

c. The APCSB evaluates the validity of the description of the environment resulting from the loss of environmental control systems (ventilation, heating, cooling, air-conditioning) in those areas of the plant which contain safety-related equipment, including the control room and other electrical equipment rooms. This evaluation is performed by review of the design of the respective environmental contol systems and calculation of the environment resulting from failure of the systems. The APCSB will advise EICSB of any inadequacy in the descriptions of the environments resulting from the loss of environmental control systems.

d. The APCSB, CSB, and RSB evaluate the acceptability of values provided in the SAR for the length of time that safety-related equipment is required to operate in the accident environment. This evaluation is performed by checking against the particular system or equipment operating requirements as postulated in the accident analysis. The secondary review branches will advise EICSB if any of the equipment accident environment operating times listed in the SAR are unacceptable.

e. QAB reviews the environmental qualification program to verify that the test control, documentation, inspection, and material control requirements are in accordance with IEEE Std 336-1971 (as augmented by Regulatory Guide 1.30) and with the requirements of the quality assurance program described in Chapter 17 of the SAR. The objective of this review is to ascertain that the programs described provide adequate assurance that only environmentally qualified equipment will be installed in the plant and that this equipment will be properly installed.

IV. EVALUATION FINDINGS

The review should verify that sufficient information is contained in the SAR to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has identified all the safety-related mechanical and electrical equipment, defined the normal and postulated accident environments that this equipment may be subjected to, and described the environmental qualification program that has been (for FSAR) or will be (for PSAR) performed to demonstrate its required environmental capability. It is concluded from this information that there is assurance that all items of safety-related equipment will be capable of performing needed safety functions under normal and accident environmental conditions."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records;" Criterion 4, "Environmental and Missile Design Bases;" and Criterion 23, "Protection System Failure Modes."

2. 10 CFR Part 50, Appendix B, Section XI, "Test Control."

3. IEEE Std 279-1971 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

4. IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

5. *IEEE Std 317-1972, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

6. **IEEE Std 323-1974, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

7. *IEEE Std 334-1971, "Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

8. *IEEE Std 336-1971, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

9. *IEEE Std 382-1972, "Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

10. *IEEE Std 383-1974, "Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

11. *Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (this guide supplements IEEE Std 336-1971).

12. *Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants" (this guide supplements IEEE Std 334-1971).

13. *Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants" (this guide supplements IEEE Std 317-1972).

14. *Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" (this guide supplements IEEE Std 382-1972).

*Acceptance criteria or evaluation guidance.
**Basic acceptance criteria.

APPENDIX

STANDARD REVIEW PLAN 3.11

CHEMICAL AND RADIOLOGICAL ENVIRONMENT
IN CONTAINMENT DURING POSTULATED ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

Detailed methods of defining the radiological environment during postulated accidents are now under development by an IEEE standards committee for inclusion in IEEE-323. (Appendix A to IEEE-323 currently gives illustrative examples of environmental conditions but is not part of the standard.) When this standard has been completed, reviewed, and accepted by the staff, it will form the basis for evaluation. Review of source terms by the AAB will then be required only if unusual situations arise. Until the IEEE standard is available, the staff review of the chemical and radiological environment in the containment during postulated accidents will be in accordance with this appendix. This review is implemented primarily by comparing the applicant's proposed chemical and radiological source terms with those previously computed for similar plants. The purpose of this review is to assure that safety equipment inside containment will function in design basis accident environments.

II. ACCEPTANCE CRITERIA

1. The applicant's estimate of the chemical environment is acceptable if it reflects the chemical composition of all fluids and additives present in the primary system or added to the containment environment in the course of the accident for various modes of equipment operation.

2. The applicant's estimate of the radiation environment is acceptable if it reflects source terms comparable to those postulated in Regulatory Guides 1.3, 1.4, and 1.7 (Refs. 1, 2, 3) and results in equipment exposure levels similar to those presented in other applications and checked by independent staff calculations. The radiological source term for qualification tests in a radiation environment for pressurized water reactor (PWR) and boiling water reactor (BWR) equipment, such as pumps and seals, which normally is exposed to a water environment, should be based on the same source terms as given in Reference 3, i.e., 50% of the halogens and 1% of the solid fission products present in the core are intimately mixed with the coolant water. For PWR and BWR equipment, such as instrumentation

From the source term information, the reviewer may calculate the radiation dose rates and integrated doses in the containment, ESF filters, and in equipment rooms housing ESF components. For exposed organic material in ESF systems, a source term for both beta and gamma radiation is used. The methods, techniques, and appropriate data to be used in the calculations can be found in radiation shielding references such as those listed in References 6 through 8. The results are compared with those of the applicant. The evaluation findings of the chemical and radiation environmental source terms are given to EICSB and MEB when there is a disagreement with the applicant's submittal.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for each particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The reviewer confirms that the estimates of chemical and radiation environments given by the applicant are comparable with those of similar plants recently reviewed and approved or are comparable to those that may be determined by an independent calculation on a typical plant. If an independent calculation is determined to be necessary, the procedure outlined below may be followed.

1. Chemical Environment

   The chemical environment inside the containment can be established by considering the total quantity of injection liquid and the total quantity of additives (e.g., NaOH, Na_2SO_3, N_2H_2). From this information the reviewer may calculate the weight and volume percent of the additive. The pH of the resulting solution can be calculated for appropriate combinations of equipment operation using generally accepted values of dissociation constants (Ref. 4). (This information should be cross-checked with Section 6.5.2.9 of the applicant's safety analysis report.) See also Standard Review Plan (SRP) 6.5.2 and SRP 6.1.3.

2. Radiation Environment

   A radiation source term consistent with Regulatory Guides 1.3, 1.4, and 1.7 (Refs. 1, 2, 3) is assumed as appropriate to the air or water environment under consideration. If an independent calculation is desirable, the ORIGEN computer code (Ref. 5) may be used to calculate the core inventory as a function of burnup. The construction of the source term is based on the use of the maximum activity reached by each of the selected radionuclides. Calculations may be made independently for each environment (water and containment air) because conservative fission product assumptions for one environment may be nonconservative for another. The average energy of the fission product radiations and the total number of curies can be calculated from the information given in the ORIGEN output; this information is calculated for 0 to 30 days after shutdown in one-day increments. Separate energies for beta and gamma radiations are derived when this calculation is made.

When the IEEE standard is developed and reviewed by the staff, both the standard and position C.2 as given in the draft, dated April 7, 1975, of Regulatory Guide 1.89, Revision 1, will form the basis for evaluation. Individual review and independent calculation of the radiation environment at that point will be required only in exceptional cases.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant's chemical and radiation source terms that define the environmental conditions to be used in design of the ESF mechanical and electrical equipment are appropriate for the postulated design basis accidents."

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2.

3. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

4. "Handbook of Chemistry and Physics,: The Chemical Rubber Co., Cleveland, Ohio. (Any recent edition.)

5. M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laboratory, May 1973.

6. T. Rockwell, "Reactor Shielding Design Manual," D. Van Nostrand Co., Princeton, New Jersey (1956).

7. R. E. Malenfant, "QAD - A Series of Point-Kernal General-Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratories, October 1966.

8. R. Jaeger, Ed., "Engineering Compendium on Radiation Shielding. Volume 1, Shielding Fundamentals and Methods," Springer-Verlag, New York (1968).

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The mechanical, thermal, and chemical design of the fuel assembly is evaluated by CPB. The fuel assembly is generally a square array of fuel rods (varying from 36 rods to 264 rods) which are mechanically secured together. The fuel rods are laterally supported by grid subassemblies at intervals along their length to maintain the assembly geometry. Some fuel assemblies allow control rods to be inserted within the square array. Those parts of the control rods which are inserted into the core although not considered as part of the assembly will be evaluated under this section. The fuel assembly is considered to include fuel pellets, burnable poisons, fill gas, getters, cladding, springs, end closures, spacer grids and springs, end fittings, guide thimbles, and channel boxes.

The review considers specific aspects of fuel behavior which affect and limit the safe and reliable operation of the plant. Steady state, anticipated reactor transient, and design basis accident conditions, including loss-of-coolant accidents (LOCA), are evaluated for both initial and reload cores. The specific aspects of interest are listed below:

1. The cladding mechanical property limits are reviewed. Mechanical properties include Young's modulus, Poisson's ratio, design dimensions, and allowable tolerances on wall thickness, diameters, and ovality as well as material strength and ductility properties. Yield and ultimate strength, uniform and total ductility, and creep rupture limits must reflect the effects of temperature and neutron fluence on these properties. Dimensional changes due to temperature, pressure, and neutron effects are reviewed.

2. The design against fatigue failure from either flow-induced vibration or power cycling is reviewed for spacer grids, fuel rods, springs, guide thimbles, and flow channel boxes. The consideration of stress levels, amplitudes of vibration, and life fraction are included. The form of the design criteria used may be curves of strain or stress amplitude versus the number of cycles.

3. The review includes an evaluation of the predicted time to cladding creep collapse into a fuel stack axial gap or the gas plenum. The appropriate design creep rate for predicting radial fuel-clad gap closure is considered separately from that for creep collapse because the safety implications differ.

4. The analytical model for fuel densification is evaluated, including both the extent and the kinetics of densification during operation. The effects of densification, which may cause changes in the stored energy, linear thermal output, axial gap, and thermal impedance are evaluated in the fuel design review (Ref. 2).

5. The fuel system is reviewed for maximum permissible power density to assure the appropriate margin between anticipated duty and the power density at which fuel rod failure would be expected. The permissible power densities should include local peaking as affected by anticipated transients.

6. The total internal pressure in the fuel rod is evaluated to assure the adequacy of the gas plenum design against rod burst. Additionally, the internal pressure calculations are reviewed for the effects of internal pressure on predictions of flow blockage during transients and accidents.

7. The potential for adverse chemical interactions either among the fuel assembly components or between a fuel component and the reactor environment is evaluated. The potential for adverse chemical interactions among the control rod subassembly components must be evaluated.

8. The fuel system design and the control rod subassembly design are evaluated for the physically feasible combinations of chemical, thermal, mechanical, and hydraulic interaction. Evaluation of these interactions includes the effects of normal reactor operation, anticipated transients, and postulated design accidents. Examples of possible interactions are: fuel-cladding mechanical interaction, fuel fission product-cladding attack, stress-accelerated corrosion, fretting corrosion, fuel rod burn-out, crevice corrosion, crud deposition, material wastage due to mass transfer, axial thermal expansion of fuel against collapsed cladding, and thermal and creep-induced dimensional changes.

9. The fuel system design is reviewed to assure that the appropriate physical and thermal properties for the materials used are being employed. These properties include thermal expansion (may be direction dependent), thermal conductivity, thermal diffusivity, specific heat, specific gravity, and temperatures of phase changes.

10. The potential for subassembly flow blockage arising from either external or internal causes is reviewed.

11. The review includes the effects of shock loadings (including LOCA) on both the fuel assembly geometry and fuel rod integrity. The effects of combined shock and seismic loads are analyzed.

12. The completeness of the applicant's design analysis is reviewed to assure that all criteria and the appropriate margins have been considered. The analysis is reviewed to assure that some surveillance of actual performance is included as a verification of the design.

13. The applicant's proposed technical specifications related to areas covered in this plan are reviewed for operating license (OL) cases.

The primary review responsibility rests with the Core Performance Branch. Other branches provide assistance as requested by CPB. The QAB provides consultation on matters concerning the representative nature of test results and the characterization of the component materials. The MEB provides consultation both on the interaction of the fuel assembly with adjacent core components and on the applied mechanics used in design. In addition, the Advanced Program Development Branch of the Office of Inspection and Enforcement may be consulted by CPB on fuel vendor practices and reactor performance of specific design features.

II. ACCEPTANCE CRITERIA

The general purpose of the review is to establish that all safety-related aspects of the fuel system design have been adequately considered and that the proposed fuel design limits have appropriate margin and are acceptable, as required by General Design Criterion 10 (Ref. 1). The specific criteria for the fuel system design are listed below:

1. The fuel cladding mechanical properties used in the design should be consistent with generally accepted values and characteristics of the material.

2. The general membrane stress limits for the cladding must be reasonably less than the corresponding material strengths for the design service temperatures and neutron fluences. The procedure for calculation of the maximum cladding strain fatigue should be one approved by the staff.

3. The cumulative number of strain fatigue cycles should be significantly less than the design fatigue life of the particular material. For example, design allowances may be based on appropriate data which has been modified by a factor of 2 on stress amplitude or of 20 on the number of cycles (Ref. 3).

4. The predicted time to cladding creep collapse should be compatible with the allowable peak cladding temperature (PCT) for LOCA analysis. When no zircalloy cladding collapse is expected, the calculated PCT should be less the 2200°F. For reactor service beyond the predicted time to collapse, the calculated PCT should be less than 1800°F. The analytical model used for the prediction should be one approved by the staff. Staff approval of a model will be based in part on a comparsion with results from a staff creep collapse code, e.g., BUCKLE (Ref. 4), or COVE (Ref. 5).

5. The analytical thermal performance model for the fuel should be one approved by the staff and should include the effects of fuel densification, fission gas release, and

burnable poisons. The size and probability of fuel column axial gaps should be predicted by an approved method. To be approved, the analytical model should be capable of predicting appropriate test data and be corroborated by a staff thermal performance code, e.g., GAPCON (Ref. 6). The results of calculations with the model should show compliance with design limits such as fuel temperatures and maximum stored energy.

6. The maximum power density in the fuel should be less than the value at which fuel rod failure is predicted. A margin should be included that allows for calculational uncertainty, experimental error, and operational transients.

7. The calculated differential pressure across the fuel rods cladding during normal in-reactor service should be less than the pressure at which cladding failure would be expected.

8. The calculations for waterlogged rods during anticipated transients should include the hydrostatic pressure contribution of the contained water. Two elements to be considered in the analysis are: (a) the amount of water available inside the cladding and (b) the rate of change of temperature during a transient. The amount of water in a waterlogged rod may be determined either by inspection and test data or by a bounding calculation which determines the amount of water to equalize the system and internal pressures. The appropriate rate of change of temperature for rupture may be determined by test data, e.g., from the SPERT tests (Ref. 7).

   The flow blockage associated with rupture from internal pressure should be consistent with appropriate test data.

9. The potential adverse chemical interactions should be considered on the basis of satisfactory operating experience of similar designs and other appropriate date.

10. Fuel system thermal-mechanical interactions may be evaluated by fuel behavior codes such as LIFE-II (Ref. 8), CYGRO (Ref. 9), or FRAP (Ref. 10). The results of analyses of fuel-clad mechanical interaction should compare favorably with correlations of data relating fuel performance and power density conditions that would occur during normal and transient operations. The design provisions for prevention of excessive fretting should be shown to be adequate by data from design verification or proof tests.

11. The mechanical aspects of flow blockage should be determined by examination of appropriate data. Analysis of the thermal aspects of flow blockage should be done by methods approved by the staff as a part of the reactor thermal-hydraulic design review, or previously approved in case or generic reviews.

12. Design methods for predicting creep deformation and plasticity should be verified against appropriate test data approved by the staff. Values of creep deformation to be used for design purposes generally require some margin from predicted values, to account for the scatter inherent in creep data. The magnitude and direction of the margin depends upon both the extent of scatter of the data and the design application.

When a creep prediction is used in gap conductance calculations, it is acceptable if it underpredicts a significant fraction of the appropriate test data. When a creep prediction is used for cladding collapse calculations, it is acceptable if it overpredicts measured deformations.

13. Calculations of the effects of shock loadings on the fuel, including those from LOCA, should be based upon established methods and codes. The methods may be either time history, shock spectrum, or statistical, and should include any environmental degradation effects in the material.

14. The completeness of the design analyses should be demonstrated by a listing of all design criteria, the corresponding design values, and the "best engineering estimates" for normal operating values. The criteria, design values, and "best estimates" may be in the form of stresses, strains, times, or cycles. Results from or plans for a surveillance program should provide a reasonable means of verifying the actual fuel performance.

15. The design must assure that reactivity control materials remain below their melting point and that it provides a means of accommodating or venting gaseous fission products.

III. REVIEW PROCEDURES

The reviewer should assure that the intent of each of the acceptance criteria of Section II has been complied with fully. The assurance is provided by a systematic evaluation of the design against each criterion above. The various aspects of the design may be considered adequate based upon corroborating computer code calculations, confirmatory hand calculations, generally accepted engineering conventions and industry standards, comparisons with appropriate data, or results of operating experience. A list of commonly used codes, standards, and specifications is given in Table 4.2-1, for information only. The reviewer should assure that:

1. The data base used for the fuel system design is applicable to the particular design.

2. The design parameters pertinent to safety have been appropriately considered in relation to each particular design aspect.

3. The expected variance in parameter values has been accommodated.

Most of the detailed safety review of fuel systems designs is accomplished on a continuing generic basis outside the docketed applications. Thus, there are no unique review procedures for the evaluation of a fuel system design. The CPB deals directly with fuel vendors and evaluates the engineering methods employed in each aspect of a fuel system design (Refs. 11-17). Consequently, much of the review procedure in evaluating a specific plant is directed to assuring that the design methods used have been approved by the staff and are being correctly applied.

The full scope of a fuel system design safety evaluation at the operating license (OL) stage is covered in this plan. The safety evaluation at the construction permit (CP) stage need not be specific in all aspects of the fuel system design. Those aspects that may change between the CP and OL stages need not be addressed, e.g., degree of fuel densification, cladding mechanical properties, level of prepressurization, and as-manufactured dimensions. The primary interests at the CP stage are:

1. The completeness of the design analysis should be assured, such that all the design criteria have been or will be addressed.

2. The engineering methods being employed in the design should be either already approved by the staff or review of the methods should have progressed sufficiently that approval may be reasonably anticipated prior to the OL stage.

3. The proposed fuel system design should be consistent with that of previously approved plants of the same type.

Review of the technical specifications related to the fuel system is carried out as part of the review for operating license applications. Appropriate technical specifications for limiting power density values are developed in the review. Various aspects of fuel systems components that must be considered in the design and evaluated by the reviewer in the OL review are listed below.

1. For the fuel cladding, the design must consider dimensions, composition, thermal-mechanical processing, and the optimum strength and ductility capabilities of the tubing for the expected duty. The cladding design should be such as to accommodate the fission gas evolved in operation, so that the fuel can reach design burnup without exceeding the cladding structural design criteria. These design aspects require adequate plenum volume and cladding thickness, including allowances for surface defects and manufacturing tolerances. Cladding design requires calculations of mechanical limits, e.g., by computer codes such as BUCKLE (Ref. 4) and COVE (Ref. 5) and of the effects of operation on cladding geometry, e.g., by computer codes such as FRAP (Ref. 10) and CYGRO (Ref. 9). Computer models which have been indexed to appropriate test data may be used by the reviewer, e.g., the computer codes GAPCON (Ref. 6) and LIFE-II (Ref. 8).

2. For the fuel assembly, the design must consider rigidity during shock loading, hydraulic loading, and transportation loadings both before and after reactor service. The potential dimensional changes of components resulting from thermal, chemical, mechanical, and irradiation-induced degradation; loads applied by the core restraint system; and loads applied during grappling (in fuel handling), including those from misaligned handling tools, are considered in the fuel assembly performance review. The assembly end fittings must properly mate with the assembly positioning system and preclude misorientation or mislocation of the assembly.

3. The spacer grid design must consider spring loads, dimensional tolerances, materials, and joining methods. The design must consider axial and radial growth due to temperatures, burn-up, and neutron irradiation. The spacer must prevent radial oscillations, allow adequate cooling by maintaining the specified pitch to diameter ratio, and remain chemically compatible with the fuel cladding material.

4. For oxide fuel, the design must consider the size, shape, density, and composition of the pellets. The size considerations include the effects of temperature, density, and thermal performance margins. The shape is dependent upon design exposures, anticipated methods of operation, and cladding characteristics. Design densities are affected by thermal performance, fuel rod lengths, manufacturing variations, and pellet shape. The pellet composition requires consideration of nuclear and thermal performance requirements and compatibility with other fuel rod components during the anticipated service, including the effects of burnable neutron poisons. The complexity and interaction of all these components necessitates the use of sophisticated analytical computer models.

5. For springs, the design must consider the dimensions required for the requisite positioning of components. Spring dimensional considerations include allowances for thermal and irradiation-enhanced stress relaxation. Of particular importance is the cumulative effect of a series of springs.

IV. EVALUATION FINDINGS

The general scope of the review of the fuel system design is the same at the CP stage as at the OL stage. However, the review for an OL is more detailed and specific than for a CP. At the OL stage, the generically approved design methods and codes, the detailed materials properties, and the appropriate reactor environmental conditions for the as-fabricated fuel system are utilized to establish that the particular plant has met the design criteria. The staff's safety evaluation report (SER) should reflect this difference. The following typical evaluation findings are given for OL and CP reviews:

1. Operating License

"The fuel system for the ___________ plant includes the fuel assembly, which is composed of ___________ fueled rods, ___________ nonfueled tubes, ___________ spacer grids, and end-fittings. The fuel rod includes fuel pellets, plenum springs, cladding, end closures, and thermal and chemical buffers. The basic mechanical function of the fuel system is to provide a controlled core geometry during normal operations, anticipated transients, and accidents. The review has considered the specific aspects of fuel system behavior which affect and can limit the safe, reliable operation of the plant.

"The evaluation of the fuel system mechanical design was based upon mechanical tests, in-reactor operating experience, and engineering analyses. Additionally, the in-reactor performance of the fuel system design will be subject to the continuing surveillance programs of the fuel system vendor, the staff, and the applicant. These programs provide confirmatory performance information. In reviewing the engineering analyses, the applicability of the design criteria and the rigor of the applied methods were evaluated to confirm compliance with design objectives.

"Part of the basis for acceptance in the staff review has resulted from a systematic evaluation of the fuel system design with regard to the design criteria. The engineering analyses are considered adequate based upon corroborating computer code calculations using staff-approved methods, confirmatory hand calculations, generally accepted engineering conventions and standards, and comparisons with appropriate test data.

"Further bases for acceptance are favorable results of out-of-reactor mechanical tests and in-reactor performance of directly comparable fuel systems. The staff has reviewed the tests and found the quality of the reactor simulation adequate for that aspect of the fuel system being examined. Fuel systems of similar design have been successfully irradiated for up to ________ years and have had peak exposures of ________ MWD/MT.

"The staff concludes that, based upon operating experience with similar fuel systems, results of out-of-reactor tests, technical specification requirements to monitor and limit off-gas and effluent activity, and the continuance of a fuel rod surveillance program including destructive and non-destructive post-irradiation examinations, the integrity of the fuel system will be maintained during both normal operations and incidents of moderate frequency, and the proposed fuel design limits are adequate and acceptable, as required by General Design Criterion 10. Further, we conclude that accidents or earthquake-induced loads will not result in an inability to cool the fuel ________ or significant interference with control rod insertion."

2. Construction Permit

(Some paraphrasing of the first paragraph for an OL).

"The analytical models employed by the applicants have been shown to be acceptable by comparison with measurements on fuel rods which have been subjected to reactor operating conditions. These models, described in topical reports, are based on data for fuel similar to that proposed for use in ___________. These analytical models, which have been reviewed in detail by the staff, provide acceptable assessments of the anticipated fuel rod behavior.

"On the basis of our review of the proposed analytical models and the confirmatory results from tests on irradiated fuel rods, we have concluded that, (1) the fuel rod mechanical design will provide acceptable engineering safety margins for normal operation, and (2) the effects of densification will be acceptably accounted for in the fuel design."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. Regulatory Staff, "Technical Report on Densification of Light Water Reactor Fuels," U. S. Atomic Energy Commission, November 14, 1972.

3. W. J. O'Donnel and B. F. Langer, Nuclear Science and Engineering, Vol 20, 1-12 (1964).

4. P. J. Pankaskie, "BUCKLE, An Analytical Computer Code for Calculating Creep Buckling of an Initially Oval Tube," BNWL-B-1974, Battelle Northwest Laboratories, May 1974.

5. C. Mohr, "COVE, A Finite Difference Creep Collapse Code for Oval Fuel Pin Cladding Material," BNWL-1896, Battelle Northwest Laboratories, March 1975.

6. C. R. Hann, C. E. Beyer, and L. J. Prachen, "GAPCON Thermal-1: A Computer Program for Calculating the Gap Conductance in Oxide Fuel Pins," BNWL-1778, Battelle Northwest Laboratories, September 1973.

7. L. A. Stephan, "The Effects of Cladding Material and Heat Treatment on the Response of Waterlogged UO_2 Fuel Rods to Power Bursts," IN-ITR-III, Idaho Nuclear Corporation, January 1970.

8. V. Z. Jankus and R. W. Weeks, "LIFE-II - A Computer Analysis of Fast Reactor Fuel Element Behavior As a Function of Reactor Operating History," Nuclear Engineering and Design, Vol. 18, No. 1, January 1972.

9. E. Duncombe, C. M. Friedrich, and J. K. Fischer, "CRYGRO 3 - A Computer Program to Determine Temperature, Stress, and Deformations in Oxide Fuel Rods," WAPD-TM-961, Bettis Atomic Power Laboratory, March 1970.

10. J. A. Dearien and L. J. Siefken, "FRAP-T: A Computer Program for Calculating the Transient Response of Oxide Fuel Rods," I-243-3-51.1, Aerojet Nuclear Company, (to be published).

11. Regulatory Staff, "Technical Report on Densification of Babcock and Wilcox Reactor Fuels," U. S. Atomic Energy Commission, July 6, 1973.

12. Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission, in the matter of Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant Units 1 & 2, Dockets 50-317/318, Supplements 1 and 2, October 1973.

13. Regulatory Staff, "Technical Report of Densification of Exxon Nuclear BWR Fuels," U.S. Atomic Energy Commission, September 3, 1973.

14. Regulatory Staff, "Technical Report on Densification of General Electric Reactor Fuels," U.S. Atomic Energy Commission, August 23, 1973, and Supplement 1, December 14, 1973.

15. Regulatory Staff, "Technical Report on Densification of Gulf United Nuclear Fuels Corporation Fuels for Light Water Reactors," U. S. Atomic Energy Commission, November 21, 1973.

16. Regulatory Staff, "Technical Report on Densification of Westinghouse PWR Fuel," U.S. Atomic Energy Commission, May 14, 1974.

17. Regulatory Staff, "Technical Report on Densification of Combustion Engineering Reactor Fuels," U.S. Atomic Energy Commission, August 19, 1974.

TABLE 4.2-1

REFERENCE CODES, STANDARDS, AND SPECIFICATIONS

CODE, STANDARD, OR SPECIFICATION		TITLE
ASME		Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components
ASTM	E-8	Tension Testing of Metallic Materials
ASTM	E-21	Short Time Elevated Temperature Tension Testing of Materials.
ASTM	E-112	Estimating Average Grain Size of Metals
ASTM	G-2	Aqueous Corrosion Testing of Samples of Zirconium and Zirconium Alloys
ASTM	E-29	Indicating Which Place of Figures are to be Considered Significant in Specified Limiting Values
MIL-STD-105D		Sampling Procedures and Tables for Inspection by Attributes
ASTM	A-370	Mechanical Testing of Steel Products
ASTM	A-393	Recommended Practice for Conducting Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steels
ASTM	A-262	Recommended Practice for Detecting Susceptibility to Intergranular Attack in Stainless Steel
ASTM	E-94	Recommended Practice for Radiographic Testing
RDT	M3-28T	Austenitic Stainless Steel Tubing for LMFBR Core Components
RDT	M1-16T	Zirconium and Zirconium Alloy Bare Welding Rods
RDT	M2-9T	Zirconium and Zirconium Alloy Forgings and Extrusions
RDT	M5-6T	Zirconium and Zirconium Alloy Plate, Sheet, and Strip
RDT	M7-9T	Zirconium and Zirconium Alloy Bars, Rod, and Wire
RDT	M-10-1T	Zirconium and Zirconium Alloy Ingots
Bureau of Mines		Helium Grade A Specification

11/24/75

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The review of the nuclear design of the fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core.

The review of the nuclear design under this plan, the review of the fuel system design under Standard Review Plan (SRP) 4.2, the review of the thermal and hydraulic design under SRP 4.4, and the review of the accident analyses under the SRP for Chapter 15 of the applicant's safety analysis report (SAR), are all necessary in order to confirm that the requirements defined above are met.

The specific areas of interest in the nuclear design include:

1. Confirmation that design bases are established as required by the appropriate general design criteria.

2. The areas concerning core power distribution. These are:

   a. The presentation of expected or possible distributions including normal and extreme cases for steady state and allowed load-follow transients and covering a full range of reactor conditions of time in cycle, allowed control rod positions, and possible fuel burnup distributions. The power distributions should include power spikes from fuel densification.

   b. The presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in accident analyses.

c. The translation of the design power distributions into operating power distributions, including instrument-calculation correlations, operating procedures and measurements, and necessary limits on these operations.

d. The requirements for instruments, the calibration and calculations involved in their use, and the uncertainties involved in translation of instrument readings into power distributions.

e. Limits and setpoints for actions, alarms, or scram for the instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits.

f. Measurements in previous reactors and critical experiments and their use in the uncertainty analyses, and measurements to be made on the reactor under review, including startup confirmatory tests and periodically required measurements.

g. The translation of design limits, uncertainties, operating limits, instrument requirements, and setpoints into technical specifications.

3. The areas concerning reactivity coefficients. These are:

a. The applicant's presentation of calculated nominal values for the reactivity coefficients such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power, and the extremes reached in transient and accident analyses. It includes the extremes of time in cycle and an appropriate range of control rod insertions for the reactor states.

b. The applicant's presentation of uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude by examination of the accuracy of the methods used in calculations (SAR Section 4.3.3), and comparison where possible with reactor experiments.

c. The applicant's combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady state analysis (primarily control requirements, SAR Section 4.3.2.4), stability analyses (SAR Section 4.3.2.8), and the transient and accident analyses presented in SAR Chapter 15.

4. The areas concerning reactivity control requirements and control provisions. These are:

a. The control requirements and provisions for control necessary to compensate for long term reactivity changes of the core. These reactivity changes occur because of depletion of the fissile material in the fuel, depletion of burnable poison in some of the fuel rods, and buildup of fission products.

b. The control requirements and provisions for control needed to compensate for the reactivity change caused by changing the temperature of the reactor from the hot, zero power condition to the cold shutdown condition.

c. The control requirements and provisions for control needed to compensate for the reactivity effects caused by changing the reactor power level from full power to zero power.

d. The control requirements and provisions for control needed to compensate for the effects on the power distribution of the high cross section Xe135 isotope.

e. The adequacy of the control systems to insure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation.

f. The applicant's analysis and experimental basis for determining the reactivity worth of a "stuck" control rod of highest worth.

g. The provision of two independent control systems.

5. The areas of control rod patterns and reactivity worths. These are:

a. Descriptions and figures indicating the control rod patterns expected to be used throughout a fuel cycle. This includes operation of single rods or of groups or banks of rods, rod withdrawal order, and insertion limits as a function of power and core life.

b. Descriptions of allowable deviations from the patterns indicated above, such as for misaligned rods, stuck rods, or rod positions used for spatial power shaping.

c. Descriptions, tables, and figures of the maximum worths of individual rods or banks as a function of position for power and cycle life conditions appropriate to rod withdrawal transients and rod ejection or drop accidents. Descriptions and curves of maximum rates of reactivity increase associated with rod withdrawals, experimental confirmation of rod worths or other factors justifying the reactivity increase rates used in control rod accident analyses, and equipment, administrative procedures, and alarms which may be employed to restrict potential rod worths should be included.

d. Descriptions and graphs of scram reactivity as a function of time after scram initiation and other pertinent parameters, including methods for calculating the scram reactivity.

6. The area of criticality of fuel assemblies. Discussions and tables giving values of K_{eff} for single assemblies and groups of adjacent fuel assemblies up to the number required for criticality, assuming the assemblies are dry and also immersed in water, are reviewed.

7. The areas concerning analytical methods. These are:

   a. Descriptions of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup, and stability.

   b. The data base used for neutron cross sections and other nuclear parameters.

   c. Verification of the analytical methods by comparison with experiments.

8. The areas concerning pressure vessel irradiation. These are:

   a. Neutron flux spectrum above 1 MeV in the core, at the core boundaries, and at the inside pressure vessel wall.

   b. Assumptions used in the calculations; these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel.

   c. Computer codes used in the analysis.

   d. The data base for fast neutron cross sections.

   e. The geometric modeling of the reactor, support barrel, water annulus, and pressure vessel.

   f. Uncertainties in the calculation.

The RSB reviews the adequacy of limits on power distribution during normal operation in connection with their analyses of the thermal-hydraulic design, anticipated operational occurrences, and accidents, under SRP 4.4 and the plans for Chapter 15 of the SAR.

The EICSB, under the plans for SAR Chapter 7, reviews the adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating state within defined limits.

II. ACCEPTANCE CRITERIA

1. The basic acceptance criteria in the area of nuclear design are the general design criteria (GDC) related to the reactor core and reactivity control systems.

   a. GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

   b. GDC 11 requires that in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

   c. GDC 12 requires that power oscillations which could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

d. GDC 13 requires provision of instrumentation and controls to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation and accident conditions, and to maintain them within prescribed operating ranges.

e. GDC 20 requires automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions.

f. GDC 25 requires that no single malfunction of the reactivity control system (this does not include rod ejection or dropout) cause violation of the acceptable fuel design limits.

g. GDC 26 requires that two independent reactivity control systems of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

h. GDC 27 requires that the reactivity control systems have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

i. GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

2. The following discussions present less formal criteria and guidelines used in the review of the nuclear design.

a. There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady state or load-following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7) and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 2200°F peak cladding temperature allowed for loss-of-coolant accidents (LOCA), are not exceeded during design basis accidents. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation.

It is a branch position that these limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed.

The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that:

(1) A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.

(2) A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).

Branch positions on acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram, frequency and extent of power distribution measurements, and use of excore and incore instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review (Ref. 2). The CPB has enunciated a branch technical position for Westinghouse reactors which employ constant axial offset control (Ref. 7).

Acceptance criteria for power spike models can be found in staff technical reports on fuel densification (Ref. 3).

Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.

b. The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity", and is considered to be satisfied in light water reactors by the existence of the Doppler and negative power coefficients. There are no criteria or branch positions that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient such as may exist in pressurized water reactors at beginning of core life.

   The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgment to be made under this plan is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations; experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.

c. Acceptance criteria relative to control rod patterns and reactivity worths include:

   (1) The control rod worths and reactivity insertion rates predicted in this section must be reasonable bounds to values that may occur in the reactor. These values are used in the accident analysis and judgment as to the adequacy of the uncertainty allowances are made in the review of the accident analyses.

   (2) Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a CPB position to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.

d. There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and data base should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate and encompass a sufficient range.

III. REVIEW PROCEDURES

The review procedures below apply in general to both the construction permit (CP) and operating license (OL) stage reviews. At the CP stage, parameter values and certain design aspects may be preliminary and subject to change. At the OL stage, final values of parameters should be used in the analyses presented in the SAR. The review of the nuclear design of a plant is based on the information provided by the applicant in the safety analysis report, as amended, and in meetings and discussions with the applicant and his contractors and consultants. This review in some cases will be supplemented by independent calculations performed by the staff or staff consultants.

1. The reviewer confirms, as part of the reviews of specific areas of the nuclear design outlined below, that the design bases, design features, and design limits specified by the GDC listed in Section II are established in conformance with those GDC.

2. The reviewer examines the information presented in the SAR to determine that the core power distributions for the reactor can reasonably be expected to fall within the design limits throughout all normal (steady state and load-follow) operations, and that the instrument systems employed, along with the information processing systems and alarms will reasonably assure the maintenance of the distributions within these limits for normal operation.

   For a normal review, many areas related to core power distribution will have been examined in generic reviews or earlier reviews of reactors with generally similar core characteristics and instrument systems. A large part of the review on a particular case may then involve comparisons with information from previous application reviews. The comparisons may involve the shapes and peaking factors of normal and limiting distributions over the range of operating states of the reactor, the effects of power spikes from densification, assigned uncertainties and their use, calculation methods and data used, correlations used in control processes, instrumentation requirements, information processing methods including computer use, setpoints for operational limits and alarm limits, and alarm limits for abnormalities such as flux asymmetries.

   An important part of this review, at the OL stage, covers the relevant sections of the proposed technical specifications, where power distributions and related controls such as control rod limits are discussed. Here the instrument requirements, limit settings, and measurement frequencies and requirements are set forth in full detail. The comparison of technical specifications should reveal any differences between essentially identical reactors or any lack of difference between reactors with changed core characteristics. Where these occur, the reviewer must assess the significance and validity of the differences or lack of differences.

   This review and comparison may be supplemented with examinations of related topical reports from reactor vendors, generic studies by staff consultants, and startup reports from operating reactors which contain information on measured power distributions (Ref. 4). Multigroup computer calculations by the reviewer or staff consultants are not done as a part of the normal review.

3. The reviewer determines from the applicant's presentations that suitably conservative reactivity coefficients have been developed for use in reactor analyses such as those for control requirements, stability, and accidents. The reviewer examines:

   a. The applicability and accuracy of methods used for calculations including the use of more accurate check calculations such as the use of Monte Carlo techniques for Doppler models.

   b. The models involved in the calculations such as the model used for effective fuel temperature in Doppler coefficient analyses.

   c. The reactor state conditions assumed in determining values of the coefficients. For example, the pressurized water reactor (PWR) moderator temperature coefficient to be used in the steam line break analysis is usually based on the reactor condition at end of life with all control rods inserted except the most reactive rod, and the moderator temperature in the hot standby range.

   d. The applicability and accuracy of experimental data from critical experiments and operating reactors used to determine or justify uncertainty allowances. Measurements during startup and during the cycle of moderator temperature coefficients and full power Doppler coefficients in the case of PWR's, and results of measurements of transients during startup in the case of boiling water reactors (BWR's), should be examined. As part of the review, comparisons are made between the values and uncertainty allowances for reactivity coefficients for the reactor under review and those for similar reactors previously reviewed and approved. Generally, many essential areas will have been covered during earlier reviews of similar reactors. The reviewer notes any differences in results for essentially identical reactors and any lack of differences for reactors with changed core characteristics, and judges the significance and validity of any differences or lack of differences.

   e. In special cases, audit calculations may be performed by the reviewer or staff consultants in specific areas to confirm the applicant's analyses. CPB maintains files of generic audit calculations made by staff consultants, for reference by the reviewer (Ref. 5).

4. The review procedures in the area of reactivity control requirements and control provisions are as follows:

   a. The reviewer determines that two independent reactivity control systems of different design are provided.

   b. The reviewer examines the tabulation of control requirements, the associated uncertainties, and the capability of the control systems, and determines by inspection and study of the analyses and experimental data that the values are realistic and conservative.

c. The reviewer determines that one of the control systems is capable of returning the reactor to the cold shutdown condition and maintaining it in this condition, at any time in the cycle. It is necessary that proper allowance be made for all of the mechanisms that change the reactivity of the core as the reactor is taken from the cold shutdown state to the hot, full power operating state. The reviewer should determine that proper allowance is made for the decrease in fuel temperature, moderator temperature, and the loss of voids (in BWR's) as the reactor goes from the power operating range to cold shutdown.

d. The reviewer determines that one of the control systems is capable of rapidly returning the reactor to the hot standby (shutdown) condition from any power level at any time in the cycle. This requirement is met by rapid insertion of control rods in all current light water reactors. Proper allowance for the strongest control rod being stuck in the full-out position must be made. In PWR's, operational reactivity control is carried out by movement of control rods and by adjustments of the concentration of soluble poison in the coolant. The reviewer must pay particular attention to the proposed rod insertion limits in the power operating range, to assure that the control rods are capable of rapidly reducing the power and maintaining the reactor in the hot standby condition. This is an important point because the soluble poison concentration in the coolant could be decreased in order to raise reactor power, while the control rods were left inserted so far that in the event of a scram (rapid insertion of control rods), the available reactivity worth of the control rods on full insertion would not be enough to shut the reactor down to the hot standby condition.

e. The reviewer determines that each of the independent reactivity control systems is capable of controlling the reactivity changes resulting from planned, normal power operation. This determination is made by comparing the rate of reactivity change resulting from planned, normal operation to the capabilities of each of the two control systems. Sufficient margin must exist to allow for the uncertainties in the rate.

5. The review procedures in the area of control rod patterns and reactivity worths are:

   a. The reviewer determines by inspection and study of the information described in Section I.5 of this plan that the control rod and bank worths are reasonable. This determination involves evaluation of the appropriateness of the analytical models used, the applicability of experimental data used to validate the models, and the applicability of generic positions or those established in previous reviews of similar reactors.

b. The reviewer determines the equipment, operating restrictions, and administrative procedures that are required to restrict possible control rod and bank worths, and the extent to which the alarm criterion in II.2.c.(2) is satisfied. If the equipment involved is subject to frequent downtime, the reviewer must determine if alternative measures should be provided or the extent of proposed outage time is acceptable.

c. The reviewer will employ the same procedures as in a, above, to evaluate the scram reactivity information described in I.5. The scram reactivity is a property of the reactor design and is not easily changed, but if restrictions are necessary the procedures in b, above, can be followed as applicable.

d. The reviewer or staff consultants may perform check calculations in this area as necessary to complete the review.

6. The information presented on criticality of fuel assemblies is reviewed in the context of the applicant's physics calculations and the ability to calculate criticality of a small number of fuel assemblies. This information is related to information on fuel storage presented in SAR Section 9.1 and reviewed by the Auxiliary and Power Conversion Systems Branch (APCSB). The APCSB reviewer assumes that the applicant's criticality calculations have been reviewed by CPB and are acceptable. Independent criticality audit calculations may be done by the reviewer or staff consultants as necessary to complete the review.

7. The reviewer exercises professional judgment and experience to ascertain the following about the applicant's analytical methods:

a. The computer codes used in the nuclear design are described in sufficient detail to enable the reviewer to establish that the theoretical bases, assumptions, and numerical approximations for a given code reflect the current state of the art.

b. The source of the neutron cross sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm that the cross sections are comparable to those in the current ENDF/B data files (Ref. 6). If modifications and normalization of the cross section data have been made, the bases used must be determined to be acceptable.

c. The procedures used to generate problem-dependent cross section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the U-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, Pu-240); calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculations including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies; and calculations of fuel and burnable poison depletion and fission product buildup.

d. The gross spatial flux calculations that are used in the nuclear design are discussed in sufficient detail so that the reviewer can confirm that the following items are adequate to produce results of acceptable accuracy; the method of calculation (e.g., diffusion theory, S_n transport theory, Monte Carlo, synthesis); the number of energy groups used; the number of spatial dimensions (1, 2, or 3) used; the number of spatial mesh intervals, when applicable; and the type of boundary conditions used, when applicable.

e. The calculation of power oscillations and stability indices for diametral xenon reactivity transients, axial xenon reactivity transients, other possible xenon reactivity transients, and non-xenon-induced reactivity transients, are discussed in sufficient detail so that the reviewer can confirm for each item that the method of calculation (e.g., modal analysis, diffusion theory, transport theory, synthesis) and the number of spatial dimensions used (1, 2, or 3) are acceptable.

f. Verification of the data base, computer codes, and analysis procedures has been made by comparing calculated results with measurements obtained from critical experiments and operating reactors. The reviewer ascertains that the comparisons cover an adequate range for each item and that the conclusions of the applicant are reasonable.

8. The analysis of neutron irradiation of the reactor vessel may be used in two ways. It may provide the design basis for establishing the vessel material nil-ductility transition temperature as a function of the fluence, nvt. Or, it may provide the relative flux spectra at various positions between the pressure vessel and the reactor core so that the flux spectrum for various test specimens may be estimated. This information is used by the Materials Engineering Branch in determining the reactor vessel material surveillance program requirements and pressure-temperature limits for operation. CPB reviews the calculational method, the geometric modeling, and the uncertainties in the calculations under this plan. The review procedures for pressure vessel irradiation include determinations that:

a. The calculations were performed by higher order theory than diffusion theory.

b. The geometric modeling is detailed enough to properly estimate the relative neutron spectrum at various positions from the reactor core boundary to the pressure vessel wall.

c. The peak vessel wall fluence for the design life of the plant is less than 10^{20} n/cm^2 for neutrons of energy greater than one MeV. If the peak fluence is found to be greater than this value, the Materials Engineering Branch is notified.

IV. EVALUATION FINDING

The reviewer verifies that sufficient information has been provided and his review supports the following type of evaluation finding, which is to be included in the staff's safety evaluation report:

"The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the ________________ plant.

"To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a _____%k/k subcritical margin in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position.

"On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design;" Criterion 11, "Reactor Inherent Protection;" Criterion 12, "Suppression of Reactor Power Oscillations;" Criterion 13, "Instrumentation and Control;" Criterion 20, "Protection System Functions;" Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions;" Criterion 26, "Reactivity Control System Redundancy and Capability;" Criterion 27, "Combined Reactivity Control Systems Capability;" and Criterion 28, "Reactivity Limits."

2. Staff safety evaluation reports and plant technical specifications. Examples of these are:

   a. Safety Evaluation Report, General Electric Standard Safety Analysis Report (GESSAR), Section 4.3, Docket No. STN 50-447, U. S. Atomic Energy Commission, November 1974.

b. Safety Evaluation Report, Combustion Engineering Standard Safety Analysis Report (CESSAR), Section 4.3, Docket No. STN 50-470, U. S. Nuclear Regulatory Commission, to be published.

c. Safety Evaluation Report, Jamesport Nuclear Power Station Units 1 and 2, Section 4.3, Docket No. STN 50-516/517, U. S. Nuclear Regulatory Commission, to be published.

d. Safety Evaluation Report, Greenwood Energy Center Units 2 and 3, Section 4.3, Docket Nos. 50-452/453, U. S. Atomic Energy Commission, July 17, 1974.

e. Technical Specifications, Browns Ferry Nuclear Plant Unit 1 and Unit 2, Sections 2.1 and 3.2 through 3.5, License No. DPR-33 and 52, June 28, 1974.

f. Technical Specifications, Millstone Point Nuclear Power Station Unit No. 2, Sections 2.1, 3.1, and 3.2, Docket No. 50-336, to be published.

g. Technical Specifications, D. C. Cook Nuclear Plant Unit 1, Sections 2.1 and 3.1 through 3.3, License No. DPR-58, October 25, 1974.

h. Technical Specifications, Arkansas Nuclear One Unit 1, Sections 2.1, 3.1, and 3.5, License No. DPR-51, May 21, 1974.

3. Staff technical reports on fuel densification:

a. Regulatory Staff, "Technical Report on Densification of Light Water Reactor Fuels," U. S. Atomic Energy Commission, November 14, 1972.

b. Regulatory Staff, "Technical Report on Densification of Babcock and Wilcox Reactor Fuels," U. S. Atomic Energy Commission, July 6, 1973.

c. Regulatory Staff, "Technical Report on Densification of Exxon Nuclear BWR Fuels," U. S. Atomic Energy Commission, September 3, 1973.

d. Regulatory Staff, "Technical Report on Densification of Gulf United Nuclear Fuels Corporation Fuels for Light Water Reactors," U. S. Atomic Energy Commission, November 21, 1973.

e. Regulatory Staff, "Technical Report on Densification of Westinghouse PWR Fuel," U. S. Atomic Energy Commission, May 14, 1974.

4. Topical and startup test reports which are current and applicable to the reactor under review. Examples of these are:

a. G. N. Kear and M. J. Ruderman, "An analysis of Methods in Control Rod Theory and Comparison with Experiment," GEAP-3937, General Electric Company, May 1962.

b. J. S. Moore, "Power Distribution Control of Westinghouse PWR's," WCAP-7811, Westinghouse Electric Corporation, December 1971.

c. J. O. Cermak, et al, "Pressurized Water Reactors pH-Reactivity," WCAP-3696-8, Westinghouse Electric Corporation, October 1968.

d. "Surry Power Station - Unit 2, Startup Test Report," Virginia Electric Power Company, July 31, 1973.

e. J. E. Outz, "Plant Startup Test Report, H. B. Robinson Unit No. 2," WCAP-7844, Westinghouse Electric Corporation, January 1972.

f. R. H. Clark and T. G. Pitts, "Physics Verification Experiments, Core I," BAW-TM-455, Babcock and Wilcox Company, June 1966.

g. R. H. Clark, "Physics Verification Experiments, Cores II and III," BAW-TM-458, Babcock and Wilcox Company, July 1966.

h. D. R. Jones and J. G. Harsum, "Field Testing Requirements for Fuel Curtains and Control Rods," NEDO-10017, General Electric Company, June 1969.

i. R. Barry, et al, "Nuclear Design of Westinghouse PWR's with Burnable Poison Rods," WCAP-9000-L, Revision 1, Westinghouse Electric Corporation, June 1969.

j. G. V. Kumar, "Startup Test Results - Dresden NPS Unit 3," NEDC-10692, General Electric Company, December 1972.

k. E. J. Dean, "Quad Cities Units 1 and 2 - Startup Test Results," NEDC-10812, General Electric Company, March 1973.

l. J. D. LeBlanc, "Maine Yankee Atomic Power Station Startup Test Report," Maine Yankee Atomic Power Company, June 1973.

5. Brookhaven National Laboratory interim report files maintained by Core Performance Branch, Task 2, "Moderator Coefficients," and Task 3, "Control Rod Worths."

6. M. K. Drake, ed., "Data Formats and Procedures for the ENDF Neutron Cross Section Library," BNL-50274 (ENDF-102), National Neutron Cross Section Center, Brookhaven National Laboratory (1970).

7. Branch Technical Position CPB 4.3-1, "Westinghouse Constant Axial Offset Control," July 1975, attached to Standard Review Plan 4.3.

BRANCH TECHNICAL POSITION CPB 4.3-1
WESTINGHOUSE CONSTANT AXIAL OFFSET CONTROL (CAOC)

A. BACKGROUND

In connection with the staff review of WCAP-8185 (17x17), we reviewed and accepted a scheme developed by Westinghouse for operating reactors in such a fashion that throughout the core cycle including during the most limiting power maneuvers the total peaking factor, F_Q, will not exceed the value consistent with the LOCA or other limiting accident analysis. This operating scheme, called constant axial offset control (CAOC), involves maintaining the axial flux difference within a narrow tolerance band around a burnup-dependent target in an attempt to minimize the variation of the axial distribution of xenon during plant maneuvers.

Originally (early '74), the maximum allowable F_Q (for LOCA) was 2.5 or greater. Later (late '74), when needed changes were made to the ECCS evaluation model, Westinghouse, in order to meet physics analysis commitments to all its customers at virtually the same time, did a generic analysis (one designed to suit a spectrum of operating and soon-to-be-operating reactors) and showed that most plants could meet the requirements of Appendix K and CFR 50.46 (i.e., 2200^{o}F peak clad temperature) if $F_Q \leq 2.32$. Also, Westinghouse showed that CAOC procedures employing a $\pm$ 5% target band would limit peak F_Q for each of these reactors to less than 2.32.

We recognized at that time, however, that not all plants needed to maintain F_Q below 2.32 to meet FAC, or, needed to operate within a $\pm$ 5% band to achieve $F_Q \leq 2.32$. In fact, Point Beach was allowed to operate with a wider band because the Wisconsin Electric Power Company demonstrated to our satisfaction that the reactors could be maneuvered within a wider band (+6,-9%) and still hold F_Q below 2.32. We fully expected that in time most plants would have individual CAOC analyses and procedures tailored to the requirements of their plant-specific ECCS analyses.

Therefore, when we accepted CAOC it was not just F_Q = 2.32 and a $\pm$ 5% band width we were approving, but the CAOC methodology. This is analogous to our review and approval of ECCS and fuel performance evaluation models.

The CAOC methodology, which is described in WCAP-8385 (Ref. 1), entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for the cycle which maximizes plant flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with out-of-core detectors.

Westinghouse argues that point 3, above, is achieved by calculating all of the load-follow maneuvers planned for the proposed cycle and showing that the maximum power densities expected are within limits. These calculations are performed with a radial/axial synthesis method which has been shown to predict conservative power densities when compared to experiment. While we have accepted CAOC on the basis of these analyses, we have also required that power distributions be measured throughout a number of representative (frequently, limiting) maneuvers early in cycle life to confirm that peaking factors are no greater than predicted. Additionally, we are sponsoring a series of calculations at BNL to check aspects of the Westinghouse analysis.

The power distribution measurement tests described above will, of course, automatically relate incore and excore detector responses, and thereby validate that power distribution control can be managed with excore detectors.

B. BRANCH TECHNICAL POSITION

Whenever an applicant or licensee proposes CAOC for other than $F_Q = 2.32$ and $\Delta I = \pm 5\%$ he is expected to provide:

1. Analyses of F_Q x power fraction showing the maximum $F_Q(z)$ at power levels up to 100% and DNB performance with allowed axial shapes relative to the design bases for overpower and loss of flow transients. The envelope of these analyses must be shown to be valid for all normal operating modes and anticipated reactor conditions. (See Table 1 of Reference 2 for the cases which must be analyzed to form such an envelope.)

2. A description of the codes used, how cross-sections for cycle were determined, and what F_{xy} values were used.

3. A commitment to perform load-follow tests wherein F_Q is determined by taking incore maps during the transient. (NOTE: Westinghouse has outlined for both the NRC staff and the ACRS an augmented startup test program designed to confirm experimentally the predicted power shapes. The details of this program will be disclosed in a soon-to-be-issued WCAP report. The tests will be carried out at several representative - both 15x15 and 17x17 - reactors. We have endorsed these tests as has the ACRS in its June 12, 1975 Diablo letter. In addition, for the near term, we plan to require that those licensees who propose to depart from the previously approved peaking factor and target band width perform similar tests, precisely which ones to be determined on a case-by-case basis, to broaden our confidence in analytical methods by extending the comparison of prediction with measurement to include more and more burnup histories.)

C. REFERENCE

1. T. Morita et al., "Power Distribution Control and Load Following Procedures," WCAP-8403, Westinghouse Electric Corporation, September 1974.

2. C. Eicheldinger, Westinghouse Electric Corporation, letter to D. B. Vassallo, U.S. Nuclear Regulatory Commission, July 16, 1975.

11/24/75

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The objectives of the review are to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods; is equivalent to or is a justified extrapolation from proven designs; provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational transients; and is not susceptible to thermal-hydraulic instability. This plan describes the normal review of thermal and hydraulic design, i.e., that for a plant similar in core and primary coolant system design to previously reviewed plants. The review of new prototype plants or new design methods require in addition the independent audit analyses discussed in the appendix to this plan.

The review includes evaluation of the proposed technical specifications regarding safety limits and limiting safety system settings, to ascertain that these are consistent with the power-flow operating map for boiling water reactor (BWR) plants or the temperature-power operating map for pressurized water reactor (PWR) plants.

The review also includes determination of the largest hydraulic loads on core and reactor coolant system components during normal operation and postulated accident conditions. This information is provided to the Mechanical Engineering Branch for use in the review of reactor components and structures.

To accomplish the objectives, the reviewer examines features of core and RCS components, key process variables for the coolant system, calculated parameters characterizing thermal performance, data serving to support new correlations or changes in accepted correlations, and assumptions in the equations and solution techniques used in the analyses. The reviewer determines that the applicant has used approved analysis methods in the manner specified by topical reports describing the methods and by staff reports approving the methods. If an applicant has used previously unapproved correlations or analysis methods, the reviewer initiates a generic evaluation.

The CPB, as described in Standard Review Plan (SRP) 4.3, provides technical consultation on matters related to core physics calculations and their integration with power distribution assumptions made for the core thermal and hydraulic analysis. The CPB, on request, participates in generic evaluation of new thermal and hydraulic analysis methods.

II. ACCEPTANCE CRITERIA

1. General Design Criterion 10 (Ref. 1) requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   There are two acceptable approaches to meet this criterion:

   a. For departure from nucleate boiling ratio (DNBR) or critical heat flux ratio (CHFR) correlations there should be a 95% probability, at the 95% confidence level, that no fuel rod in the core experiences a departure from nucleate boiling condition during normal operation or transients that are anticipated to occur with moderate frequency.

   b. For critical power ratio (CPR) correlations, the limiting (minimum) value of CPR is to be established such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core wide transients. For transients that effect only a portion of the core, the same value of CPR is used to provide additional conservatism.

   Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions. As guidance to the reviewer, the correlations listed below have been found acceptable for previously reviewed plants.

   a. BWR's - The minimum CHFR calculated with the Hench - Levy correlation (Ref. 8) should exceed 1.0 at all times. The value of the minimum CPR calculated with the GETAB analysis will vary for different plants and/or product lines. Typically, the value will exceed 1.06.

   b. PWR's - For 14 x 14 or 15 x 15 rod arrays the minimum DNBR calculated with due allowance for mixing grids (Refs. 3, 4, and 5) should exceed 1.32 using the BAW-2 correlation (Ref. 6) and 1.30 using the W-3 correlation (Ref. 7).

2. As problems affecting DNBR or CPR limits arise, such as fuel densification or rod bowing, experimental and/or analytical methods for determining the appropriate design penalties are included in the review. Subchannel hydraulic analysis codes such as those described in References 8, 9, and 10 should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments

or power reactor cores. Calculations of BWR fluid conditions for use in CHF correlations have been in accordance with the models specified in Reference 11.

3. The maximum value of the linear heat generation rate (LHGR) anywhere in the core, including all hot spot and hot channel factors, should be such that the centerline temperature of the fuel is below the melting temperature. For most core designs, full power steady-state operation is not the operating mode which is most limiting in regard to LHGR. Rather, ECCS performance following a postulated loss-of-coolant accident or various anticipated transients is more limiting. As guidance for the reviewer, the following values of LHGR have been found acceptable for previously reviewed designs:

LHGR (kW/ft)

BWR	PWR
17.5 kW/ft for 1965 product line	18.5 kW/ft for 15 x 15 array
18.5 kW/ft for 1967 product line	18.5 kW/ft for 14 x 14 array
18.5 kW/ft for 1969 product line	13.0 kW/ft for 17 x 17 array
13.4 kW/ft for BWR-6	13.0 kW/ft for 16 x 16 array

While these values do not constitute criteria for acceptance, any design in which they are exceeded must be supported by sufficient analysis to demonstrate that all acceptance criteria are met. Other operating conditions such as fuel densification may reduce these values.

4. For PWR and BWR fuel, the maximum clad strain calculated for operational transients and at end-of-life should be less than one percent. These analyses should consider the pressure associated with gaseous fission products.

5. The reactor should be demonstrated to be free of undamped oscillations or other hydraulic instabilities for all conditions of steady-state operation, for all operational transients, for all load-following maneuvers, and for partial loop operation. Typical methodologies are described in References 12 and 13.

6. Methods for calculating single-phase and two-phase fluid flow in the RCS piping and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as the following, these relationships should be confirmed empirically:

   a. Steam generator (Ref. 14).

   b. Reactor vessel (Ref. 15).

   c. Jet pump (Ref. 16).

   d. Core flow distribution (Refs. 15, 17, and 18).

7. The proposed technical specifications should be established such that the plant can be safely operated at steady state conditions under all of the expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, such that acceptance criterion 1, above, is satisfied. The limiting conditions of operation must provide appropriate operating restrictions. For example, the limiting conditions of operation must assure that the reactor coolant pumps have adequate net positive suction head for all expected modes of operation.

8. Any changes to accepted codes, correlations, and analytical procedures, or the addition of new ones must be justified on theoretical or empirical grounds.

9. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68 (Ref. 19), as regards measurements, and confirmation of thermal hydraulic design aspects.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this review plan. For operating license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to safety limits, limiting safety system settings, and conditions of operation.

The reviewer must begin with an understanding of currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods; from standard texts and other technical literature which establish the methodology and the nomenclature of this technology; and from documents which summarize current staff positions concerning acceptable design methods.

Much of the review described below is generic in nature and is not performed for each plant. That is, the RSB reviewer is to compare the core design and operating parameters to those of previously reviewed plants. He then devotes the major portion of his review effort to those areas where the application is not identical to previously reviewed plants.

The reviewer is to compare the information in the applicant's safety analysis report (SAR) to the documents referenced by the applicant or in this plan to determine conformance to the bounds established by such documents. The reviewer must confirm that void, pressure drop, and heat transfer correlations used to estimate fluid conditions (flow, pressure, quality) are within the ranges of applicability specified by their authors or in previous staff reviews, that the analysis methods are used in the manner specified by the developers or in previous staff reviews, that the reactor design falls within the ranges of applicability specified for accepted analysis methods, and that the design is within the criteria specified in II, above, and is not an unexplained or unwarranted extrapolation of other thermal-hydraulic designs.

The review does not routinely involve calculations by the staff. On occasion, e.g., if a new model or correlation is proposed, independent analyses are performed by the staff or by consultants under the direction of the RSB. These analyses establish the range of applicability and associated accuracy of the new model or correlation and the reviewer ensures it is applied accordingly.

The reviewer is to establish that the thermal-hydraulic design and its characterization by MCHFR or DNBR have been accomplished and are presented in a manner which accounts for all possible reactor operating states as determined from operating maps. In this regard, the reviewer must confirm, with the aid of the CPB, that the power distribution assumptions of SAR Section 4.4 are a conservative (i.e. worst-case) accounting of the power distributions derived in Section 4.3 from core physics analyses, and that the latter analyses include an acceptable calculation of local void fractions. He must also confirm that the mass flux used in these calculations takes into account the core flow distribution (including that for partial loop operation) and the worst case of core bypass flow. The reviewer confirms that the primary coolant flow range shown in the operating map will be verified by pre-startup measurements.

The reviewer ensures that adequate account is taken of the effect of crud in the primary coolant system, such as in the calculation of CHF in the core, heat transfer in the steam generators, and pressure drop throughout the RCS.

The reviewer is to examine the calculation of hydraulic loads for normal operations and postulated accidents, ensure they are accurately estimated for the worst cases, and supply the worst case values to the Mechanical Engineering Branch for their review of reactor components and supports.

The reviewer should be aware of the vibration and loose-parts monitoring equipment and procedures used on other comparable plants and, taking into account pertinent differences, ensure that an adequate system is provided for the plant under review.

The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 19). At the OL stage, the reviewer is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria. (See par. C.2.b of Reference 19.)

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the core and reactor coolant system will satisfy functional requirements. As an alternative to this detailed evaluation, the reviewer may compare the core and reactor coolant system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs are adequate for the core and reactor coolant system.

If the core or the reactor coolant system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

The proposed technical specifications that relate to the core and the reactor coolant system are evaluated. This evaluation is to cover all of the safety limits and bases that could affect the thermal and hydraulic performance of the core. The limiting safety system settings are reviewed to ascertain that acceptable margins exist between the values at which reactor trip occurs automatically for each parameter (or combinations of parameters) and the safety limits. The reviewer confirms that the limiting safety system settings and limiting conditions for operation, as they relate to the reactor coolant system, do not permit operation with any expected combination of parameters that would not satisfy criterion 1 of Section II.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> "The thermal-hydraulic design of the core for the ________ plant was reviewed. The scope of review included the design criteria, preliminary core design, and the steady-state analysis of the core thermal-hydraulic performance.* The review concentrated on the differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff. It was found that all such differences were satisfactorily justified by the applicant. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.
>
> "The staff concludes that the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

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*For an OL review this sentence should be modified to include the implementation of the design criteria as represented by the final core design.

3. F. F. Cadek, F. E. Motley, and D. P. Dominicis, "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-L (proprietary), Westinghouse Electric Corporation, June 1972.

4. F. E. Motley and F. F. Cadek, "DNB Test Results for New Mixing Vane Grids (R)," WCAP-7695-L (proprietary), Westinghouse Electric Corporation, July 1972.

5. F. E. Motley and F. F. Cadek, "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB," WCAP-7988, Westinghouse Electric Corporation, October 1972. (See also WCAP-8030.)

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APPENDIX

STANDARD REVIEW PLAN 4.4

INDEPENDENT AUDIT ANALYSIS

The Core Performance Branch may be requested to perform independent analyses of thermal-hydraulic and physics phenomena for both steady-state and transient conditions. These analyses may be requested by various technical groups within the staff.

The required analyses may be in the following forms:

1. Independent computer calculations to substantiate reactor vendor analyses of steady-state or transient events.

2. Evaluations of vendor computer programs and analysis methods.

3. Reductions and correlations of experimental data to verify processes or phenomena which are applied to reactor design. These independent audit analyses may also be undertaken in support of Standard Review Plans (SRP) 4.2 and 4.3, in addition to the independent analysis discussed in SRP 4.4.

TYPES OF ANALYSES

The types of analyses that are performed are the following:

1. <u>Steady-State Analyses</u>

    a. The steady-state reactor core flow distribution, steam quality, void distribution, and pressure drop have been calculated for PWR-type fuel assemblies using the multichannel boiling code, COBRA III-C (Ref. 1). From these quantities COBRA III-C also calculates the fuel thermal margin in terms of the ratio of the local predicted critical heat flux to the operating heat flux. The W-3 and B&W-2 critical heat flux correlations (Refs. 2 and 3) are used in the code. From these results, the thermal margin and fuel clad temperature calculated by the vendor's computer program can be verified. To the extent possible, inputs to computer programs used by the staff correspond to those used by the reactor vendors.

    b. Through the use of consultants, independent comparisons and correlations are made of data from experimental programs. These reviews also include analyses of experimental techniques, test repeatability, and data reduction methods.

2. <u>Transient Analyses</u>

    Independent computer calculations are performed to provide an audit on the adequacy of a particular analysis performed by an applicant. The thermal-hydraulic phenomena associated with the transient are calculated with the RELAP-3 (Ref. 4) or RELAP-4 (Ref. 5) computer programs. The fuel performance is calculated by the COBRA III-C program, which obtains the necessary thermal-hydraulic parameters from the above programs.

3. <u>Loss-of-Coolant Accident (LOCA) Analyses</u>

Independent calculations are performed by the staff to verify the LOCA analyses submitted by applicants in accordance with the requirements of 10 CFR §50.46 and Appendix K of 10 CFR Part 50. These calculations are performed to check the blowdown phenomena, ECCS response, and fuel cladding temperature transients. The ECCS performance criteria are specified in Appendix K. Also, sensitivity studies are performed to verify the convergence of analytical techniques, and the sensitivity to various postulated break sizes, types, and locations.

Evaluations are also made of the computer programs used by the vendors to perform ECCS evaluations. These computer programs are checked to determine conformance with the required features specified in Appendix K. In addition, the analysis methods and heat transfer correlations are evaluated by comparison with existing experimental data.

4. <u>Reactivity Analyses</u>

Independent analyses are performed by consultants to provide a check on the adequacy of a particular analytical method and the basic assumptions. These include items such as maximum control rod worth, power distribution, and reactivity coefficients such as the Doppler and moderator temperature coefficients.

Staff consultants assess the conservatism of the vendors' models, either by comparison with experiment or with more sophisticated models. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes are investigated and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant are investigated.

5. <u>ATWS (Anticipated Transients Without Scram) Analyses</u>

Independent audit analysis for ATWS serves three specific functions:

a. To confirm the vendors' and applicants' interpretations of WASH-1270 (Ref. 6).

b. To evaluate the adequacy of backup protection systems.

c. To confirm the accuracy of the calculation of consequences of ATWS and of the models used in the analysis.

The RELAP 3B computer code (Ref. 7) will be used by the staff and its consultants for the ATWS studies. The preparation of data for an independent audit computation requires a careful review of all reactor systems to ascertain if operational credit can be taken for them in the analysis. The process will then serve as a means of confirming the vendors' and applicants' interpretations of WASH-1270.

An evaluation of the effectiveness and the response of a backup protection system is achieved by an audit computation. The degree of protection can be evaluated by conducting analyses with and without the backup system.

A calculation of the consequences of an ATWS transient serves as a means of evaluating vendors' analytical models and the accuracy of the results. For example, an ATWS loss of load event for a PWR can result in very high primary pressure. The magnitude of the pressure response is a function of the performance of the pressurizer safety valves in discharging water. An independent audit computation would verify the analytical model used for discharging water and the magnitude of the pressure response. The pressure response is important in evaluating the integrity of the reactor vessel.

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.5.1 CONTROL ROD SYSTEM STRUCTURAL MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - Core Performance Branch (CPB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

General Design Criterion 26 requires that one of the reactivity control systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences. The areas listed below relating to materials considerations in the design of the control rod system are reviewed. The review areas are similar to those given in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials." For the purpose of this plan, the control rod system extends only to the coupling interface with the reactivity control (poison) elements in the reactor vessel.

The mechanical aspects of the control rod system other than the reactivity control elements are reviewed by the Mechanical Engineering Branch in accordance with Standard Review Plan 3.9.4.

The mechanical design, thermal performance and chemical compatibility of the reactivity control elements are addressed by the Core Performance Branch in accordance with Standard Review Plan 4.2.

1. Mechanical Properties
The mechanical properties of the materials used in the control rod system are reviewed from the standpoint of adequate performance throughout the design life of the plant (or the component). The systems generally include control rods and control rod drives. Materials commonly used include austenitic stainless steels (which may be cold worked), nitrided or chromium-plated stainless steels, martensitic stainless steels, precipitation-hardening stainless steels such as 17-4 pH, and other special-purpose materials such as cobalt-base alloys (stellites), Inconel-750, Colmonoy-6, and Graphitar-14.

2. <u>Austenitic Stainless Steel Components</u>

The use of sensitized stainless steels should be controlled to prevent stress-corrosion cracking of the material during operation of the plant. Welding procedures should be controlled to reduce the probability of sensitization and microfissure formation. Cold-worked stainless steels should not have too high a yield stress, to minimize the probability of stress-corrosion cracking during operation of the plant.

3. <u>Other Materials</u>

Special requirements for the other materials include minimum tempering and aging temperatures for martensitic and precipitation-hardening stainless steels to prevent their deterioration by stress corrosion during operation of the plant. The compatibility of these materials with the reactor coolant is reviewed to assure that they will continue to perform satisfactorily throughout the design life of the component.

4. <u>Cleaning and Cleanliness Control</u>

Proper care should be taken in handling the materials and parts of the control rod system during fabrication, shipping, and on-site storage to assure that all cleaning solutions, processing compounds, degreasing agents, and other foreign materials are completely removed, and that all parts are dried and properly protected following any flushing treatment with water.

II. <u>ACCEPTANCE CRITERIA</u>

The acceptance criteria for the areas of review listed in Section I of this plan are as follows:

1. <u>Mechanical Properties</u>

The mechanical properties of the materials selected for the control rod system must be equivalent to those given in Appendix I to Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), or Part A of Section II of the Code, except that cold-worked austenitic stainless steels shall have a 0.2% offset yield strength no greater than 90,000 psi, to minimize the probability of stress corrosion cracking occurring in these systems.

2. <u>Austenitic Stainless Steel Components</u>

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptable methods for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding. Nitrided stainless steel components may be in the sensitized condition, as indicated in Regulatory Guide 1.44. Branch Technical Position - MTEB No. 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," (Ref. 9) describes acceptable criteria for assuring the integrity of welds in stainless steel components of these systems.

3. Other Materials

All materials for use in this system must be selected for their compatibility with the reactor coolant, as described in Articles NB-2160 and NB-3120 of the Code. The minimum tempering temperature of martensitic and aging temperature of precipitation-hardening stainless steels should be specified to provide assurance that these materials will not deteriorate by stress-corrosion cracking in service. Acceptable minimum treatment temperatures include aging at 1100°F for Type 17-4 PH and 1050°F for Type 410 stainless.

4. Cleaning and Cleanliness Control

Onsite cleaning and cleanliness control should be in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants" (Ref. 5).

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case. To ascertain that the acceptance criteria given in Section II are met, the reviewer examines the review areas listed in Section I for the required information, using the following procedures:

1. Mechanical Properties

The reviewer compares the mechanical properties of the materials proposed for the control rod system with Appendix I to Section III of the Code, or Part A of Section II of the Code. He verifies that cold-worked austenitic stainless steels used in fabrication of the reactivity control systems are in conformance with Section II.1, above.

2. Austenitic Stainless Steel Components

The methods of controlling sensitized stainless steel are examined by the reviewer and compared with the positions given in Regulatory Guide 1.44, especially with respect to cleaning and protection from contamination during handling and storage, verification of non-sensitization of the material, and qualification of welding procedures using ASTM A-262-70 (Ref. 3). If alternative methods of testing the qualification welds for degree of sensitization are proposed by the applicant, the reviewer determines if these are satisfactory, taking into account branch positions taken on previous applications and the degree of equivalence of the alternate methods. The reviewer may ask the applicant to justify the technical basis for any departures from the cited positions. Alternative tests that have been accepted by the branch include the use of ASTM A-262-70 as amended by Westinghouse Process Specification 84201 MW (Ref. 6), for qualifying welds and testing raw materials for nonsensitization, and the use of ASTM A-393 specifications (Ref. 4) for testing the qualification welds for degree of sensitization.

The methods of controlling and measuring the amount of delta ferrite in stainless steel weld deposits are examined by the reviewer and compared to the positions in Regulatory Guide 1.31, "Control of Stainless Steel Welding," especially with respect to the filler metal acceptance procedures for delta ferrite content and the examination of production welds. If alternative positions are proposed by the applicant, the reviewer determines if these are satisfactory, taking into account branch positions taken on previous applications. The reviewer may ask the applicant to justify the technical basis for any departures from the acceptance criteria stated in Section II.2.

3. Other Materials
The reviewer examines the information provided in the applicant's safety analysis report (SAR) on the compatibility of the materials (other than austenitic stainless steels) to be used in contact with the reactor coolant. He determines that the materials are compatible with the service environment so that corrosion or stress corrosion of the component will not occur during the lifetime of the component.

The reviewer determines that minimum tempering temperatures of all martensitic stainless steels and minimum aging temperatures of precipitation-hardening stainless steels have been specified, and are in accordance with the acceptance criteria stated in Section II.3.

4. Cleaning and Cleanliness Control
The reviewer verifies that onsite cleaning and cleanliness control procedures are satisfactory and in accordance with Section II.4.

5. General
If the information contained in the SAR or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information, but also, the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

When the reviewer has verified that sufficient and acceptable information has been provided in accordance with the requirements of this review plan, conclusions of the following type are prepared, to be included in the staff's safety evaluation report:

"The mechanical properties of materials selected for the control rod system components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code, or Part A of Section II of the Code, and with the staff position that the yield strength of cold-worked austenitic stainless steel should not exceed 90,000 psi.

"The controls imposed upon the austenitic stainless steel of the systems conform to the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding" and Regulatory Guide 1.44, "Control of the use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these recommendations provide added assurance that stress-corrosion cracking will not occur during the design life of the component. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria for Articles NB-2160 and NB-3120 of Section III of the Code. Both martensitic and precipitation-hardening stainless steels have been given tempering or aging treatments in accordance with staff positions. Cleaning and cleanliness control are in accordance with ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plant."

"Conformance with the codes, standards, and Regulatory Guides indicated above, and with the staff positions on the allowable maximum yield strength of cold-worked austenitic stainless steel, the minimum tempering or aging temperatures of martensitic and precipitation-hardened stainless steels constitutes an acceptable basis for meeting in part the requirements of General Design Criterion 26."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

2. ASME Boiler and Pressure Vessel Code, Section III, Articles NB-2160 and NB-3120, and Appendix I, and Section II, Part A, American Society of Mechanical Engineers.

3. ASTM A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Intergranular Attack in Austenitic Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

4. ASTM A-393-63, "Recommended Practice for Conducting Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

5. ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," Draft 2, Revision 0, November 15, 1973, American National Standards Institute.

6. Process Specification 84201 MW, "Corrosion Testing of Wrought Austenitic Stainless Steel," Westinghouse Electric Corporation.

7. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

8. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

9. Branch Technical Position - MTEB No. 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended to Standard Review Plan 5.2.3.

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SECTION 4.5.2 REACTOR INTERNALS MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1 and 14 require that structures, systems, and components important to safety shall be designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The following areas in the applicant's safety analysis report (SAR) relating to reactor internals materials are reviewed:

1. Material Specifications

The review includes the material specifications for austenitic stainless steels, including weld materials, to be used for major components of the reactor internals and core support structures. These specifications should include, for boiling water reactors (BWR's), materials for shrouds, shroud supports, top guides, fuel support pieces, control rod drive tubes, jet pump assemblies, shroud head and steam separator assembly, and steam dryers; or for pressurized water reactors (PWR's), materials for the lower core support structures, including the core barrel, neutron shield pad assembly, core baffle, lower core plate, and core supports, the upper core support structures including the top support plate, beam sections, upper core plate, support columns, and guide tube assemblies, and the in-core instrumentation support structure.

The adequacy and suitability of the materials specified for the above applications are reviewed in terms of their mechanical properties, stress-corrosion resistance, and fabricability.

2. Controls on Welding

The review includes the controls on welding of materials used for reactor internals.

3. Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

The review includes the information submitted by the applicant on the nondestructive examinations used for inspection of the subject product forms.

4. Austenitic Stainless Steel

Quantities of austenitic stainless steels, in a variety of product forms, are used for construction of components in the reactor internals. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316 are normally used.

Since these compositions are susceptible to stress-corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material, and to minimize exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking. The review includes information submitted by the applicant in these areas, as described in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials."

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Material Specifications

Permitted material specifications are those shown in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, NG-2121, and Tables 1-1.1 and 1-1.2 of Appendix I. These materials are described in detail in the Code, Section II, Parts A, B, and C.

2. Controls on Welding

The welds of components for reactor internals, fabricated in accordance with the Code, Section III, NG-4400, must meet the acceptance criteria shown in NG-5000.

3. Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

The acceptance criteria for eddy-current and ultrasonic examination of wrought seamless tubular products and fittings are given in Regulatory Guide No. 1.66, "Nondestructive Examination of Tubular Products."

The acceptance criteria for radiographic examination of such products are given in the Code, Section III, NG-5320.

4. Austenitic Stainless Steels

The acceptance criteria for this area of review are given in item II.4 of Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials."

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For the areas of review described in Section I of this plan, the review procedure is as follows:

1. Material Specifications

   The list of the materials of construction of the components of the reactor internals that are exposed to the reactor coolant is reviewed.

   The material specifications for each major pressure-retaining component or part used in the reactor coolant boundary are compared with the acceptable specifications listed in the Code, Sections II and III, as shown in the acceptance criteria. Any exceptions to the Code materials specifications are clearly identified. The reviewer evaluates the basis for the exceptions, taking into account precedents set in earlier cases, and determines the acceptability of the proposed exceptions.

2. Controls on Welding

   The information submitted by the applicant is reviewed to provide assurance that welding of materials used for components of the reactor internals is in accordance with the procedures of the Code, Section III, NG-4400. The controls on welding of austenitic stainless steels, discussed in Standard Review Plan 5.2.3, are considered applicable to welding of reactor internals, and information in this area is verified. The reviewer assures that any special welding process or welding control conforms to the qualification requirements of the Code, Section IX, or that justification is made for any deviation.

3. Nondestructive Examination of Wrought Seamless Tubular Products

   The information submitted by the applicant is reviewed to determine methods used for nondestructive examination. The Code, Section III, NG-2551(d) specifies that examination by either radiographic or ultrasonic examination is acceptable.

   However, Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products" provides new calibration standards and procedures for ultrasonic examination which are considered more sensitive, and more consistently able to detect defects regardless of shapes or orientation. The reviewer verifies that ultrasonic examinations of the subject product form are specified to be in accordance with this Regulatory Guide.

4. Austenitic Stainless Steel

   The materials and fabrication procedures used for reactor internals are reviewed. The specific area of review and review procedures follow closely those spelled out in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials." Environmental conditions must be controlled and welding procedures must be such that the probability of sensitization and microfissuring is reduced. In addition, the reviewer verifies that the material and reactor coolant compositions have been selected to assure compatibility, and that the fabrication and cleaning controls imposed on stainless steel components will minimize contamination with chloride ~~and~~ fluoride ions.

5. Additional Information Request

If the information contained in the SAR does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, the reviewer prepares a request for additional information for transmittal to Reactor Projects. Such requests not only identify the additional information required, but also specify the changes needed in the SAR or the plant Technical Specifications to meet acceptance criteria. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of components of the reactor internals have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code.

"The materials for reactor internals exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion on all materials is expected to be negligible.

"The controls imposed on reactor coolant chemistry provide reasonable assurance that the reactor internals will be adequately protected during operation from conditions which could lead to stress corrosion of the materials and loss of component structural integrity.

"The controls imposed upon components constructed of austenitic stainless steel, as used in the reactor internals, satisfy the recommendations of Regulatory Guide No. 1.31, "Control of Stainless Steel Welding," Regulatory Guide No. 1.34, "Control of Electroslag Weld Properties," Regulatory Guide No. 1.44, "Control of the Use of Sensitized Stainless Steel," and Regulatory Guide No. 1.66, "Nondestructive Examination of Tubular Products." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendation provide reasonable assurance that the austenitic stainless steel used for reactor internals will be in a metallurgical condition which precludes susceptibility to stress-corrosion cracking during service. Conformance with these Regulatory Guides constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 14."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, and Section III, American Society of Mechanical Engineers.

3. ASTM A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

4. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants."

5. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

6. Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products."

7. Regulatory Guide 1.71, "Welder Qualification for Limited Accessibility Areas."

8. Standard Review Plan 5.2.3 and Branch Technical Position - MTEB No. 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended to Standard Review Plan 5.2.3.

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SECTION 4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The RSB reviews the combined functional performance of all the reactivity control systems to confirm that the systems can effect a safe shutdown, respond within acceptable limits during anticipated transients, and prevent or mitigate the consequences of postulated accidents.

The reactivity control systems whose functional performance is reviewed by the RSB include: control rod drive system (CRDS), chemical and volume control system (CVCS) for pressurized water reactors (PWR's), standby liquid control system (SLCS) for boiling water reactors (BWR's) and the recirculation flow control system (RFCS) for BWR's. Other aspects of each of these systems are evaluated by other reviewers as noted below.

The CPB in Standard Review Plan (SRP) 4.3 verifies the reactivity control requirements of the combined reactivity control systems. The negative reactivity available in the reactivity control systems, the allowable reactivity insertion or withdrawal rates, and the values of reactivity parameters throughout plant life are evaluated. Matters relating to steady-state core physics calculations and their integration with power distribution assumptions are considered in the CPB review.

The EICSB reviews in SRP 7.7 the control system for the RFCS. The intent of the EICSB review is to assure that failures of the control system would not impair the protection system capability in any significant manner. The EICSB also assists the RSB in reviewing the time delays for the actuation of each of the reactivity control systems. The EICSB in SRP 7.2 evaluates the results of failure modes and effects analyses to assure that a single failure occurring in the control system, or an operator error, will not result in the loss of capability for safe shutdown.

The APCSB, with the aid of the CPB reviewer, reviews the functional capability of the CVCS (for PWR's) and the SLCS (for BWR's) in SRP 9.3.4 and SRP 9.3.5, respectively, to determine the adequacy of each system to perform its function of reactivity control for the reactor.

The MEB reviews in SRP 3.9.4 the CRDS to evaluate the adequacy of the system to perform its mechanical function (e.g., rod insertion and withdrawal, scram operation and time) and to maintain the reactor coolant pressure boundary. The pressure-containing components of the CRDS are reviewed by the RSB in SRP 3.2.1 and SRP 3.2.2 to determine that design code requirements, as applicable to the assigned safety class and seismic category, are met.

II. ACCEPTANCE CRITERIA

Acceptability of the information presented in Section 4.6 of the applicant's safety analysis report (SAR), including related sections, is based on meeting the general design criteria (Ref. 1). The acceptance criteria for the areas of review are the following:

1. General Design Criterion 20, "Protection System Functions," as related to the automatic actuation of the reactivity control systems in accident conditions.

2. General Design Criterion 21, "Protection System Reliability and Testability," as related to system design requirements for high functional reliability and capability to meet the single failure criterion.

3. General Design Criterion 23, "Protection System Failure Modes," as related to failing into a safe state.

4. General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions," as related to the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded for malfunction of any reactivity control system.

5. General Design Criterion 26, "Reactivity Control System Redundancy and Capability," as related to the capability of the reactivity control system to regulate the rate of reactivity changes resulting from operational occurrences.

6. General Design Criterion 27, "Combined Reactivity Control Systems Capability," as related to the combined capability of reactivity control systems and emergency core cooling systems to cool the core under accident conditions.

7. General Design Criterion 28, "Reactivity Limits," as related to postulated reactivity accidents.

III. REVIEW PROCECURES

The RSB reviewer evaluates the capabilities of the combined operation of the reactivity control systems to effect reactor shutdown for all postulated operating conditions.

The review procedures set forth below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design as set forth in the applicant's preliminary safety analysis report (PSAR) meet the acceptance criteria given in Section II of this review plan. During the operating license (OL) review, the reviewer verifies that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report (FSAR).

1. The RSB reviews the CRDS design with respect to fluid systems and possible single failures. The review of the system description includes piping and instrumentation diagrams (P&IDs), layout drawings, process flow diagrams, and descriptive information on essential supporting systems. The SAR is reviewed to ascertain that failure modes and effects analyses have been completed to determine that the control rod drive system (not the individual drives) is capable of performing its safety-related function following the loss of any active component. The RSB reviewer further confirms, on the basis of previously approved systems or independent failure modes and effects analyses, that the minimum system requirements are met for the failure conditions.

2. The CRDS, P&IDs, layout drawings, and component description and characterictics are reviewed by the RSB to verify that essential portions of the system are correctly identified and are isolable from non-essential portions. The essential portions should be protected from the effects of high or moderate energy line breaks. Layout drawings of the system are reviewed to assure that no high or moderate energy piping systems are close to the CRDS, or that protection is provided from the effects of high or moderate energy pipe breaks.

3. For plants containing control rod drive cooling systems (e.g., using air or water as coolant), the description and drawings are reviewed to determine that the systems meet the design requirements. Essential equipment should be delineated in the SAR. The major function of the cooling system in PWR's is to cool the drive mechanism and remove heat from the CRDS motors to preclude motor burnout or damage. Failure of a CRDS motor could result in a rod drop. In BWR's, the major function of the cooling water is to cool the drive mechanism and its seals to preclude damage resulting from long-term exposure to reactor temperatures. The control rod drive hydraulic system includes the cooling function as part of its design. The RSB reviewer confirms by failure modes and effects analysis that the cooling system is capable of maintaining the CRDS temperature below the applicant's maximum temperature criterion. The EICSB reviewer in SRP 7.2 confirms that there are sufficient instrumentation and controls available to the reactor operator to provide information in the control room to monitor the CRDS conditions, including the more significant parameters such as coolant flow, temperature, and pressure and stator temperature.

4. In coordination with the MEB, the RSB reviews the functional tests of the CRDS as related to rod insertion and withdrawal and scram operation and time. The reviewers check the elements of the test program to ensure that all required thermal-hydraulic conditions have been included for all postulated operating conditions. Experimental

verification of system operation where a single failure has been assumed should be included in the test program, e.g., accumulator leakage for hydraulic CRDS and stuck rod operation.

5. The applicant's proposed preoperational and initial startup test programs are reviewed to determine if they provide reasonable assurance that the CRDS will perform its safety function. This aspect of the review is to verify that sufficient information is provided to identify the test objectives, methods of testing, and test acceptance criteria. If the design is essentially identical and if the proposed test programs are essentially the same as those of previously reviewed plants, the reviewer may conclude that the proposed test programs are adequate. If the proposed CRDS differs from that of prior designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are evaluated.

6. The plant technical specifications are reviewed by the RSB as follows:

   a. For CP's, the reviewer confirms the suitability of the limiting conditions of operation to ensure that the specified operating parameters (scram time, CRDS temperature, operation with inoperable rods) are within the bounds of the analyzed conditions.

   b. For OL's, the reviewer confirms that the content and intent of the technical specifications proposed by the applicant are in agreement with the requirements developed as a result of the staff's review. Where necessary, the review will include requirements for system functional testing, minimum performance, and surveillance requirements.

   c. The reviewer verifies by comparison with other plant reviews that the frequency and scope of periodic surveillance testing is adequate.

7. The reactivity control systems are evaluated to verify that redundant reactivity control systems are not vulnerable to common mode failures. The RSB identifies the common mode failures and the EICSB, MEB, and APCSB assist the RSB reviewer in connection with their responsibilities in SRP 7.4, 3.9.4, and 9.3.4 or 9.3.5, respectively. In addition, the reviewer determines that inadvertent operation of any component or system (e.g., inadvertent scram of axial power shaping rods or inadvertent dilution of boron concentration) would not cause degraded system conditions beyond the capabilities of the safety systems.

8. The RSB reviewer examines all transients and accidents in Chapter 15 of the SAR that require reactivity control systems to function. The RSB reviewer, with the CPB and EICSB reviewers, ascertains that the reactivity and response characteristics of the reactivity control system are conservative with respect to the parameters assumed in the Chapter 15 analyses. In the Chapter 15 review, the RSB reviewer verifies that no credit has been taken for the RFCS (in BWR's) to mitigate any accident. (Although the

RFCS controls reactor power level over a limited range, it is not required for shutdown.) In addition, the reviewer reviews the operation of the RFCS to confirm that a malfunction or failure of the system will not degrade the capabilities of plant safety systems or lead to plant conditions more severe than those considered in the accident analyses (e.g., by determining the effects of a failure of the system following a loss-of-coolant accident or steam line break). The RSB, in SRP 15.4.5, reviews the results of the most limiting transient from a malfunction of the RFCS.

IV. EVALUATIONS FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The functional designs of the reactivity control systems for the ______________ plant have been reviewed to confirm that the systems have the capability to shut down the reactor with appropriate margin during normal, abnormal and accident conditions. The reactivity control systems reviewed included the CRDS and _________ (CVCS for PWR's or SLCS and RFCS for BWR's). The scope of review included process flow diagrams, layout drawings, piping and instrumentation diagrams, and descriptive information for the systems and for the supporting systems that are essential for operation of the systems. [The applicant's proposed design criteria and design bases for the reactivity control systems and the adequacy of those criteria and bases have been reviewed. (CP)] [The manner in which the design of the reactivity control systems and supporting systems conform to the proposed design criteria and bases has been reviewed. (OL)]

"The basis for staff acceptance has been conformance of the applicant's designs, design criteria, and design bases for the reactivity control systems and their supporting systems to the Commission's regulations as set forth in the general design criteria of 10 CFR Part 50.

"The staff concludes that the designs of the reactivity control systems conform to all applicable regulations and are acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 5.2.1.1* COMPLIANCE WITH 10 CFR § 50.55a

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

In order to establish that pressure-retaining components of the reactor coolant pressure boundary of water-cooled nuclear power plants are in compliance with 10 CFR § 50.55a, an applicant is required to provide a table in his safety analysis report (SAR) identifying pressure vessels, piping, pumps and valves and the component code, code edition, applicable addenda, and component order date (where applicable) of each such component. Pressure-retaining components of the reactor coolant pressure boundary are designated as Class I components under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (hereafter the Code). 10 CFR § 50.55a requires that these components meet the requirements for Class I components under the Code.

For construction permit (CP) and operating license (OL) applications, the RSB will determine the acceptability of the information presented in the SAR, to assure that the applicant is in compliance with the rules of Section 50.55a.

In the event there are cases where conformance to Section 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, the applicant must provide a complete description of the circumstances and the basis for proposed alternate requirements. The applicant must describe how an equivalent and acceptable level of safety and quality will be provided by the proposed alternate requirements.

When required, the MEB and MTEB will provide assistance in establishing acceptability in the event an applicant invokes the "hardship" clause, and does not conform in all respects with Section 50.55a.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1. This criterion requires that structures, systems, and components important to safety shall be designed, fabricated

*Formally 5.2.1.3

erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR §50.55a. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components within the reactor coolant pressure boundary of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

3. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

III. REVIEW PROCEDURES

The table provided by the applicant identifying pressure vessel components, piping, pumps, and valves and the corresponding component code, code edition, applicable addenda, and when required, the component order date of each Code Section III, Class 1 component within the reactor coolant pressure boundary, is checked for compliance with Section 50.55a of 10 CFR Part 50. This review is applicable to CP and OL applications.

For those components within the reactor coolant pressure boundary not in compliance with the rules of Section 50.55a, a review of the code and code addenda is performed, to identify the specific sections with which the component does not comply. A decision to accept a component which is not fully in compliance with the rules is based on a judgement of the relative importance of the specific provisions in the code or code addenda not complied with, and a determination that any noncompliance will not result in an unacceptable level of safety and quality.

If the staff's concerns are not resolved in a satisfactory manner, a staff position is taken requiring conformance with the rules of Section 50.55a.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information is contained in the SAR and amendments and that his evaluation supports conclusions of the following type, which are to be included in the staff's safety evaluation report:

"The components of the reactor coolant pressure boundary, as defined by the rules of 10 CFR §50.55a, have been properly identified and classified as ASME Code Section III Class I components. These components will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR §50.55a.

"The staff concludes that construction of the components of the reactor coolant pressure boundary in conformance with these codes provides assurance that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. 10 CFR § 50.55a, "Codes and Standards Rule."

4. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1974).

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SECTION 5.2.1.2* APPLICABLE CODE CASES

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The RSB determines the acceptability of American Society of Mechanical Engineers (ASME) and American National Standards Institute (ANSI) code case interpretations specified in the safety analysis report (SAR). These code cases must be approved before being applied to ASME Boiler and Pressure Vessel Code, Section III, Class 1 pressure-retaining components within the reactor coolant pressure boundary, as stated in the Codes and Standards Rule, Section 50.55a(a)(2)(ii) of 10 CFR Part 50. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A.

The MEB and MTEB, on a generic basis, determine the acceptability of ASME and ANSI code case interpretations that may be applied to ASME Code Section III, Class 1 pressure-retaining components within the reactor coolant pressure boundary (Quality Group Classification A). These branches review each revision to applicable code cases. Code cases pertaining to materials, fabrication, and nondestructive testing are evaluated by the MTEB. All other areas covered by ASME code cases are evaluated by the MEB.

II. ACCEPTANCE CRITERIA

1. 10 CFR Part 50, Appendix A, General Design Criterion 1. This criterion requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR § 50.55a. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components within the reactor coolant pressure boundary of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

*Formally 5.2.1.4

3. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

4. Regulatory Guide 1.84, "Code Case Acceptability in ASME Section III - Design and Fabrication." This guide lists those Section III ASME code cases oriented to design and fabrication which are acceptable to the staff for implementation in the licensing of nuclear power plants.

5. Regulatory Guide 1.85, "Code Case Acceptability in ASME Section III - Materials." This guide lists those Section III ASME code cases oriented to materials and testing which are acceptable to the staff for implementation in the licensing of nuclear power plants.

III. REVIEW PROCEDURES

The table provided by the applicant identifying those ASME code cases applied to Class 1 pressure-retaining components within the reactor coolant pressure boundary is checked for compliance with the list of acceptable code cases identified in Regulatory Guides 1.84 and 1.85.

In the event an applicant should propose the use of a code case not previously reviewed by the staff, a review of the code case is requested of the MEB or MTEB, as appropriate.

IV. EVALUATION FINDINGS

The staff review should verify that only acceptable ASME and ANSI code cases are specified in the SAR in order to arrive at conclusions of the following type, which are to be included in the staff's safety evaluation report:

"The specified ASME and ANSI code cases whose requirements will be applied in the construction of pressure-retaining ASME Code Section III, Class 1 components within the reactor coolant pressure boundary (Quality Group Classification A), are acceptable to the staff. The staff concludes that compliance with the requirements of these code cases will result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. 10 CFR §50.55a, "Codes and Standards Rule"

4. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers (1974).

5. Regulatory Guide 1.84, "Code Case Acceptability in ASME Section III - Design and Fabrication."

6. Regulatory Guide 1.85, "Code Case Acceptability in ASME Section III - Materials."

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SECTION 5.2.2 OVERPRESSURIZATION PROTECTION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Overpressure protection for the reactor coolant pressure boundary (RCPB) is provided by means of relief and safety valves. For RCPB overpressure protection, the relief and safety valves operate in conjunction with the reactor protection system and the steam generator safety valves. The reviewer examines the design bases, system and component descriptions, and system analyses and tests described in the applicant's safety analysis report (SAR) in order to evaluate the adequacy of the overpressure protection which is provided. The areas of review for a boiling water reactor (BWR) are the reactor coolant system relief and safety valves. For a pressurized water reactor (PWR), the areas of review are the pressurizer safety and relief valves, and the piping from these valves to the quench tank. The review of anticipated transients without scram is described in Standard Review Plan (SRP) 15.8.

The adequacy of the proposed preoperational and initial startup test programs is examined as a part of this review. The reviewer also evaluates the proposed technical specifications to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The overpressure protection components are also reviewed to assure that they have the proper seismic and quality group classification. This aspect of the review is performed as a portion of the effort described in SRP 3.2.1 and SRP 3.2.2.

The MEB, as described in SRP 3.9.3, reviews the design and installation criteria for the overpressure protection components to assure that they are in conformance with ASME Boiler and Pressure Vessel Code requirements.

The EICSB, as described in SRP 7.6, evaluates the adequacy of controls and instrumentation of the overpressure protection components with regard to the required features of automatic actuation, remote sensing and indication, remote control, emergency onsite power, and connections to the reactor protection system.

The CPB provides generic evaluation of the mathematical models used to analyze transients that result in an increase in the pressure within the reactor coolant system.

The APCSB reviews the adequacy of the pressure relief and safety valves for the secondary system of PWR's.

II. ACCEPTANCE CRITERIA

The fundamental criterion against which an evaluation of overpressure protection is to be made is General Design Criterion 15 (Ref. 1): "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." Further, the preoperational and initial startup test programs are to meet the intent of Regulatory Guide 1.68 (Ref. 2).

To be acceptable, adequate relief and safety valve capacity must be provided for the primary systems of PWR's and BWR's. For PWR's, the secondary system must also be provided with relief and safety valves having adequate capacity.

1. Relief Valves

   For the design basis normal operational transients, the relief valve capacity must be sufficient to limit the pressure so as to prevent safety valve discharge directly to the containment, with the following assumptions:

   a. The reactor is operating at licensed core thermal power level.

   b. All system and core parameters are at the values within the normal operating ranges which would produce the highest transient pressure.

   c. All components, instrumentation, and controls function normally.

2. Safety Valves

   For the most severe abnormal operational transient, with reactor scram, the safety valve capacity should be sufficient to limit the pressure to less than 110% of the RCPB design pressure, as specified by the ASME Boiler and Pressure Vessel Code (Ref. 3), with sufficient margin to account for uncertainties in the design and operation of the plant and assuming:

   a. The reactor is operating at a power equal to the licensed core thermal power level plus an increment sufficient to account for power measurement uncertainties.

   b. All system and core parameters are at the values within the normal operating range, including uncertainties and technical specification limits, which would result in the highest transient pressure.

c. The reactor scram is initiated either by the high pressure signal or by the second signal from the reactor protection system, whichever is later.

d. The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code for each type of valve.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report and in the report on overpressure protection. The latter report is required by the ASME Code and is to serve as the basis for many of individual review steps outlined below during the OL review. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the RSB reviewer in determining that the acceptance criteria of Section II have been met. These steps should be applied to CP and OL reviews as appropriate. Previously reviewed designs may be used as a guide; however, the reviewer must verify that any changes are justified.

1. The piping and instrumentation diagrams are examined to determine the number, type, and location of the safety and relief valves in both the primary and secondary systems, and of discharge lines, instrumentation, and other components.

2. All other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems are identified. The effects of these other functions or systems on operation of the overpressure protection system are determined.

3. The valve descriptions are examined to determine type and manufacturer and to evaluate reliability (e.g., new or standard design).

4. The capacities, set points, and setpoint tolerances for all safety and relief valves are identified.

5. All of the reactor trip signals which occur during overpressure transients, including their setpoints and setpoint tolerances, are identified.

6. All transients analyzed in Chapter 15 of the SAR that result in an increase in the pressure experienced by the RCPB are examined. The peak predicted pressures are identified and the operating conditions and setpoints used in the analysis are reviewed to assure that they are suitably conservative.

The information below is provided to the reviewer as guidance and is based on typical previously-reviewed designs.

a. BWR's - For relief valve sizing, in transients in which a scram is initiated by closure of the main steam isolation valves or fast closure of the turbine stop valves, the highest pressure results from instantaneous loss of condenser vacuum or a turbine trip without bypass. For safety valve sizing, in transients in which scram is initiated by high flux or high pressure, the highest pressure results from closure of all main steam isolation valves. Analysis of previously acceptable designs has shown that the peak pressure is at least 25 psi below the allowable, assuming the reactor is operating at 105% of rated power, pressure is 1040 psia, no credit is taken for relief valve capacity, one safety valve fails to open, and scram is initiated by high pressure.

b. PWR's - For relief valve sizing, the valve capacity has been sufficient to accommodate the surge from the design basis step load change. Safety valve sizing is usually based on the maximum surge rate that results from a turbine trip without bypass. Analysis of previously acceptable designs has shown that the discharge flow from the safety valves in the primary and secondary systems is typically 86% and 100% of their respective rated capacities assuming the reactor is initially at 102% of rated power, the uncertainties in power, pressure, and temperature are 5%, 30 psi, and 4°F, respectively, scram is initiated by low level in the steam generator; and no credit is taken for relief valve operation or Doppler or moderator temperature reactivity feedback.

7. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 2). At the OL stage, this aspect of the review is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria (See par. C.2.b of Regulatory Guide 1.68.)

The reviewer evaluates the proposed test programs to determine if they provide a reasonable assurance that the components that provide overpressure protection will perform their safety function. As an alternative to this detailed evaluation, the reviewer may compare the overpressure protection design to that of a previously reviewed plant. If the design is essentially identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs for overpressure protection are adequate.

If the proposed design differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

8. The proposed plant technical specifications are reviewed to:

   a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

   b. Verify that the frequency and scope of periodic surveillance testing is adequate.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

1. BWR's

   "The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressure protection will be provided by ______ safety and relief valves located on the four main steam lines between the reactor vessel and the first isolation valve inside the drywell. The relief and safety valves are distributed among the four main steam lines such that a single accident cannot disable the safety, relief, or automatic depressurization functions. The valves discharge through piping to the suppression pool. The valves operate as spring-loaded safety valves with set pressures that range from _____ to _____ psig. Their total capacity at their set pressure is _____% of rated steam flow.

   "To determine the ability of the pressure relief system to prevent overpressurization, the applicant analyzed the severe transient of main steam isolation valve closure. The analysis was performed assuming that: a) the plant is in operation at design conditions (*% of rated steam flow and a reactor vessel dome pressure of * psig), and b) the reactor is shut down by a high pressure scram. The calculated peak pressure at the bottom of the vessel is ______ psig, a margin of _____ psi below the code allowable of _____ psig (110% of vessel design pressure). The staff concludes that the design of the pressure relief systems conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards and is acceptable."

2. PWR's

   "The pressure relief system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressure protection for the reactor coolant pressure boundary is accomplished by utilizing the _____ safety valves. These valves discharge to the pressurizer quench tank through a common header from the pressurizer. The reactor coolant system (RCS) safety valves, in conjunction with the steam generator safety valves, and the reactor protection system, will protect the RCS against overpressure in the event of a complete loss of heat sink.

*Normally, BWR's are analyzed at 105% rated steam flow at a pressure of 1040 psig.

"The peak RCS pressure following the worst transient is limited to the ASME Code allowable (110% of the design pressure) with no credit taken for operation of RCS relief valves, steam line relief valves, steam dump system, RCS pressurizer level control system, or pressurizer spray. The _______ plant was assumed to be operating at design conditions (_% of rated power) and the reactor is shut down by a _____ scram. The calculated pressure at the bottom of the vessel is _____ psig, a margin of ______ psi below the code allowable of _____ psig (110% of vessel design pressure).

The staff concludes that the design of the pressure relief system conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

2. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

3. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

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SECTION 5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1 and 14 require that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of a rapidly propagating failure and of a gross rupture. In addition, the reactor coolant pressure boundary shall be tested to quality standards commensurate with the importance of the safety function to be performed.

The following areas, which relate to materials of the reactor coolant pressure boundary (RCPB), are reviewed:

1. Material Specifications

   The specifications for pressure-retaining ferritic materials and austenitic stainless steels, including weld materials, that are used for each component (e.g., vessels, piping, pumps, and valves) of the reactor coolant pressure boundary, are reviewed.

   The adequacy and suitability of the ferritic materials specified for the above applications are reviewed. Similarly, the adequacy and suitability of stainless steels and nonferrous metals specified for the above applications are reviewed.

2. Compatibility of Materials with the Reactor Coolant

   Corrosion and stress-corrosion cracking induced by impurities in the reactor coolant can cause failures of the reactor coolant pressure boundary.

   The chemistry of the reactor coolant and the additives (such as inhibitors) whose function is to control corrosion are reviewed. The water chemistry includes the permissible concentrations of chlorides, fluorides, oxygen, hydrogen, and soluble poisons, the methods used to control the concentrations of impurities, and the pH.

The review includes the compatibility of the materials of construction employed in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant is reviewed. Similarly, a review is made of possible uses of austenitic stainless steels in the sensitized condition.

3. <u>Fabrication and Processing of Ferritic Materials</u>

   a. The fracture toughness properties of ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary are reviewed.

      The fracture toughness tests performed on all ferritic materials used for pressure-retaining RCPB components (i.e., vessels, pumps, valves, and piping) are reviewed.

      The test procedures used for Charpy V-notch impact and dropweight testing are reviewed.

      Fracture toughness of the material is characterized by its reference temperature, RT_{NDT}. This temperature is the higher of the nil-ductility temperature (NDT) from the dropweight test and the temperature that is 60°F below the temperature at which Charpy V-notch impact test data are 50 ft-lbs and 35 mils lateral expansion. The limiting RT_{NDT} temperature of the material is reviewed.

   b. The control of welding in ferritic steels is reviewed.

      (1) The quality of welds in low alloy steels can be increased significantly by proper controls. In particular, the propensity for cold cracks or reheat cracks to form in areas under the bead and in heat-affected zones (HAZ) can be minimized by maintaining proper preheat temperatures of the base metal concurrent with controls on other welding variables. The minimum preheat temperature and the maximum interpass temperature are reviewed.

      (2) The quality of electroslag welds in low alloy steel components can be increased by maintaining a weld solidification pattern that possesses a strong intergranular bond in the center of the weld. The welding variables, which have a significant effect on the weld solidification pattern, must be controlled. The welding variables, solidification patterns, macro-etch tests, and Charpy V-notch impact tests of electroslag welds are reviewed.

      (3) Experience shows that a welder qualified to weld low-alloy steel or carbon steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted. Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation. The adequacy of accessibility during the welding of ferritic components is reviewed.

   c. The requirements for non-destructive examination of ferritic wrought seamless tubular products used for components of nuclear power plants are specified in Paragraph NB-2550, ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III. These Code requirements cover the examination of several product forms (pipe, tubing, flanges, fittings) under a single category, "Seamless and Welded (Without Filler Metal) Tubular Products and Fittings," without specifying the examination method to be used for each product form. Instead, the Code simply states that the products must be examined by one of several methods listed. The methods of examination specified for nondestructive examination are reviewed.

4. Fabrication and Processing of Austenitic Stainless Steel

Austenitic stainless steels in a variety of product forms are used for construction of pressure-retaining components in the reactor coolant pressure boundary. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, are normally used. Because these compositions are susceptible to stress-corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking.

a. Sensitization is caused by intergranular precipitation of chromium carbide in austenitic stainless steels that are exposed to temperatures in the approximate range of 800°F to 1500°F. Precipitation increases with increasing carbon content and exposure time. Control of the application and processing of stainless steel is needed to eliminate the occurrences of stress-corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data and service experience demonstrate that sensitized stainless steel is significantly more susceptible to stress-corrosion cracking than nonsensitized (solution heat treated) stainless steel.

b. The following areas are reviewed: requirements for solution heat treatment of stainless steel; plans to avoid partial or severe sensitization during welding, including information on welding methods, heat input, and interpass temperatures; and a description of the material inspection program that will be used to verify that unstabilized austenitic stainless steels are not susceptable in service to intergranular attack.

   Contamination of austenitic stainless steel with halogens and halogen-bearing compounds (e.g., die lubricants, marking compounds, and masking tape) must be avoided to the maximum degree possible to avoid stress-corrosion craking. Plans for cleaning and protecting the material against contaminants capable of causing stress-corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of components and systems are reviewed. Any pickling used in processing austenitic stainless steel components and the restrictions placed on pickling sensitized materials are reviewed. The upper limit on the yield strength of austenitic stainless steel materials is reviewed.

c. Whether sensitized or not, austenitic stainless steel is subject to stress corrosion and must be protected from contaminants that can promote cracking. Thermal insulation is often employed adjacent to, or in direct contact with, stainless steel piping and components. The contaminants present in the thermal insulation may be leached by spilled or leaking liquids and deposited on the stainless steel surfaces. The controls on the use of nonmetallic thermal insulation are reviewed.

d. Austenitic stainless steel is subject to hot cracking (microfissuring) during welding if the weld metal composition or the welding procedure is not properly controlled. Because cracks formed in this manner are small and difficult to detect by non-destructive testing methods, welding procedures, weld metal compositions, and delta ferrite percentages that minimize the possibility of hot cracking must be specified. The adequacy of the proposed welding procedures is reviewed.

   The assurance of satisfactory electroslag welds for austenitic stainless steel components can be increased by maintaining a weld solidification pattern with a strong intergranular bond in the center of the weld. The welding variables that have a significant effect on the weld solidification pattern must be controlled. The welding variables that have a significant effect on the weld solidification pattern must be controlled. The welding variables, solidification patterns, and macro-etch tests used in the electroslag welding of austenitic stainless steel are reviewed.

   Experience has shown that a welder qualified to weld stainless steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted. Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation.

   The adequacy of accessibility for welding austenitic stainless steel components is reviewed.

e. The requirements for nondestructive examination of wrought seamless tubular products used for components of nuclear power plants are specified in Paragraph NB-2550 of the Code, Section III. Nondestructive examination techniques applied to tubular products used for components of the RCPB, or other safety-related systems that are designed for pressure in excess of 275 psig or temperatures in excess of 200°F, must be capable of detecting unacceptable defects regardless of defect shape, orientation, or location in the product.

   The nondestructive examination procedures used for inspection of tubular products are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Material Specifications

   The specifications for permitted materials are those identified in the Code, Section III, Appendix I, and described in detail in the Code, Section II, Parts A, B, and C.

2. Compatibility of Materials with the Reactor Coolant

   In boiling water reactors (BWR's), high purity water is maintained. The purity is monitored through on-line reading of the conductivity of the coolant and by continuously sampling and chemically analyzing it for chloride content. An on-line water treatment plant maintains the coolant within Technical Specification limits. In reactor coolants used for BWR's, oxygen seeks a natural level, and no attempt is made to control the amount of oxygen contained in the solution. The acceptance criteria for chemistry of the BWR reactor coolant are specified in Table 2 of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

In reactor coolants used for pressurized water reactors (PWR's), the conductivity measurements tend to show high values due to interference from additions of boric acid. These additions tend to mask the effect of other impurities. Therefore, sampling and chemical analysis for chlorides, fluorides, and oxygen must be performed on a scheduled basis. The acceptance criteria for pressurized water reactor coolant purity are stated in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." They require that chloride and fluoride ions be less than 0.15 parts per million (ppm) at all times and that the dissolved oxygen concentration be maintained below 0.10 ppm during periods when the material is above 250°F.

Ferritic low alloy steels and carbon steels, which are used in many principal pressure-retaining components, are clad with a layer of austenitic stainless steel. If cladding is not required by the Code, conservative corrosion allowances must be indicated for all exposed surfaces of carbon and low alloy steels, as indicated in the Code, Section III, NB-3120, "Corrosion."

Unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB must conform to requirements of Regulatory Guide No. 1.44, including verification of nonsensitization of the material by an approved test.

3. <u>Fabrication and Processing of Ferritic Materials</u>
   a. The acceptance criteria for fracture toughness are provided by General Design Criterion 31; the Code, Section III; and 10 CFR Part 50, Appendix G.

The pressure-retaining components of the RCPB that are made of ferritic materials must meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. With respect to absorbed energy in ft-lbs and lateral expansion as shown by Charpy V-notch (C_v) impact tests, all materials must meet the acceptance standards of Article NB-2330 of the Code, Section III, and the requirements of Sections IV.A.2, IV.A.3, and IV.B of Appendix G, 10 CFR Part 50, as follows:

(1) The special acceptance requirements for fracture toughness of reactor vessels are covered by Standard Review Plan 5.3.1, "Reactor Vessel Materials."
(2) Materials for piping (i.e., pipes, tubes, and fittings), pumps, and valves, excluding bolting materials, must meet the requirements of the Code, Section III, Paragraph NB-2332, and Appendix G, Paragraph G-3100. The required C_v values for piping are specified in Table NB-2332-1 of the Code, Section III.
(3) Materials for bolting for which impact tests are required must meet the requirements of the Code, Section III, Paragraph NB-2333, and Appendix G, Paragraph G-4100.

(4) Calibration of instruments and equipment must meet the requirements of the Code, Section III, Paragraph NB-2360.

b. The acceptance criteria for control of ferritic steel welding are listed below:

(1) The amount of specified preheat must be in accordance with the requirements of the Code, Section III, Appendix D, Paragraph D-1200, supplemented by Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steel."

The supplemental acceptance criteria for control of preheat temperature are as follows:

The welding procedure qualification requires that minimum preheat and maximum interpass temperatures be specified and that the welding procedure be qualified at the minimum preheat temperature.

For production welds, the preheat temperature should be maintained until a post-weld heat treatment has been performed.

Production welding should be monitored to verify that the limits on preheat and interpass temperature are maintained.

In the event that the above criteria are not met, the weld is subject to rejection. However, the soundness of the weld may be verified by an acceptable examination in accordance with the requirements of NB-5000, Code Section III.

(2) The acceptance criteria for electroslag welds are presented in positions C.1 through C.5 of Regulatory Guide 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns and impact test limits (for qualification of welds in Class 1 and Class 2 components) and the criteria for verifying conformance during production welding.

(3) Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for requalification of welders: the performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cms (12-14 inches) in any direction from the joint; and requalification is required for different restricted accessibility conditions or when any of the essential variables listed in the Code, Section IX, are changed.

c. Acceptance criteria for nondestructive examination of ferritic steel tubular products are given in Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." This guide provides new calibration standards and procedures for ultrasonic examination which have sufficient sensitivity to consistently detect defects regardless of shape or orientation.

4. <u>Fabrication and Processing of Austenitic Stainless Steel</u>

a. The acceptance criteria for testing, alloy compositions, and heat treatment, to avoid sensitization in austenitic stainless steels, are covered in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," items B and C.

b. Controls to avoid stress-corrosion cracking in austenitic stainless steels are also covered in Regulatory Guide 1.44. This guide provides acceptance criteria on the cleaning and protection of the material against contaminants capable of causing stress-corrosion cracking. The quality of water used for final cleaning or flushing of finished surfaces during installation is in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants."

   Laboratory stress-corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 90,000 psi.

c. The compatability of austenitic stainless steel materials with thermal insulation is dependent upon the type of insulation. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation which meets the criteria of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," is used. The acceptance criteria for nonmetallic insulation for stainless steel are based on the levels of leachable contaminants in the material and are presented in position C.2.b and Figure 1 of the guide.

d. The interim acceptance criteria for delta ferrite in austenitic stainless steel welds is shown in Branch Technical Position MTEB 5-1, which is appended. These acceptance criteria cover: (1) acceptance tests of weld filler metals, (2) the production welds that should be examined, and (3) the acceptance criteria for production welds.

   The acceptance criteria for electroslag welds in austenitic stainless steel are given in Regulatory Guide 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns for qualification of austenitic stainless steel welds and the basis for verifying conformance during production welding.

   Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for requalification of welders:

   (1) The performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cms (12-14 inches) in any direction from the joint.
   (2) Requalification is required for different restricted accessibility conditions or when other essential variables listed in the Code, Section IX, are changed.

e. The acceptance criteria for nondestructive examination of austenitic stainless steel tubular products are shown in Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." New calibration standards and procedures for ultrasonic examination are incorporated that provide a sensitivity which will consistently detect defects regardless of shape or orientation.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review described in Section I of this plan, the following review procedures are followed.

1. Material Specifications

   The material specifications for each major pressure-retaining component or part used in the RCPB are compared with the acceptable specifications listed in the Code, Sections II and III, as stated in the acceptance criteria. Exceptions to the material specifications of the Code are clearly identified, and the basis evaluated. The reviewer judges the significance of the exceptions and, taking into account precedents set in earlier cases, determines the acceptability of the proposed exceptions. In those instances where the Materials Engineering Branch takes exception to the use of a specific material or questions certain aspects of a specification, the applicant is advised which material is not acceptable, and for what reason.

2. Compatibility of Materials with the Reactor Coolant

   The reviewer verifies that the following information is provided at each respective stage of the review process:

   a. At the construction permit stage of review:

      (1) A list of the materials of construction of the components of the reactor coolant pressure boundary that are exposed to the reactor coolant, including a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed in service.

      (2) A list of the materials of construction of the RCPB, and a description of material compatibility with external insulation and with the environment in the event of reactor coolant leakage.

      (3) The limitations imposed on concentrations of chloride and fluoride ions and oxygen in the reactor coolant, and the extent of monitoring such limitations.

      (4) The fabrication and cleaning controls imposed on stainless steel components to minimize contamination with chloride and fluoride ions.

      (5) The controls and limits that are specified for leachable impurities in thermal insulation, as identified in Section II.4.c.

      (6) For BWR's, the demineralizer capacity and performance monitoring suggested in Regulatory Guide No. 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

   b. At the operating license stage of the review process:

      (1) The items listed under 2.a above, to provide assurance that any changes are noted that may have occurred during the period between the submittal of SAR's.

      (2) A list of the instrumentation and equipment that will monitor and control the purity of the reactor coolant, including water purity indicators and alarms provided in the control room.

3. Fabrication and Processing of Ferritic Materials

a. The information submitted by the applicant relative to tests for fracture toughness is reviewed for conformance with the acceptance criteria stated in Section II.3.a. These tests include Charpy V-notch impact and dropweight tests. A description of the tests is reviewed, and the locations of the test specimens and their orientation are verified. Information regarding calibration of instruments and equipment is reviewed for conformance with the acceptance criteria stated in Section II.3.a.(4).

   In the event that none of the fracture toughness tests has been performed, the preliminary safety analysis report (PSAR) must contain a statement of the applicant's intention to perform this work in accordance with the Code, Section III, Paragraph NB 2300 and Appendix G; and the requirements of 10 CFR 50, Appendix G.

   The final safety analysis report (FSAR) is reviewed to assure that all the impact tests required by NB 2320 have been performed.

b. The control of welding in ferritic steels is reviewed as described below:

   (1) The information submitted by the applicant regarding the control of preheat temperatures for welding low alloy steel is reviewed for conformance with the acceptance criteria stated in Section II.3.b.(1).

   (2) The electroslag weld information submitted by the applicant is reviewed for conformance to the acceptance criteria discussed in Section II.3.b.(2). A number of electroslag welding process variables, such as slag pool depth, electrode feed rate and oscillation, current, voltage, and slag conductivity, have been shown to influence the weld solidification pattern. If the combination of process variables produces a deep pool of molten weld metal, the crystal (dendritic) growth direction from the pool sides will join at an obtuse angle at the center of the weld, and cracks may develop because of the weaker centerline bond between dendrites. A proper combination of process variables promotes a dendritic growth pattern with an acute joining angle, which results in a strong centerline bond.

   The information in the SAR is reviewed to verify that macroetch tests have been made (to assure that an acceptable weld solidification pattern is obtained) and that impact tests specified in Regulatory Guide 1.34 meet the acceptance criteria discussed previously in Section II.3.b.(2).

   (3) The ASME Code, Section III, requires adherence to the requirements of Section IX, "Welding Qualifications." One of the requirements is welder qualification for production welds. However, there is a need for supplementing this section of the Code because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by qualifying the welder under conditions simulating the space limitations under which the actual welds will be made.

Regulatory Guide 1.71, "Welder Qualification for Limited Accessibility," provides the necessary supplement to the Code, Section IX, in this respect. The information submitted by the applicant is reviewed for conformance with acceptance criteria discussed in Section II.3.b.(3).

c. The ASME Code, Section III, NB 2552 specifies the ultrasonic method for examination of ferritic steel tubular products. Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products" provides new calibration standards and procedures for ultrasonic examinations which are considered more sensitive, and consistently able to detect defects regardless of shape or orientation. The reviewer verifies that ultrasonic examination of the subject product are in accordance with this Regulatory Guide.

4. Fabrication and Processing of Austenitic Stainless Steels

a. The information submitted by the applicant in the following areas is reviewed for conformance with the acceptance criteria stated in Section II.4.a regarding:

(1) The desirable stage in the sequence of processing for solution heat treatment, the rates of cooling, and the quenching media.

(2) Controls to prevent sensitization during welding including:

i. Avoiding welding practices that result in the generation of high heat.

ii. Maintaining low heat input by controlling current, voltage, and travel speed.

iii. Limiting interpass temperature.

iv. Using stringer bead techniques and avoiding excessive weaving.

v. Limiting the carbon level where section thickness makes the material more prone to sensitization.

(3) Controls to verify non-sensitization are described in position C.3 of Regulatory Guide 1.44.

In the event that information in the above areas is not supplied, sufficient justification for the deviation must be presented.

b. The information submitted by the applicant is reviewed for conformance with the acceptance criteria discussed in Section II.4.b as follows:

Verification is sought that process controls are exercised during all stages of component manufacture and reactor construction to minimize the exposure of austenitic stainless steels to contaminants that could lead to stress-corrosion cracking.

Information is also checked to assure that precautions have been taken to require removal of all cleaning solutions, processing compounds, degreasing agents, and any other foreign material from the surfaces of the component at any stage of processing prior to any elevated temperature treatment and prior to hydrotests. The reviewer verifies that a statement is contained in the SAR that pickling of austenitic stainless is avoided and that the quality of water used for final cleaning or flushing of finished surfaces during installation is in accordance with acceptance criteria discussed in Section II.4.b.

Because excessive cold work in austenitic stainless steel can render this material susceptible to stress-corrosion cracking, control must be exerted by the applicant, by placing an upper limit on the yield strength, in accordance with the acceptance criteria discussed in Section II.4.b. Verification is obtained that the applicant has such a control measure.

c. The information submitted by the applicant is reviewed to determine the type of insulation used and to determine its compatibility with the austenitic stainless steel used in construction of the component.

There are no compatibility concerns with the use of reflective metal insulation; the chief compatibility concern is with the use of nonmetallic insulation. A review is performed to assure that any such material specified by the applicant is in conformance with the acceptance criteria stated in Section II.4.c. Verification is obtained that the material has been chemically analyzed by methods equivalent to those prescribed in Regulatory Guide 1.36 and that evidence is obtained that the levels of leachable contaminants are such that stress corrosion of stainless steel will not result from use of the insulation.

d. The information submitted by the applicant regarding control of delta ferrite in austenitic stainless steel welds is reviewed to determine its conformance with the acceptance criteria stated in Section II.4.d. The information submitted must state that appropriate filler metal acceptance tests have been conducted and that a certified materials test report has been received. The information should state, also, the applicant's program for testing production welds and his sampling plan for examination of welds having less than 3% average of delta ferrite.

The information submitted by the applicant regarding control of electroslag weld properties for austentic stainless steel materials is reviewed for conformance with the acceptance criteria discussed in Section II.4.d.

The review of information on the control of electroslag weld properties in austenitic stainless steels is essentially the same as that discussed previously for ferritic steels. However, because electroslag-welded austenitic stainless steels have very high impact resistance and because the Code, Section III, is not concerned with impact testing of these welds, the checks are: (1) a macroetch test is used to provide assurance that the solidification pattern is in accordance with the requirement of the acceptance criteria shown in Section II.4.d, and (2) wrought stainless steel parts are solution heat treated after welding.

The review procedure for information submitted on welder qualification for limited accessibility areas, applicable to austenitic stainless steels, is the same as that for ferritic steels, which has been discussed previously under Section III.3.b.(3).

e. The procedures for review of nondestructive examination of tubular products fabricated from austenitic stainless steel are the same as those discussed for similar ferritic products in Section III.3.c of this plan, and the acceptance criteria are as shown in Section II.4.e.

5. General

If the information contained in the safety analysis reports or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information but also the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the applicable acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Special requirements of the applicant with regard to control of residual elements in ferritic materials have been identified and are considered acceptable.

"The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials, except unclad carbon and low alloy steel, is negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of the Code, Section III.

"The materials of construction for the RCPB are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels."

"The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," and Regulatory Guide 1.56, "Maintenance of Water Purity in BWR's," and provide reasonable assurance that the RCPB components will be adequately protected during operation from conditions that could critically lead to stress corrosion of the materials and loss of structural integrity of a component. The instrumentation and sampling provisions for monitoring reactor coolant water chemistry provide adequate measurement capability for detecting significant changes on a timely basis. Compliance with the recommendations of these Regulatory Guides constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 14 and 31.

"The fracture toughness tests required by the ASME Code, augmented by Appendix G, 10 CFR 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant pressure boundary.

"The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and AEC Regulations in establishing safe operating procedures, provides adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

"The controls imposed on welding preheat temperatures are in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

"The controls imposed on electroslag welding of ferritic steels are in accordance with the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties," and provide assurance that welds fabricated by the process will have high integrity and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance, and postulated accident conditions.

"The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary conform to the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding," Regulatory Guide 1.34, "Control of Electroslag Weld Properties," Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products." Material selection, fabrication practices, examination procedures, and protection procedures performed in accordance with these recommendations provide reasonable assurance that the austenitic stainless steel in the reactor coolant pressure boundary will be in a metallurigical condition which precludes susceptibility to stress-corrosion cracking during service. Conformance with these Regulatory Guides constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 14."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, Section III, and Section IX, American Society of Mechanical Engineers.

4. ASTM, A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, Part 3, American Society of Testing and Materials.

5. ASTM E 23-72, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, American Society of Testing and Materials.

6. ASTM E-208-69, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

7. Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

8. Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

9. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants."

10. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel."

11. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

12. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

13. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

14. Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products."

15. Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

16. Branch Technical Position MTEB 5-1, "Interim Position of Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended.

17. Standard Review Plan 5.3.1, "Reactor Vessel Materials."

INTERIM POSITION ON REGULATORY GUIDE 1.31, "CONTROL OF STAINLESS STEEL WELDING"

A. Background

This interim position is required until such time as Regulatory Guide 1.31, Revision 2, is issued by the AEC.

Fabrication of welded austenitic stainless steel in Code Class 1, 2, 3, and CS components should comply with the requirements of Section III and Section IX of the ASME Boiler and Pressure Vessel Code supplemented by the Branch Technical Position given below.

B. Branch Technical Position

1. The weld filler materials used shall meet the acceptance test requirements of Section III of the ASME Boiler and Pressure Vessel Code, plus the following additional requirements:

   a. Delta ferrite determinations should be performed on undiluted weld deposits for each lot and heat of austenitic stainless steel weld metal (Par. QW 422 of Section IX), except that delta ferrite determinations will not be required for SFA-5.4 Type 16-8-2 weld metal, nor for austenitic stainless steel weld filler metal to be used for weld metal cladding. Delta ferrite determinations for consumable inserts, rod, or wire filler metal, used with the gas tungsten arc (GTA) process or the plasma arc welding process, may be predicted by using an applicable constitution diagram,[1] to demonstrate compliance with the position on amount of delta ferrite given in 1.b, below.

   b. For all processes other than GTA and plasma arc, delta ferrite determinations shall be made on undiluted weld deposits. An acceptable method for achieving this employs a weld pad made and tested in conformance with the applicable sketch and methods described in the American Welding Society Specification SFA-5.4. The undiluted weld deposits should contain between 5 and 20 percent delta ferrite or its equivalent ferrite number.

   c. Chemical analyses should be performed on undiluted weld deposits, except when GTA or plasma arc processes are used, as indicated above.

2. The results of the destructive and nondestructive tests required in position 1 above should be included in a Certified Materials Test Report as required by the ASME Code, Section III, NB-2130 or NB-4130.

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[1] Schaeffler, Modified Schaeffler, or Delong Diagram, American Society for Metals Handbook, Vol. 6, pp. 246-247.

The provisions of positions 3 and 4 in Regulatory Guide 1.31 are no longer deemed necessary and are deleted. Revisions to positions 5 and 6 of the guide have been made and are shown in the new positions 3 and 4 given below. Position No. 7 of Regulatory Guide 1.31 has been deleted.

3. a. Production welds, except fillet welds having a throat dimension 3/8 inch or less, and butt welds less than 1/4 inch in thickness should be examined by magnetic measurement methods to verify that adequate delta ferrite levels are present. Welds 1 inch or greater in thickness shall be examined on a 100% basis. A sampling plan may be used for examination of welds less than 1 inch in thickness. The examination should show that each weld contains at least 3% delta ferrite based on the average of four test readings taken on the face of the completed weld deposit, at the centerline of the weld and at 1/4 weld-length intervals. Instrument readings should not be taken at "start and stop" locations or in weld beads adjacent to the base materials. The four instrument readings used for determination of the average should not include any reading below one percent delta ferrite. Weld locations that show 1% or less delta ferrite may be reexamined, to determine whether the reading represents a local condition.

   The magnetic instruments used for examination of weld pad delta ferrite should have been calibrated using secondary standards traceable to National Bureau of Standards standards, and to a Magne-gage using the procedures shown in the Welding Council document of July 1, 1972, "Calibration Procedure for Instruments to Measure the Delta Ferrite Content of Austenitic Stainless Steel Weld Metal," and as supplemented by procedures shown in American Welding Society AWS Specification A 4.2-74.

   b. The upper limit on delta ferrite shown in position 1.b above need not be applied for welds that do not receive heat treatment subsequent to welding, nor for consumable inserts.

4. In the event that position 3.a above is not met, the non-conforming production welds may be evaluated for acceptability using either a or b, below.

   a. An analysis of service requirements of the weldment, and comparison of these requirements with the ASME Code criteria for fatigue strength, but using conservative fatigue data that account for weld metal with microfissures.

   b. An examination of the weld or welds to demonstrate the absence of unacceptable fissures or cracks. Where the production weld is below the minimum acceptable level of delta ferrite, a sample of the weld shall be removed and a metallographic examination or a bend test shall be made on a transverse section to determine the presence or absence of excessive fissures. The acceptance criteria are as follows: fissures 1/64 inch and less shall not be counted. The presence of a single tear or fissure larger than 1/16 inch, or of a greater number than 3 of size between 1/64 inch and 1/16 inch in any 0.25 square inches of weld metal surface shall constitute failure of the test.

c. Welds found unacceptable shall be repaired and reexamined for delta ferrite content in accordance with the procedure shown in 3.a above.

An example of a suitable examination plan is discussed in Attachment 1.

EXAMPLE OF A SUITABLE STATISTICAL SAMPLING PLAN

A. Examination (See Table I)

1. Production Welds Greater Than One (1) Inch in Thickness:
Delta ferrite determinations will be made on the completed surface of all such welds. When observed average delta ferrite levels of 3% or more are indicated, all of the welds will be considered acceptable, and no further testing is needed.

   In the event that the delta ferrite level in some welds is lower than an average of 3%, a metallographic examination or a macroscopic examination performed on transverse side-bend specimens to determine the presence of microfissures will be made. The specimens for metallographic or macroscopic examination will be selected from the welds exhibiting delta ferrite levels lower than 3% average in accordance with Table II.

2. Production Welds One Inch or Less in Thickness: (See Table I)
Delta ferrite determinations will be made on the completed surface of such welds, selected in accordance with Table II. If observed delta ferrite levels of 3% or more are indicated for the sample welds, (i.e., Column 2, Table II) the entire batch of welds (i.e., Column 1), that the sample represents shall be considered acceptable. If the number of welds in the sample size having less than an average of 3% delta ferrite equals or exceeds the rejection level (Column 3), all the welds in the batch (Column 1) will have to be inspected.

   For welds having an average delta ferrite level less than 3%, a macroscopic examination will be performed on transverse side-bend specimens to determine the presence of microfissures, or metallographic examination will be made on specimens cut from the welds. Specimens for such examination will be selected from welds exhibiting delta ferrite levels lower than an average 3% in accordance with Table II.

3. Sample lots or batches for welds greater than one inch in thickness, and for welds one inch or less in thickness, will not be grouped together, and will be macroscopically examined on a separate batch basis.

B. Acceptance and Rejection for Delta Ferrite Levels and Microfissures (All production welds)

1. All welds having an average delta ferrite level of 3% or more are acceptable.

2. Welds having less than 3% average delta ferrite shall have transverse side-bend tests taken and the welds shall be examined macroscopically for fissures, or the welds shall be examined metallographically, with sampling to be in accordance with Table II. Microfissuring detected during these inspections shall be evaluated by the following criteria: fissures 1/64 inch and less shall not be counted. The presence of a single tear or fissure larger than 1/16 inch, or of a greater number than 3 of a size between 1/64 inch and 1/16 inch in any 0.25 square inches of weld metal surface, shall constitute failure of the test.

3. Welds found unacceptable shall be repaired and reexamined by the above procedure.

TABLE I

Thickness of weld	No. of Welds to be magnetically inspected	Welds having average 3% or more delta ferrite	Welds having less than average 3% delta ferrite
one (1) inch or greater	all welds	OK - no further examination necessary	Inspect for fissures per Table II
Less than one (1) inch	Inspect on a sampling basis per Table II	If all of sample batch are OK no further exam. required for entire group	Inspect for fissures per Table II

TABLE II

Column 1	Column 2	Column 3
Total No. of welds	Sample Size - No. of welds to be examined	Rejection level*
2-8	2	1
9-15	4	1
16-25	6	2
26-50	10	2
51-90	16	2
91-150	26	2
151-280	40	4
281-500	64	5
501-1200	100	7
1201-3200	160	9
3201-10,000	250	13

*If the welds examined and found unacceptable reach the figure shown in column 3, the welds shown in the representative batch in column 1 shall be rejected.

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SECTION 5.2.4 REACTOR COOLANT PRESSURE BOUNDARY INSERVICE INSPECTION AND TESTING

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires that components which are part of the reactor coolant pressure boundary (RCPB) shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The following areas relating to the inservice inspection program for AEC Quality Group A components of the RCPB are reviewed. These components are also ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Code Class 1 components. Inservice inspection programs are based on Section XI of the Code, "Rules for Inservice Inspection of Nuclear Power Components."

1. System Boundary Subject to Inspection
The inservice inspection (ISI) program for those portions of the reactor coolant pressure boundary consisting of Code Class 1 components is reviewed. Steam generator inservice inspection is covered separately in Standard Review Plan 5.4.2.2, "Steam Generator Inservice Inspection." Augmented inservice inspection for high energy fluid system piping between containment isolation valves is reviewed in Standard Review Plan 6.6.

2. Accessibility
The descriptive information that pertains to the general and specific provisions for access to components covered by the Code, Section XI, is reviewed. In addition, the remote access equipment needed to perform inspections in a radiation field is reviewed.

3. Examination Techniques and Procedures
The descriptive information that pertains to Section XI, Tables IWB-2500 and IWB-2600 is reviewed.

4. Inspection Intervals
The schedules of examinations and inspections in the applicant's safety analysis report (SAR) and plant Technical Specifications are reviewed. In addition, those inspections

which are performed during the inspection interval, such as during refueling outages, are reviewed.

5. Examination Categories and Requirements
The Technical Specification tabulation of examination categories is reviewed. The SAR areas of review include category designation and the area and extent of examination of each category.

6. Evaluation of Examination Results
   a. The proposed evaluation methods for any indications of structural defects detected during ISI examinations are reviewed.
   b. The repair procedures proposed for components that reveal unacceptable structural defects during ISI examinations are reviewed.

7. System Leakage and Hydrostatic Pressure Tests
The descriptive information on leak tests and hydrostatic pressure tests of Code Class 1 components is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I are as follows:

1. System Boundary Subject to Inspection
The applicant's definition of the RCPB is acceptable if in agreement with the following criteria: For pressurized water reactor (PWR) and boiling water reactor (BWR) nuclear power systems, the inspection requirements of Section XI of the Code must be met for all Class 1 pressure-containing components (and their supports) except for those components excluded under IWB-1220 of Section XI. The system boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:
   a. The outermost containment isolation valve in system piping that penetrates the primary reactor containment.
   b. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.
   c. The reactor coolant system safety and relief valves.

2. Accessibility
The design and arrangement of system components are acceptable if in accordance with IWA-1500, "Accessibility," of Section XI. Adequate clearances for general access are demonstrated as follows:
   a. Sufficient space is provided for personnel and equipment to perform inspections.
   b. Provisions are made for the removal and storage of structural members, shielding components, and insulating materials, to permit access to the components being inspected.
   c. Provisions are made for hoists and other handling machinery needed to handle items in b, above.
   d. Provisions are made for alternative examinations in the event structural defects or indications reveal that such alternative examinations are required.

e. Provisions are made for the necessary operations associated with repair or replacement of system components and piping.

3. Examination Techniques and Procedures

The applicant's examination techniques and procedures used for ISI of the system are acceptable if in agreement with the following criteria:

a. The visual examination techniques are acceptable if in agreement with IWA-2210 of Section XI of the Code. A visual examination must be employed as a basis for a report of the general condition of the part, component, or surface. The report must include such conditions as scratches, wear, cracks, corrosion or erosion of the surfaces, misalignment or movement of the part or component, and evidence of leaking. Surface replication methods are considered acceptable provided the surface resolution is at least equivalent to that obtainable by visual observation.

b. The surface examination techniques are acceptable if in agreement with IWA-2220 of Section XI of the Code. A surface examination is required to verify the presence of surface or near surface cracks or discontinuities. The surface must not be immersed or flooded with water at the time of examination. Acceptable techniques are magnetic particle examination and liquid penetrant examination.

c. The volumetric examination methods are acceptable if in agreement with IWA-2230 of Section XI of the Code. A volumetric examination is required to indicate the presence of subsurface discontinuities with a method or technique capable of examining the entire volume of metal beneath the surface. Specific acceptable methods of volumetric examination are radiographic examination and ultrasonic examination.

d. Alternative examination methods to those given above in a, b, and c, are acceptable provided the results are equivalent or superior. The acceptance standards for these alternative methods are given in Section XI, IWB-3100, "Evaluation of Nondestructive Examination Results."

4. Inspection Intervals

The inservice inspection program is acceptable if the required examinations and pressure tests are completed during each ten-year interval of service, hereafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of IWA-2400, "Inspection Intervals," of Section XI of the Code.

It is intended that inservice examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval. Except as specified in Table IWB-2500, "Examination Categories," at least 25% of the required examinations must be completed by the expiration of one-third of the inspection interval. Credit is allowed for no more than 33-1/3% of the required ISI even though additional examinations are completed during this period. At least 50% of the required examination must be completed by the expiration of the second one-third of the inspection interval. Credit is allowed for no more than 66-2/3% of the required inspections. The remaining required examinations shall be completed by the end of the inspection interval.

5. <u>Examination Categories and Requirements</u>

The examination categories and requirements as specified in the SAR are acceptable if in agreement with the criteria of IWB-2500 and IWB-2600 of Section XI of the Code. Every area subject to examination falls within one or more of the examination categories indicated in Table IWB-2500 and must be examined at least to the extent specified. The method of examination for the components and parts of the pressure-containing and pressure-retaining boundaries that are listed in the requirements of IWB-2600 of Section XI are tabulated in Table IWB-2600.

6. <u>Evaluation of Examination Results</u>

a. The standards for examination evaluation are acceptable if in agreement with the requirements of Section XI, IWB-3000, "Standards for Examination Evaluations." The applicant's program for flaw evaluation is acceptable if it agrees with Table IWB-3410, "Evaluation Standards."

b. The proposed program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is acceptable if in agreement with the requirements of Section XI, IWB-4000, "Repair Procedures." The criteria that establish the need for repair or replacement are described in Section XI, IWB-3000.

7. <u>System Leakage and Hydrostatic Pressure Tests</u>

The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program agrees with the requirements of Section XI, IWB-5000, "System Leakage and Hydrostatic Pressure Tests." IWB-5222, "System Hydrostatic Test Pressure," presents criteria and a table of equivalent test temperatures versus test pressures at which the system must be tested. The applicant's program is acceptable if in agreement with IWB-5222 in regard to the temperature-pressure relationship of the system at test, and if in agreement with the Technical Specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, these limitations may be more severe than IWB-5222.

III. <u>REVIEW PROCEDURES</u>

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. <u>System Boundary Subject to Inspection</u>

The information furnished in the SAR is reviewed for agreement with Section II.1 of this plan, and to verify that any differences between the applicant's definition of the RCPB and Section II.1 are identified and justified by the applicant, e.g., "Pressurizer: not applicable, as plant is a BWR." or, "no longitudinal welds in beltline region as vessel is constructed of forged rings."

2. <u>Accessibility</u>

The descriptive information concerning accessibility furnished in the SAR is reviewed for compliance with Section II.2. The reviewer verifies that the clearances supplied for general access to the system component listed in Table IWB-2500 of Section XI are adequate.

The reviewer verifies that adequate provisions are made for remote inspection of those components affected by radiation fields after plant start-up. These components include the beltline welds and reactor vessel nozzle interior surfaces. The reviewer verifies that remote inspection devices proposed for periodic inservice inspections will be used for the preservice baseline inspection program to demonstrate feasibility.

3. Examination Techniques and Procedures

The reviewer verifies that the examination techniques described by the applicant are the same as those in Section II.3. If alternative examination methods are proposed by the applicant, they are reviewed to verify that the results are equivalent or superior to those in IWA-2210, 2220, and 2230 of Section XI, and that the acceptance standards of IWB-3100 of Section XI are met.

4. Inspection Intervals

The Technical Specification program for inservice inspection is reviewed to establish that the inspection schedule for every area and component in the program is in agreement with Section II.4.

5. Examination Categories and Requirements

The descriptive information in the SAR and the Technical Specification ISI program are reviewed to establish that the applicant followed the requirements of Section II.5. The reviewer determines that the table supplied in the Technical Specifications follows Table IWB-2600 of Section XI where it is applicable to the given reactor system, and that the table contains the following headings and applicable information: Examination Category, Components and Parts to be Examined, Method, Extent and Frequency of Examinations, and Comments.

6. Evaluation of Examination Results

The criteria statements provided by the applicant are reviewed for agreement with Section II.6 as follows:

a. The reviewer verifies that the applicant's criteria incorporate IWB-3000 of Section XI regarding standards for examination evaluation.
b. The reviewer verifies that the applicant's criteria incorporate IWB-4000 of Section XI regarding repair procedures.

7. System Leakage and Hydrostatic Pressure Tests

The reviewer determines that the Technical Specification on hydrostatic pressure testing for system leakage of the RCPB adheres to Section II.7 of this plan and incorporates the table in IWB-522, Section XI. The Technical Specification on operating limitations during heatup, cooldown, and system hydrostatic pressure testing must be referenced.

IV. EVALUATION FINDINGS

The reviewer verifies that adequate information is provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. The applicant has stated that the designs of Code Class 1 components of the reactor coolant pressure boundary incorporate provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that methods will be developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel. The conduct of periodic inspections and leakage and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of General Design Criterion 32."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."
2. ASME Boiler and Pressure Vessel Code, Section XI, Section III, "Nuclear Power Plant Components," and "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

Section 5.2.5 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mecnanical Engineering Branch (MEB)

I. AREAS OF REVIEW

General Design Criterion 30 (Ref. 1) requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

The areas of the SAR relating to the reactor coolant pressure boundary (RCPB) leakage detection systems are reviewed. The descriptive information and supporting figures, tables, and graphs are reviewed to establish that sufficient information is provided to permit a reasonable evaluation of the applicant's compliance with Regulatory Guide 1.45 (Ref. 2), as follows:

1. Collection of Identified Leakage

A limited amount of leakage is expected from components of the RCPB within the containment, such as valve stem packing glands, circulating pump shaft seals, and other equipment that cannot practically be made completely leaktight. The reactor vessel closure seals and safety and relief valves should not leak significantly; however, leakage occurring via these paths or via pump and valve seals is detectable and collectable and, to the extent practical, should be isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leaks are known as "identified leakage" and are piped to tanks or sumps so that the flow rate can be established and monitored during plant operation. The provisions for collecting and monitoring leakage from known leak sources are reviewed.

2. Unidentified Leakage to Containment

Uncollected leakage to the containment atmosphere increases the humidity of the containment. The moisture removed from the atmosphere by air coolers together with any associated liquid leakage to the containment is known as "unidentified leakage" and is collected in tanks or sumps where the flow rate can be established and monitored during plant operation. Unidentified leakage to the containment atmosphere should be kept to a minimum to permit the leakage detection systems to detect positively and rapidly a small increase in flow rate. Identified and unidentified leakages should be separated so that a small unidentified leakage that is of concern will not be masked by a larger acceptable identified leakage. Provisions for the detection and control of leakage to containment are reviewed.

3. Leakage Detection Methods

Continuous monitoring of both identified and unidentified leakage rates is important. Effective systems for detecting and locating unidentified leakage are needed. The following describes some detection methods commonly used.

The primary monitors determine flow rates and flow rate changes to tanks and sumps. Methods to indicate when and where coolant is being released to the containment atmosphere include detection of changes in airborne particulate radioactivity, airborne gaseous radioactivity, containment atmosphere humidity, pressure, and temperature, condensate flow rate from air coolers.

4. Intersystem Leakage

Substantial intersystem leakage from the RCPB to other systems across passive barriers or valves is not expected. However, should such leakage occur, it should be detectable by the alarm and detection methods which are employed. For example, steam generators in pressurized water reactors (PWR's) are monitored to detect tube sheet leaks.

Since intersystem leakage does not release reactor coolant to the containment atmosphere, detection methods include monitoring of radioactivity in the connected systems where the flow is through the containment boundary, and monitoring of airborne radioactivity where such systems are vented outside the containment boundary. Another important method of obtaining indications of uncontrolled or undesirable intersystem flow is the use of a water inventory balance, designed to provide appropriate information such as abnormal water levels in tanks and abnormal water flow rates.

5. System Sensitivity and Response Time

Since leakage detection methods or systems differ in sensitivity and response time, prudent selection of detection methods should include a sufficient number of systems to ensure effective monitoring during periods when some detection systems may be ineffective or inoperable. Some of these systems should serve as early alarm systems which signal the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required. It is essential that leakage detection systems have the capability to detect significant RCPB leakage as soon after occurrence as practical to minimize the potential for a gross boundary failure. Cracks that might develop and penetrate the RCPB wall are expected to exhibit very slow growth, and to afford ample time for a safe and orderly plant shutdown after a leak is detected. An early warning signal is necessary to permit proper evaluation of all unidentified leakage.

6. Seismic Capability of Systems

Since nuclear power plants may be operating at the time an earthquake occurs and may continue to operate after earthquakes, the leakage detection systems should be designed to continue functioning after seismic events. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it is important that the operator be able to assess the condition within the containment quickly. The proper functioning of at least one leakage detection system is essential in evaluating the seriousness of the condition within the containment in the event leakage has developed in the RCPB. The MEB reviews the seismic qualification of the electrical and instrumentation portion of the leak detection system in SRP 3.10 (Ref. 3).

7. Quantitative Interpretation of Indicators and Alarms

It is important to be able to associate a signal or indication of a departure from the normal operating conditions with a quantitative leakage flow rate. Except for flow rate or level change measurements from tanks, sumps, or pumps, signals from other leakage detection systems do not provide information readily convertible to a common denominator. Approximate relationships converting these signals to units of water flow are formulated to assist the operator in interpreting signals. The instrumentation associated with the leak detection system is reviewed by EICSB in SRP 7.5 (Ref. 4). Procedures for operator evaluation of leakage conditions are reviewed by RSB.

8. Testability

Provisions for testing the various systems during plant operation should be provided. EICSB ensures that leakage detection equipment is tested and calibrated in compliance with IEEE Std. 279-1971 (Ref. 5).

9. Technical Specifications

RSB reviews the limiting conditions for operation that appear in the technical specifications. Leakage limits for unidentified and total leakage, maximum time allowed to operate after a leak is discovered, and action to take in the event of instrument malfunction are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are stated in Regulatory Guide 1.45 (Ref. 2). According to this guide the source of reactor coolant leakage should be identifiable to the extent practical. Reactor coolant pressure boundary leakage detection and collection systems are acceptable if they are in accordance with the following:

1. Collection of Identified Leakage

Leakage to the primary reactor containment from identified sources should be collected or otherwise isolated so that the flow rates are monitored separately from unidentified leakage, and the total flow rate can be established and monitored.

2. Collection and Monitoring of Unidentified Leakage

Leakage to the primary reactor containment from unidentified sources should be collected and the flow rate monitored with an accuracy of one gallon per minute (gpm) or better.

3. Leakage Detection Methods

At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be selected from either the monitoring of condensate flow from air coolers, or monitoring of airborne gaseous radioactivity.

Humidity, temperature, or pressure monitoring of the containment atmosphere are to be considered as alarms or indirect indications of leakage to the containment.

4. Intersystem Leakage

Provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage. Detection methods include radioactivity monitoring and indicators to show abnormal water levels or flow in the potentially affected systems and unaccountable increases in reactor coolant make-up flow.

5. System Sensitivity and Response Time

The sensitivity and response time of each leakage detection system employed for monitoring unidentified leakage to the containment should be adequate to detect an increase in leakage rate, or its equivalent, of one gpm in less than one hour.

6. Seismic Capability of Systems

The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown and the airborne particulate radioactivity monitoring system should be capable of remaining functional when subjected to the safe shutdown earthquake (SSE).

7. Indicators and Alarms

Indicators and alarms for each leakage detection system should be provided in the main control room and procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for the independent variables such as, in the case of an air particulate monitor, the isotope being monitored, plateout, and decay rate. Each system should be set to alarm on an increase in leakage of 1 gpm above the background level determined at the time of calibration.

8. Testing

The leakage detection systems should be equipped with provisions to permit calibration and operability tests during plant operation.

9. Technical Specifications

The technical specifications should include, in the limiting conditions for operation, the maximum permissible total and unidentified leakage, and address the availability of the leakage detection systems to ensure adequate coverage at all times. The leakage limits are established on the basis of current practice for similar types of nuclear steam supply systems.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For the operating license (OL) review, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and allowable leakage rates.

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. Collection of Identified Leakage

   Information concerning the collection of identified leakage is reviewed for agreement with Acceptance Criterion II.1. The reviewer verifies that the SAR description of the reactor vessel flange leakage monitoring, leakage monitors for other vessel flanges, and valve and pump seal leakage monitors is complete, and that this monitored leakage will be collected in tanks or sumps where its rate of accumulation will be summed to obtain an identified leak rate. The reviewer should establish that the identified leakage is not only collected and monitored but in such a fashion as to prevent identified leakages from masking unidentified leaks.

2. Collection and Monitoring of Unidentified Leakage

   Information concerning the collection and monitoring of unidentified leakage is reviewed for agreement with Acceptance Criterion II.2.

3. Leakage Detection Methods

   The information describing the number of systems and operating principles of each system, including schematic diagrams, is reviewed to assure that sufficient information is provided to comply with Acceptance Criterion II.3. The review consists of a side-by-side comparison of the applicant's methods and the acceptance criterion and a determination that the number and type of methods provided are acceptable.

4. Intersystem Leakage

   Information describing the intersystem leakage detection system is reviewed for compliance with Acceptance Criterion II.4. The reviewer determines that radiation monitoring systems have been provided for possible intersystem leakage paths, including all auxiliary cooling systems interfacing with the primary coolant, such as heat exchangers or steam generators in PWR plants. The reviewer determines that intersystem leakage monitors are sensitive to the radiation emitted by fission products such as I-131 and the radioactive isotopes of xenon and krypton. The reviewer assures that the monitoring systems and procedures to detect and control leakage are capable of keeping intersystem leakage within the limits assumed in accident analyses. For example, steam generator tube and tube sheet leaks should be detected and corrective action taken before the contamination of the secondary coolant exceeds that assumed in evaluating the steam generator tube accident without offsite power (see SRP 15.6.3).

5. System Sensitivity and Response Time

The reviewer determines that all components for the detection of unidentified leakage called for by Acceptance Criterion II.3 meet the sensitivity and response time of Acceptance Criterion II.5. Currently used systems that have been found acceptable measure leakage either directly in gpm, such as sump monitors and containment air cooler condensate monitors or indirectly in units of radiation, μCi/cc, in the containment atmosphere.

The two most-used radiation sensitive monitors are the air particulate monitor (APM) and the radiogas monitor (RGM). The threshhold sensitivity of the APM is 10^{-9} μCi/cc of containment volume and the RGM can sense 10^{-6} μCi/cc. The APM is 1000 times more sensitive than the RGM, hence its selection in Regulatory Guide 1.45 as one of the two systems a plant should have without any alternate choice. Background activity levels corresponding to assumed normal conditions of primary coolant leakage and failed fuel fraction may be used to evaluate the compliance to Acceptance Criterion II.5.

6. Seismic Capability of Systems

The SAR should state that the leakage detection systems meet the seismic capability recommendations of Acceptance Criterion II.6. The reviewer verifies that the leakage detection systems will remain functional for all seismic events not requiring a shutdown. In addition, the reviewer verifies that the APM can function after the safe shutdown earthquake. The reviewer determines that the applicant has provided the capability to take grab samples of the containment atmosphere on a periodic basis and manually analyze these samples in his radiochemistry laboratory for particulate activity and to correlate the data to primary system leakage.

7. Indicators and Alarms

Information concerning the indicators and alarms is reviewed for compliance with Acceptance Criterion II.7. The reviewer verifies that all of the leakage detection systems have readouts in the control room and are provided with alarms. Direct reading systems, such as sumps, will normally indicate in gpm. The indirect reading systems, such as the APM, will indicate in counts per minute. The reviewer determines that control room operators will have a chart or graph that permits rapid conversion of count rate into gpm, that the conversion procedures take into account the isotope being monitored and the activity of the primary coolant, and that the plant will maintain a running record of background leakage, so that its effect may be subtracted from any sudden increases in leak indication, which may be "unidentified" leakage and require prompt action. If monitoring is computerized, backup procedures should be available to the operator.

8. Testing

Information concerning operability testing and system calibration during plant operation is reviewed for compliance with Acceptance Criterion II.8. The reviewer determines that the radiation monitoring systems have a radioactive source built into the system (the SAR refers to this feature as a "check source") to permit system test and calibration during operation. He also determines that the flow of "identified" leakage, which may amount to as little as 0.05 gpm or as much as 0.25 gpm, representing a total daily flow of between 72 and 360 gallons, will be used to provide an operability check during operation for the sump monitoring systems and the containment air cooler condensate flow monitors. The directly measured quantity of flow thus obtained from the sump and air cooler monitors can be used to calibrate the radiation monitoring systems.

9. Technical Specifications

The technical specifications are reviewed for compliance with Acceptance Criterion II.9. The reviewer compares the proposed technical specification limits for unidentified and total allowable leakage to the design basis as determined in the review of items 1-5 and 7. In addition, the reviewer determines that the availability of various components of the leakage detection system and the action to be taken if a component becomes inoperative are addressed in the technical specifications. The availability of the leakage detection components has to be reviewed on a case-by-case basis due to the large number of possible component combinations, multiplicity of systems, use or lack of redundancy, or the ability to use systems not specifically called out as "leakage detection systems" but still able to perform this role as a secondary function to the primary design use of such system. An example would be containment vent radiation monitors. A suggested technical specification regarding availability is as follows:

> "Both the sump monitoring and air particulate monitoring systems shall be operable during reactor power operation. If either system becomes inoperable for any reason, reactor power operation is permissible only during the succeeding seven days unless the system is made operable sooner. If the above conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours."

10. General

Should the leakage detection system submittal contain additional data and analyses as a basis to support a system not discussed in Regulatory Guide 1.45, the reviewer should evaluate the applicant's data and analyses to determine if the proposed system has leakage detection capabilities comparable to those of the standard systems. The reviewer can also find guidance in review procedures III.1 through 9, above, and in other SAR's where applicants may have proposed similar alternate systems.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided in accordance with the requirements of this review plan, and that his evaluation provides the basis for conclusions of the following type, which should be included in the staff's safety evaluation report:

"Coolant leakage within the primary containment may be an indication of a small through-wall flaw developed in the reactor coolant pressure boundary (RCPB).

"The leakage detection system provided will include sufficiently diverse leak detection methods, with adequate sensitivity to measure small leaks and to identify the leakage sources within practical limits, with the aid of suitable control room alarms and readouts. The major systems are the containment atmosphere particulate and radiogas monitors, and level indicators on the containment sumps. Indirect indications of leakage are obtainable from the containment pressure, humidity, and temperature indicators.

"The leakage detection systems provided to detect leakage from components of the reactor coolant pressure boundary furnish reasonable assurance that structural degradation, which may develop in pressure-retaining components of the RCPB and result in coolant leakage during service, will be detected on a timely basis, so that corrective actions can be made before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of the makeup systems to replenish the coolant loss. The systems are in compliance with the recommendations of Regulatory Guide 1.45 and satisfy the requirements of General Design Criterion 30, Appendix A of 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion No. 30, "Quality of Reactor Coolant Pressure Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3. Standard Review Plan 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment."

4. Standard Review Plan 7.5, "Safety-Related Display Instrumentation."

5. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.3.1 REACTOR VESSEL MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires that the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The following areas relating to reactor vessel materials are reviewed:

1. Materials Specifications

The material specifications used for the reactor vessel and applicable appurtenances such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings are reviewed and their adequacy for use in the construction of such components is assessed on the basis of the material, mechanical, and physical properties, the effects of irradiation on these materials, their corosion resistance, and fabricability. Similarly, the specifications for austenitic steel and nonferrous metals specified for the above applications are reviewed with respect to mechanical properties, stress-corrosion resistance, and fabricability.

2. Special Processes Used for Manufacture and Fabrication of Components

Information submitted by the applicant for any special process used in the manufacture of the product forms supplied, and their fabrication into the reactor vessel or any of its appurtenances is reviewed, and the capability of these processes to provide components with suitable mechanical and physical properties is assessed. The effects of such special processes on the stress-corrosion characteristics of the material, and any aspect of the process which could cause special requirements for nondestructive examinations are reviewed.

3. <u>Special Methods for Nondestructive Examination</u>

Nondestructive examination methods differing from those described in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III are reviewed. Attention is directed towards calibration methods, instrumentation, methods of application, sensitivity, reliability, and standards used.

4. <u>Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels</u>

Information on special controls and special processes for welding ferritic steels and austenitic stainless steels is reviewed, and their adequacy is assessed. The extent to which the controls and processes deviate from code rules is reviewed. Information on welding of safe-ends during the fabrication of dissimilar metal joints is given particular attention and details of the methods, processes, and materials used are reviewed.

5. <u>Fracture Toughness</u>

Fracture toughness of ferritic materials used for reactor vessels and appurtenances thereto is reviewed to assure that such components will behave in a nonbrittle manner and that the probability of rapidly propagating fracture will be minimized under operating, maintenance, testing, and postulated accident conditions. The review includes the descriptions of the fracture toughness tests performed on all ferritic materials used for the reactor vessel and appurtenances thereto, and includes transverse Charpy V-notch impact test specimens, dropweight test specimens, and any other test specimens included by the applicant.

The test procedures specified by the applicant are reviewed and their adequacy is confirmed.

The composition of ferritic materials employed for the reactor vessel is reviewed and the amount of residual elements such as copper, sulfur, and phosphorous is checked. The results of impact tests performed on base material, weld metal, and heat-affected zones are reviewed, and the scope of the testing is checked, particularly in the area of the reactor vessel beltline region where radiation effects on the material are most significant.

Fracture toughness of the materials employed is characterized by its reference temperature, RT_{NDT}. This temperature is the higher value of the nil-ductility temperature (NDT) from the dropweight test, or the temperature that is $60^{o}F$ below the temperature at which Charpy V-notch impact test data meet a specified toughness level. The information submitted is checked to ensure that the RT_{NDT} of the materials is included with the data and test results for impact testing.

6. <u>Materials Surveillance</u>

Reactor vessel material surveillance must be performed to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled power reactors, resulting from exposure to neutron irradiation and the thermal environment. Under the surveillance programs, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

7. Reactor Vessel Fasteners

The materials for the stud bolts, washers, and nuts, or other fasteners used to hold the reactor vessel head are reviewed to determine their adequacy. Mechanical properties, including fracture toughness, are checked to ensure that all requirements are met. Lubricants or surface treatments used are reviewed to assure that the studs will be resistant to stress-corrosion cracking under the environmental conditions during service and shutdowns. The adequacy of the destructive testing procedures used to ensure initial integrity is reviewed, along with the applicable acceptance criteria.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Material Specifications

Acceptable material specifications are those listed in the Code, Section III, Appendix I, and are presented in detail in Section II, Parts A, B, and C.

The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of Code Section III, Appendix IV-1400. These data must include information on mechanical properties, weldability, and physical changes of the material.

2. Special Processes Used for Manufacturing and Fabrication

The reactor vessel and its appurtenances are fabricated and installed in accordance with Code Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with Code Section III, Paragraph NA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.

3. Special Methods for Nondestructive Examination

The acceptance criteria for examination of the reactor vessel and its appurtenances by nondestructive examination are those specified in Code Section III, NB-5000, for normal methods of examination. When special techniques or procedures are developed, they must be equivalent or superior to the techniques described in Appendix IX-3000 of Code Section III, and must be proven so, by demonstration on the specific type of component part.

4. Special Controls and Processes for Welding Ferritic Steel Components and Austenitic Stainless Steel Components

Only those welding processes capable of producing welds in accordance with the welding procedure qualification requirements of Code Sections III and IX may be used. Any process used shall be such that the records required by NB-4320 of Section III can be made, with the exception of stud welding which is acceptable only for minor nonpressure attachments.

These requirements supplement those shown in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials," concerning welding controls for ferritic and austenitic stainless steels.

5. Fracture Toughness

The acceptance criteria for this area of review are based on General Design Criterion 31, Code Section III, and Appendix G of 10 CFR Part 50.

The reactor vessel and appurtenances thereto which are made of ferritic materials must meet the following requirements for fracture toughness during system hydrostatic tests, during conditions of normal operation, and during anticipated operational occurrences:

a. The materials shall meet the acceptance standards of Paragraph NB-2330 of the Code which states that at a temperature not greater than (T_{NDT} + 60^{o}F) each Charpy C_v specimen tested (per NB-2321.2) shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs of absorbed energy.

b. When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}. In the event that the above requirements are not met, additional C_v notch impact tests are performed (in groups of three specimens, per NB-2321.2) to determine the temperature T_{cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{cv} - 60^{o}$F. Thus the reference temperature RT_{NDT}, is the higher of T_{NDT} and (T_{cv} - 60^{o}F).

c. When a C_v impact test has not been performed at (T_{NDT} + 60^{o}F), or when the C_v impact test at (T_{NDT} + 60^{o}F) does not exhibit a minimum of 50 ft-lbs and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lbs and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v impact tests performed. In addition to the above criteria, the requirements of Section IV, A.2, 3, and 4 and IV.B of Appendix G of 10 CFR Part 50 must be met.

Standard Review Plan 5.3.2, "Pressure-Temperature Limits," discusses the requirements of Section IV. A.2, 3, and 4 of Appendix G of 10 CFR Part 50 in detail.

The acceptance criteria discussed in Section IV.B of Appendix G of 10 CFR Part 50 state that reactor vessel beltline materials shall have a minimum upper-shelf energy as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with Paragraphs NB-2322.2(4) and NB-2322.6 of the Code, Section III, of 75 ft-lbs unless it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper-shelf fracture energy are adequate.

6. Materials Surveillance

The material surveillance criteria are given in Appendix H to 10 CFR Part 50, Section II.

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that have been verified by experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1 Mev) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

Reactor vessels constructed of ferritic materials which do not meet these conditions shall have their beltline regions monitoried by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," ASTM Designation: E-185-73, except as modified by Appendix H to 10 CFR Part 50.

The surveillance program shall meet the following requirements:

a. Surveillance specimens must be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G of 10 CFR Part 50. The speciment types must comply with the requirements of Section III.A of Appendix G, except that dropweight specimens are not required.

b. Surveillance capsules containing the surveillance specimens must be located near but not attached to the inside vessel wall in the beltline region, so that the neutron flux received by the specimens is at least as high but not more than three times as high as that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. The design and location of the capsules must permit insertion of replacement capsules. Accelerated irradiation capsules, for which the calculated neutron flux will exceed three times the calculated maximum neutron flux at the inside wall of the vessel, may be used in addition to the required number of surveillance capsules specified in Section II.C.3 of Appendix H, 10 CFR Part 50.

c. The required number of capsules, which will vary from three to five depending upon the adjusted reference temperature at the end of the service lifetime of the reactor vessel, and their withdrawal schedules, must be in accordance with the requirements of Section II.C.3 of Appendix H of 10 CFR Part 50.

d. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis, depending on the degree of commonality and the predicted severity of irradiation.

7. Reactor Vessel Fasteners

The Code, Section III, Appendix G to 10 CFR Part 50, and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," define the acceptance criteria for reactor vessel studs and fasteners.

Materials for reactor vessel studs (and other fasteners) that are considered suitable are SA-540 Grades B-23 and B-24, SA-193 Grade B-7, SA-194 Grade 7, and SA-320 Grade L-43.

The fastener material should not have an ultimate tensile strength over 170 ksi, and the fracture toughness tests and acceptance levels of Paragraph IV.A.4 of Appendix G to 10 CFR Part 50 must be met.

Surface treatments, plating, or thread lubricants used must be shown to be compatible with the materials, and stable at operating temperatures.

Nondestructive examination should be performed according to Section III of the Code, Subarticle NB-2580, including the additional recommendations given in Regulatory Guide 1.65.

These are:

a. The stud bolts and nuts should be ultrasonically examined after final heat treatment and prior to threading.
b. The ultrasonic examination (paragraph NB-2584) should be conducted according to ASME Specification SA-388, "Ultrasonic Examination of Heavy Steel Forgings."
c. The calibration standard used to establish the first back reflection for the ultrasonic testing should be based on good sound representative material. To assure that the material is representative, the selection of the standard should be based on a preliminary ultrasonic examination of a number of specimens (a minimum of three per standard).
d. The magnetic particle or liquid penetrant examination (paragraph NB-2583) should be performed on the studs and nuts after final heat treatment and threading.
e. The requirements of paragraph NB-2585 should be applied to all closure stud bolts and nuts.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in Section I of this plan, the review procedure is as follows:

1. Materials Specifications

The material specifications for the reactor vessel and its appurtenances are compared with the acceptable specifications listed in the Code, Section III, Appendix I, and Section II, Parts A, B, and C.

Any materials not listed in the Code, or any deviations in a listed specification, are clearly identified, and the bases for deviation or nonconformance evaluated. A study of the suitability of the material and comparisons with precedents set in earlier cases enable the reviewer to determine the acceptability of the proposed exceptions. In those instances where the Materials Engineering Branch has taken exception to the use of a specific material, or questions certain aspects of a specification, the applicant is advised which material is not acceptable, and the reason for disapproval.

2. <u>Special Processes Used for Manufacture and Fabrication of Components</u>
Information on special processes used for manufacture and fabrication of the reactor vessel and its appurtenances are reviewed to (1) identify each special process, (2) determine whether there are any code restrictions on its use, (3) establish the adequacy of the process in providing components with suitable mechanical and physical properties, (4) establish the effects of such processes on the stress-corrosion characteristics of the material, and (5) identify whether special requirements for nondestructive examination are needed if the process is used.

Since there are no specific code requirements on the use of special processes, the suitability of a process is assessed on the basis of service experience with similar parts fabricated by the process being reviewed.

3. <u>Special Methods Used for Nondestructive Examination</u>
Section V of the Code includes methods for performing nondestructive examinations to detect surface and internal discontinuities when these methods are referenced by Section III of the Code. They include the following methods: radiographic, magnetic particle, liquid penetrant, and ultrasonic. The methods as described are applicable to most geometric configurations and materials encountered in fabrication, and are applied for normal conditions. However, special configurations and materials may require modified methods and techniques. If such special procedures are developed, the reviewer must determine that they are equivalent or superior to the techniques described in Section V of the Code, and are capable of producing meaningful results under the special conditions.

Such special procedures may be modifications or combinations of methods described in Section V, or may be entirely different, but the reviewer verifies that they have been proven by demonstration to result in an examination capable of detecting discontinuities under the special conditions to the same extent that applicable normal techniques which are included in Section V would result in detection of discontinuities under normal conditions.

Such special procedures are submitted to the authorized inspector or inspecting agency for review and approval prior to use.

4. <u>Special Controls and Processes for Ferritic Steels and Austenitic Stainless Steel</u>
The controls on welding of ferritic steels and austenitic stainless steels discussed in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials," are considered applicable to welding of the reactor vessel and its components. The reviewer verifies that any special welding control or special welding process is able to conform to the qualification requirements of the Code, Section IX, or that justification is made for this deviation.

In the event that this information is lacking in the SAR, the reviewer prepares a request for additional information, for transmittal to Reactor Projects. Such a request not only identifies the additional information required, but also lists the changes needed in the SAR. Subsequent amendments received in response to these requests are reviewed for compliance with the stated criteria.

5. Fracture Toughness

The information submitted by the applicant relative to tests for fracture thoughness is reviewed for conformance with the Code, Section III, Paragraph NB-2320, and Appendix G of 10 CFR Part 50.

These tests include Charpy V-notch impact tests and dropweight tests. A description of the tests is reviewed, and the location of the test specimens and their orientation are verified.

Information regarding calibration of instruments and equipment are reviewed for conformance to Code Section III, Paragraph NB-2360.

In the event that none of the fracture toughness tests have been performed, the preliminary safety analysis report (PSAR) must contain a statement of the applicant's intention to perform this work in accordance with Code Section III, NB-2300 and Appendix G of 10 CFR Part 50.

The final safety analysis report (FSAR) is reviewed to assure that all the impact tests shown in NB-2320 have been performed. The results of the tests shall be in accordance with the acceptance criteria shown in II.5 of this plan.

If the information contained in the SAR does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, the reviewer prepares a request for additional information for transmittal to Reactor Projects. Such requests not only identify the additional information required, but also specify the changes needed in the SAR or the plant Technical Specifications to meet acceptance criteria. Subsequent amendments received in response to these requests are reviewed for compliance with the stated criteria.

6. Materials Surveillance

The reviewer verifies that the information contained in the SAR and the Technical Specifications is complete enough to determine that the surveillance program will comply with Appendix H, 10 CFR Part 50. The following information must be provided as a minimum.

a. The reviewer verifies that the PSAR states the end of life fluence calculated for the vessel beltline, the maximum predicted shift in reference transition temperature (RT_{NDT}), the number of capsules, and the number and types of specimens to be placed in the capsules, and that the program is in compliance with ASTM E 185-73 and Appendix H, 10 CFR Part 50.
b. The reviewer verifies that the FSAR provides the information listed above, and in addition, includes results of fracture toughness tests and chemical analyses of all materials in the beltline region, and provides the information needed by the reviewer to evaluate the adequacy of the program.

7. Reactor Vessel Fasteners

The reviewer verifies that the information in the SAR covers all requirements for reactor vessel studs and other fasteners, as described in the previous section. For FSAR's, the results of tensile and fracture toughness tests performed on the fastener materials are checked to ensure that all requirements are met.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the review plan, and that his evaluation supports the conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of the applicant with regard to control of residual elements in ferritic materials have been identified and are considered acceptable.

"Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code have been submitted by the applicant. Since certification has been made by the applicant that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes used are considered acceptable.

"Special methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified, and have been found equivalent or superior to the techniques described in Appendix X of Code Section III. Demonstrations have been made using these special techniques, and have satisfied all requirements of the Code. The special methods of nondestructive examination are deemed acceptable.

"Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of Code Sections III and IX. The controls imposed on welding preheat temperatures are in conformance with the requirements of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Allow Steels," and provide reasonable reassurance that cracking of components made from low alloy steels will not occur during fabrication, and will minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. The controls imposed on electroslag welding of ferritic steels are in accordance with the requirements of Regulatory Guide No. 1.34, "Control of Electroslag Weld Properties," and provides assurance that welds fabricated by the process will have high integrity, and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance, and postulated accident conditions. The controls imposed upon austenitic stainless steel welds are in conformance with Regulatory Guide No. 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

"The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G of the Code as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the Code and AEC Regulations, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC Regulations constitutes an acceptable basis for satisfying the requirements of General Design Criterion 31.

"Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation have been assessed properly, and adequate safety margins against the possibility of vessel failure are provided as the material surveillance requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50 are met. Compliance with these documents assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of General Design Criterion 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section II, Section III, Section V, and Section IX, American Society of Mechanical Engineers.

5. ASTM E-23-72, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

6. ASTM E-185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials.

7. ASTM E-208-69, "Standard Method for Conducting Drop-weight Test to Determine Nil-ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

8. Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

9. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

10. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

11. Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

12. Regulatory Guide 1.31, "Control of Stainless Steel Welding."

13. Branch Technical Position MTEB 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended to Standard Review Plan 5.2.3.

11/24/75

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.3.2 PRESSURE-TEMPERATURE LIMITS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

1. Pressure-Temperature Limits

The following pressure-temperature limits imposed on the reactor coolant pressure boundary (RCPB) during operation and tests are reviewed to assure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components of the RCPB, as required by General Design Criterion No. 31:

a. Pressure-temperature limits for preservice hydrostatic tests.
b. Pressure-temperature limits for inservice leak and hydrostatic tests.
c. Pressure-temperature limits for heatup and cooldown operations.
d. Pressure-temperature limits for core operation.

II. ACCEPTANCE CRITERIA

1. Applicable Regulations, Codes, and Basis Documents

Appendices G and H of 10 CFR Part 50 describe the conditions that require pressure-temperature limits and provide the general basis for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Appendix G, "Protection Against Non-ductile Failure," during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins whenever the reactor core is critical (except for low-level physics tests).

The Code, Section III, specifies fracture toughness testing requirements for ferritic materials. Appendix G of the Code provides a basis for determining allowable pressure-temperature relationships for normal, upset, and test conditions.

Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials," provides the detailed technical basis for the code requirements.

2. Technical Bases

a. The principles of linear elastic fracture mechanics (LEFM) are used to determine safe operational conditions. The basic parameter of LEFM is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. An analytical method is used to determine the effects of real or postulated flaws. The minimum K_I that can cause failure is defined as the critical stress intensity factor, K_{Ic}, and is the material property used in this method. The K_{Ic} of the material is either directly measured as a function of temperature, or is conservatively estimated, using information from other fracture toughness tests.

b. The Code specifies the maximum K_{Ic}, as a function of temperature, that can be assumed for the specific material, based on results of tests on the material used. This value is called K_{IR}, reference stress intensity factor. The Code also provides rules for calculating the K_I, including definitions of postulated flaws, and specifies the safety factors to be applied. The acceptance criterion is that the K_{IR} of the material must always be higher than the K_I calculated.

c. Direct measurement of the K_{Ic} as a function of temperature is expensive and time consuming and requires more sample material than is usually available. Correlations between the K_{Ic} determined directly and results of simpler fracture toughness tests are not exact, but may be used if appropriate allowances are made for variations in material behavior and data scatter. The Code gives values of K_{IR} as a function of temperature relative to a conservative determination of the nil-ductility transition temperature (NDTT) of the material. This reference temperature, RT_{NDT}, is determined for the ferritic materials of components for which operating and testing limit curves must be calculated. The effects of radiation on the fracture toughness of the material in the beltline region of the reactor vessel is accounted for by adjusting the RT_{NDT} of the affected material upward. The amount of upward shift depends on the composition of the steel (especially its copper and phosphorous content), the neutron fluence, and the temperature of irradiation. Conservative predictions of the effect of radiation on the RT_{NDT} based on data in the literature are factored into the original limit curves. The continued conservatism of these predictions throughout plant life is verified by a mandatory material surveillance program described in Appendix H to 10 CFR Part 50.

d. The Code specifies the stress components that must be used for the K_I calculations, and the factors that must be applied to each to provide adequate safety margins. The Code, by reference to WRC-175, specifies the expression to use for calculating the K_I, using the applied stresses and the postulated flaw geometry. Although calculations are usually made by a computer, curves are provided in the Code to facilitate the use of conservative hand calculations if desired.

3. Pressure-Temperature Requirements

The requirements for the pressure-temperature limits are as follows:

a. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), the K_{IR} must be greater than the K_I caused by pressure. The expression used is:

$$K_I = K_I(\text{pressure}) < K_{IR}$$

b. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests

During performance of inservice leak and hydrostatic tests, the K_{IR} must be greater than 1.5 times the K_I caused by pressure. The expression used is:

$$K_I = 1.5\, K_I(\text{pressure}) < K_{IR}$$

c. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, the K_{IR} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients. The expression used is:

$$K_I = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{IR}$$

d. Pressure-Temperature Limits for Core Operation

At all times that the reactor core is critical (except for low power physics tests) the temperature must be higher than that required for inservice hydrostatic testing, and in addition, the pressure-temperature relationship shall provide at least a 40°F margin over that required for heatup and cooldown operations.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. Preliminary Safety Analysis Report (PSAR)

Actual operating limit curves cannot be determined at the PSAR stage, because the fracture toughness and other required tests have not been performed on the actual material that will be used. Typical curves, with temperatures shown relative to the RT_{NDT}, and the basis for determining the curves are reviewed and compared with the acceptance criteria described in II above.

2. Final Safety Analysis Report (FSAR)

The limits in the plant Technical Specifications will be shown using real temperature. These curves and their bases are reviewed to determine acceptability in the following areas:

a. The limiting RT_{NDT} has been properly determined, and radiation effects are included in a conservative manner.
b. Limits are shown for all required conditions.
c. The limits proposed are consistent with the acceptance criteria described in II above.
d. The procedures for updating the limit curves, in conjunction with scheduled tests on material surveillance specimens, are well defined and included in the Technical Specifications.

3. Acceptability Determination Methods

The reviewer evaluates each limit curve for acceptability by performing check calculations using the simplified methods referenced in the Code and WRC Bulletin 175 that have been verified by the Materials Engineering Branch to yield conservative values. These methods are described in detail by examples below, and the curves necessary to perform the calculations are included herein as Figures 1, 2 and 3. Figure 4 is an example of acceptable limit curves developed by this method.

a. Preservice Hydrostatic Tests

The preservice hydrotest at 1.25 design pressure corresponds to the standard Code component hydrotest usually performed in the shop, but in this case it is the hydrotest for field welds, so it may involve the entire reactor coolant system.

The Code recommends that component hydrostatic tests be run at a temperature no lower than RT_{NDT} + 60°F, but also recommends that system tests should have more stringent requirements. The MTEB position is that the minimum temperature for the preservice test, if fuel is not in the vessel, be determined using the methods of Code Section III, Appendix G, using less stringent factors.

First, the RT_{NDT} of the vessel material must be determined. This is defined by the Code for new plants, and is essentially a conservative value of the NDTT as determined by drop weight test. Guidelines for estimating the RT_{NDT} if the prescribed tests have not been run are covered by Branch Technical Position - MTEB No. 5-2, "Fracture Toughness Requirements."

The reference temperature and the toughness of the material at any temperature is a function of the difference between the RT_{NDT} of the material and the temperature of interest. The Code provides a curve (Figure G-2110.1) for the allowable calculated stress intensity factor (K_{IR}) as a function of the temperature relative to RT_{NDT}.

The Code also provides a recommended basis for calculating K_I, including recommendations for assumed flaw size and shape, and appropriate front and back surface correction factors. Because the assumed flaw size is proportional to the wall thickness, t (flaw depth = 0.25 t and length = 1.5 t), the K_I expressions are simplified to multiplers that are a function only of wall thickness and stress level. These factors, M_m for membrane stresses and M_B for bending stresses, are provided in graphical form in Figure G-2114.1.

The criterion recommended by MTEB can be expressed as

$K_I < K_{IR}$ for the shell region.

To get K_I, the stress level and wall thickness must be known. The pressure for the hydrostatic test is 1.25 times the design pressure, so either of two simple methods can be used to approximate the membrane stress accurately enough for this purpose:

stress = 1.25 times the Code allowable (S_m)

stress = $\frac{Pr}{t}$

where P is the test pressure and r is the vessel radius. As an example, assume a vessel with a design pressure of 2500 psig, made of steel with an S_m

of 26,700 psi, and a minimum yield strength of 50,000 psi. The stress for the preservice hydrotest is then

$26{,}700 \times 1.25 = 33{,}400$ psi, or

$\frac{(1.25)\ (2500)\ (95)}{9} = 33{,}400$ psi, for a vessel with a radius of 95 inches and a wall thickness of 9 inches.

The next step is to determine the factor to apply to this stress to obtain K_I. Figure G-2114.1 (reproduced here as Fig. 1) provides several curves, depending on the ratio of the stress level to the yield strength of the material. In this case, the stress level is 33,400; the yield strength is conservatively assumed to be 50,000 so the curve for a ratio of .7 should be used. (A ratio equal to or higher than the actual ratio must be used for conservatism.) For a 9-in. thick vessel ($\sqrt{t} = 3$), the value of M_m from Figure G-2114.1 is 2.94. The K_I for this case is then:

$K_I = (M_m)$ (Membrane Stress)

$K_I = (2.94)\ (33{,}400) = 98{,}300$ psi $\sqrt{in.}$

From Figure G-2110.1 (reproduced here as Fig. 2), a temperature of at least RT_{NDT} + 120°F is necessary for a K_I of this level.

If, for example, an original RT_{NDT} of 40°F is assumed, the required temperature is then 40 + 120, or 160°F.

b. Inservice Leak and Hydrotest

The temperatures for the inservice leak and hydrotest, performed at operating pressure and about 1.1 operating pressure, respectively, are calculated in essentially the same way. The differences are that a factor of 1.5 must be applied to the calculated K_I to provide extra margin, and the stress levels are lower, so the value of M_m is taken from a lower ratio curve.

Using the same vessel as an example, with a normal operating pressure (P_o) of 2250 psi, the membrane stress for the leak test can be approximated as:

$\frac{\text{operating pressure}}{\text{design pressure}} \times$ allowable stress

or $\frac{2250}{2500} \times 26{,}700 = 24{,}000$ psi

This is about half of the minimum yield strength, so the M_m is taken from the 0.5 ratio curve, and is 2.87. The calculated K_I that must be assumed is then:

$K_I = (1.5)\ (M_m)$ (Membrane Stress)

or $K_I = (1.5)\ (2.87)\ (24{,}000) = 103{,}500$ psi $\sqrt{in.}$

From the K_{IR} curve, a temperature of about RT_{NDT} + 125°F is required. As this is an inservice test, the RT_{NDT} would probably have been increased from its original value of + 40°F by some shift caused by radiation. Assume this shift is 100°F, thus the temperature for the leak test must be at least:

40 + 100 + 125 = 265°F

The inservice hydrotest temperature (at 1.1 P_o) is determined in exactly the same way, and requires a minimum temperature of about RT_{NDT} + 133°F, or 273°F.

c. Heatup, Cooldown, and Normal Operation

For normal operation, which includes upset conditions and startup and shutdown procedures, operating limit curves must be provided that show the maximum permissible pressure at any temperature from cold shutdown conditions to full pressurization conditions.

Reactor vendors have developed computer codes to perform the necessary calculations, because thermal stresses must be included, and hand calculations of even moderate sophistication are very time consuming. WRC Bulletin 175 includes a set of curves derived from computer programs that can be used to approximate the K_I caused by thermal stresses, as a function of wall thickness and rate of temperature change. Pressure-temperature curves developed using these approximations agree fairly well with those determined using much more rigorous procedures, and can be used with confidence to evaluate the proposed operating limits given in Technical Specifications. These curves require the calculation of only 3 to 5 points. Either allowable pressure at a given temperature, or allowable temperature at a given pressure can be calculated. It is usually more convenient to calculate allowable minimum temperature, so this method will be used in the example.

Using the same reactor vessel as in the previous example, and a rate of temperature change of 50°F per hour, calculations of required temperatures for several pressures are illustrated. The curves for thermal effects given in WRC Bulletin 175 are very conservative, thus no additional margin need be applied to the K_I from thermal stress, but a factor of 2.0 is used on primary stresses. The basic expression is then:

$$K_{IR} \geq 2\ K_I(\text{membrane}) + K_I\ (\text{thermal})$$

K_I(membrane) is calculated exactly as in the previous examples. K_I(thermal) for a 9-in. thick wall, at 50°/hr is about 12,000 psi $\sqrt{\text{in.}}$ from Figure 4-5, WRC Bulletin 175 (reproduced here as Fig. 3).

Thus, for a pressure of 2250 psig, a membrane stress of 24,000 psi, and M_m of 2.87, the basic expression is given by

$$K_{IR} > (2)(24{,}000)(2.87) + 12{,}000 = 150{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, a temperature of RT_{NDT} + 158°F is required. With an RT_{NDT} of 140°F, the temperature required for operating pressure at a heatup or cooldown rate of 50°/hr is then

$$140 + 158 = 298°F$$

For a pressure of 1/2 of operating (1125 psig), the membrane stress is 1/2 of that at operating pressure, or 12,000 psi.

The M_m can be taken from the 0.5 $\frac{\sigma}{\sigma y}$ ratio curve in Figure 2114.1, so is again 2.87.

$$K_{IR} \geq (2)(12{,}000)(2.87) + 12{,}000 = 81{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the K_{IR} curve, the minimum temperature is RT_{NDT} + 100°F, or 140 + 100 = 240°F.

The same calculation for a pressure of 1/5 operating pressure (450 psig and 4800 psi stress) is similar, but in this case the stress is less than .1 of the yield strength, so the M_m (from the .1 ratio curve) is only 2.82.

$$K_{IR} \geq (2)(4800)(2.82) + 12{,}000 = 39{,}000 \text{ psi } \sqrt{\text{in.}}$$

The K_{IR} curve shows that the minimum temperature is RT_{NDT} + 0°F, or 140°F.

Three points on a 50°/hr operating limit curve for this vessel at this time in its service lifetime have thus been calculated:

Pressure (psig)	Min. Temperature (Fahrenheit)
450	140
1150	240
2250	298

A smooth curve drawn through these points will very closely approximate the results using more rigorous methods.

d. Core Operation

Appendix G, 10 CFR Part 50, specifies pressure-temperature limits for core operation to provide additional margin during actual power production.

The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in the preceding section. The minimum temperature for the inservice hydrostatic test for the vessel used in the preceding example was 273°F. A vertical line at 273°F on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve as determined in the preceding section, constitutes the limit for core operation for this example.

The information required to evaluate the adequacy of the temperature limits can all be put on one figure. The temperature limits calculated in the preceding sections along with material data used are shown in Figure 4.

e. Acceptable limit curves for several typical plants have been developed to facilitate the evaluation of acceptability of proposed limits. These are included as Figures 5 and 6 of this review plan. These are based on the simplified procedures described above, and are slightly more conservative than curves developed by more rigorous computer calculations. Curves presented in plant Technical Specifications are considered acceptable if they are as conservative as these reference limit curves. If they are based on more rigorous analytical methods, as recommended by the Code or WRC Bulletin 175, they will be considered acceptable if the variation from the reference limit curves is not more than 10°F.

If the proposed limit curves are more than 10°F less conservative than the reference limit curves, detailed bases and calculations must be submitted by the applicant for review. To be acceptable, all bases and analytical expressions used must be in accordance with Appendix G, 10 CFR Part 50, and the proposed curves must agree with check calculations made by the Materials Engineering Branch using these bases and expressions.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and that the completeness and technical adequacy of his evaluation will support the following kind of concluding statement, to be included in the staff's safety evaluation report:

"The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against nonductile or rapidly propagating failure are in conformance with established criteria, codes, and standards acceptable to the Regulatory staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

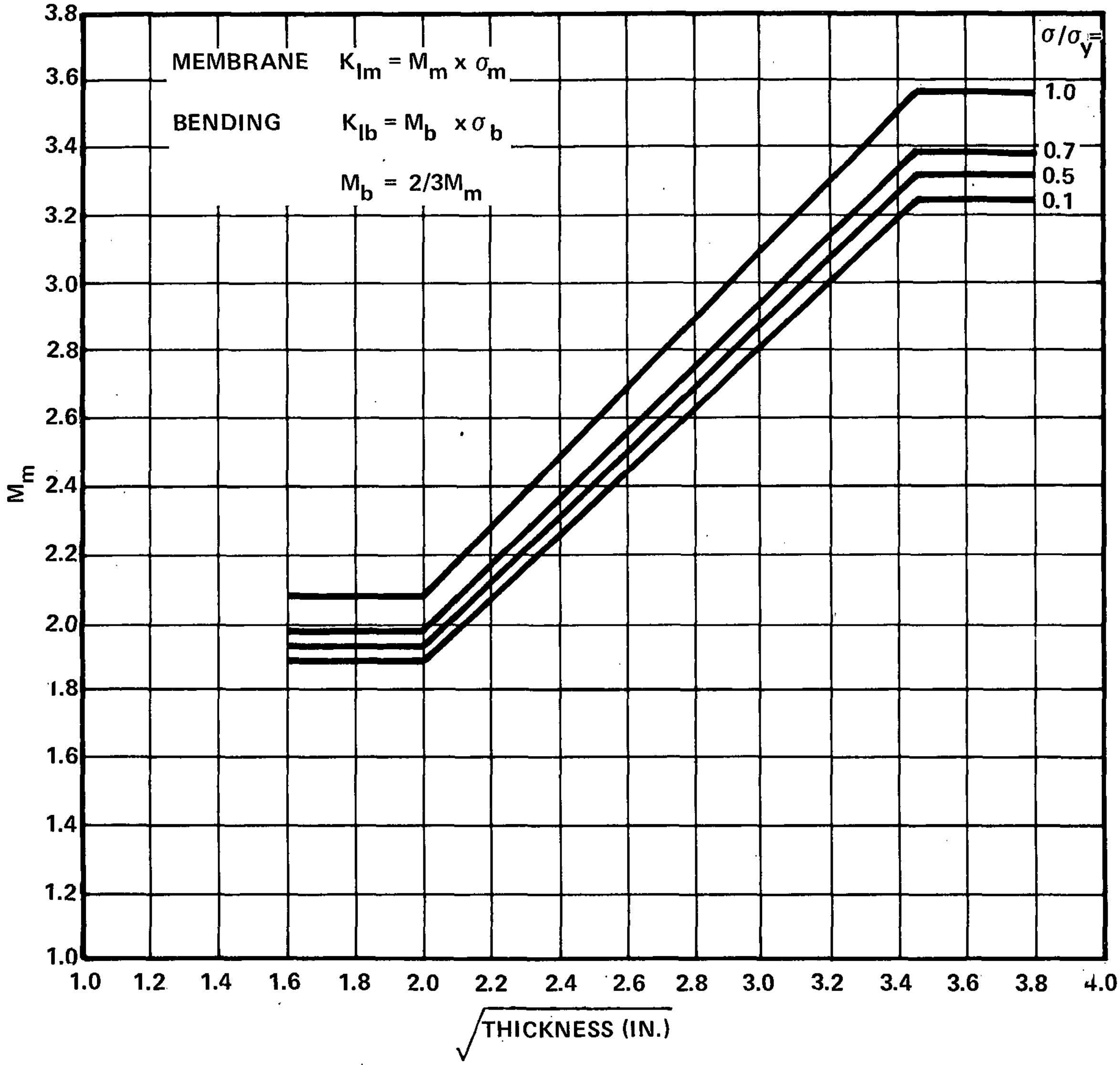
2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

5. WRC Bulletin 175, "PVRC Recommendation on Fracture Toughness," Welding Research Council.

6. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants," appended.



M_m AND M_b VS. WALL THICKNESS FOR

SEMI-ELLIPTICAL SURFACE FLAW ¼T DEEP AND 1½T LONG

FIGURE 1

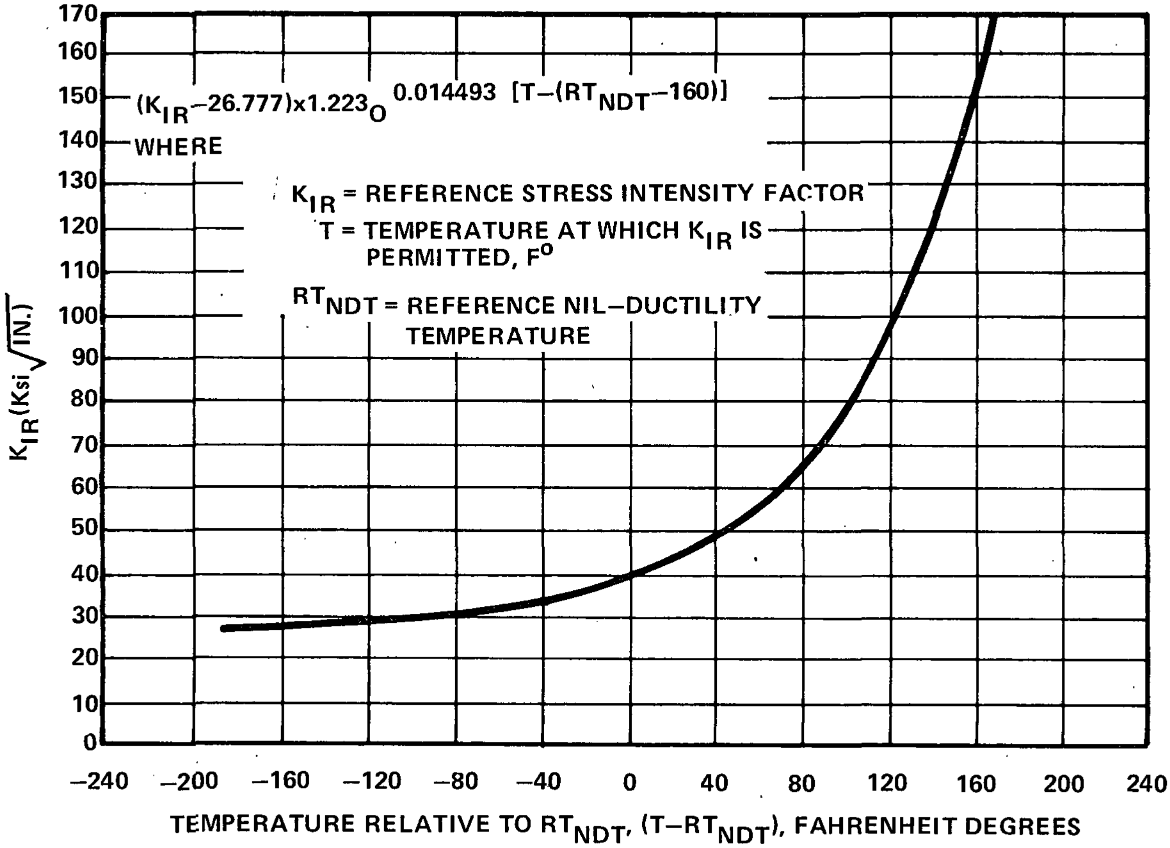
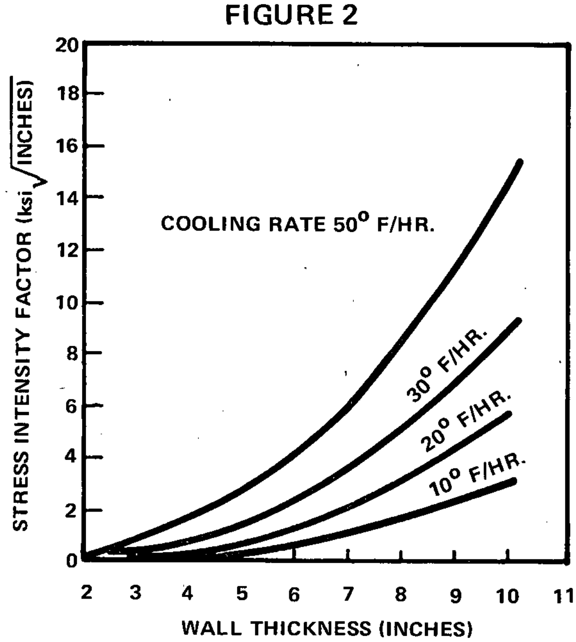


FIGURE 2



4-5 – STRESS INTENSITY FACTOR CAUSED BY THERMAL STRESS FOR CYLINDERS WITH RADIUS/THICKNESS = 10

FIGURE 3

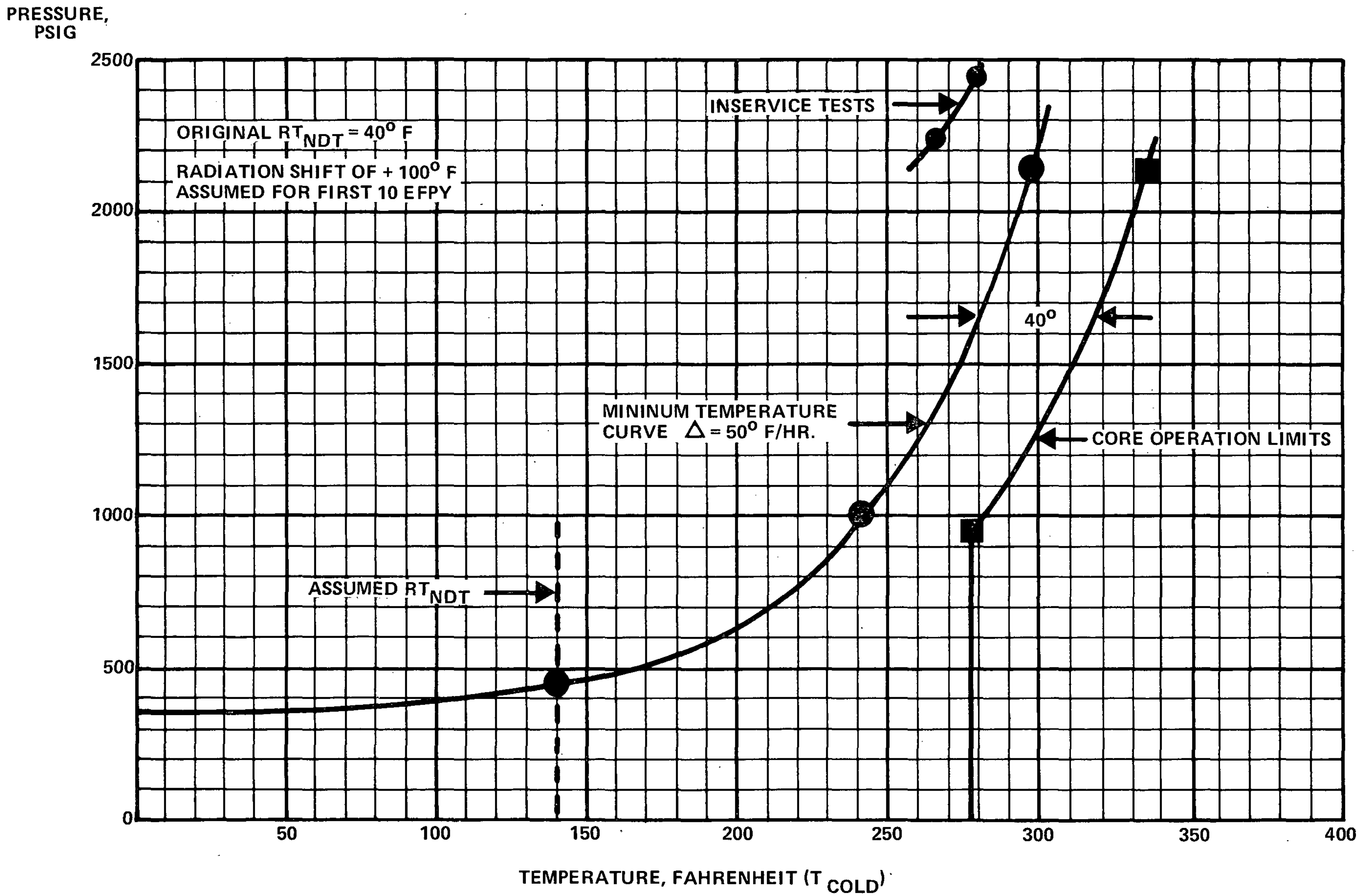


FIGURE 4 TYPICAL PWR REACTOR VESSEL TEMPERATURE LIMITS

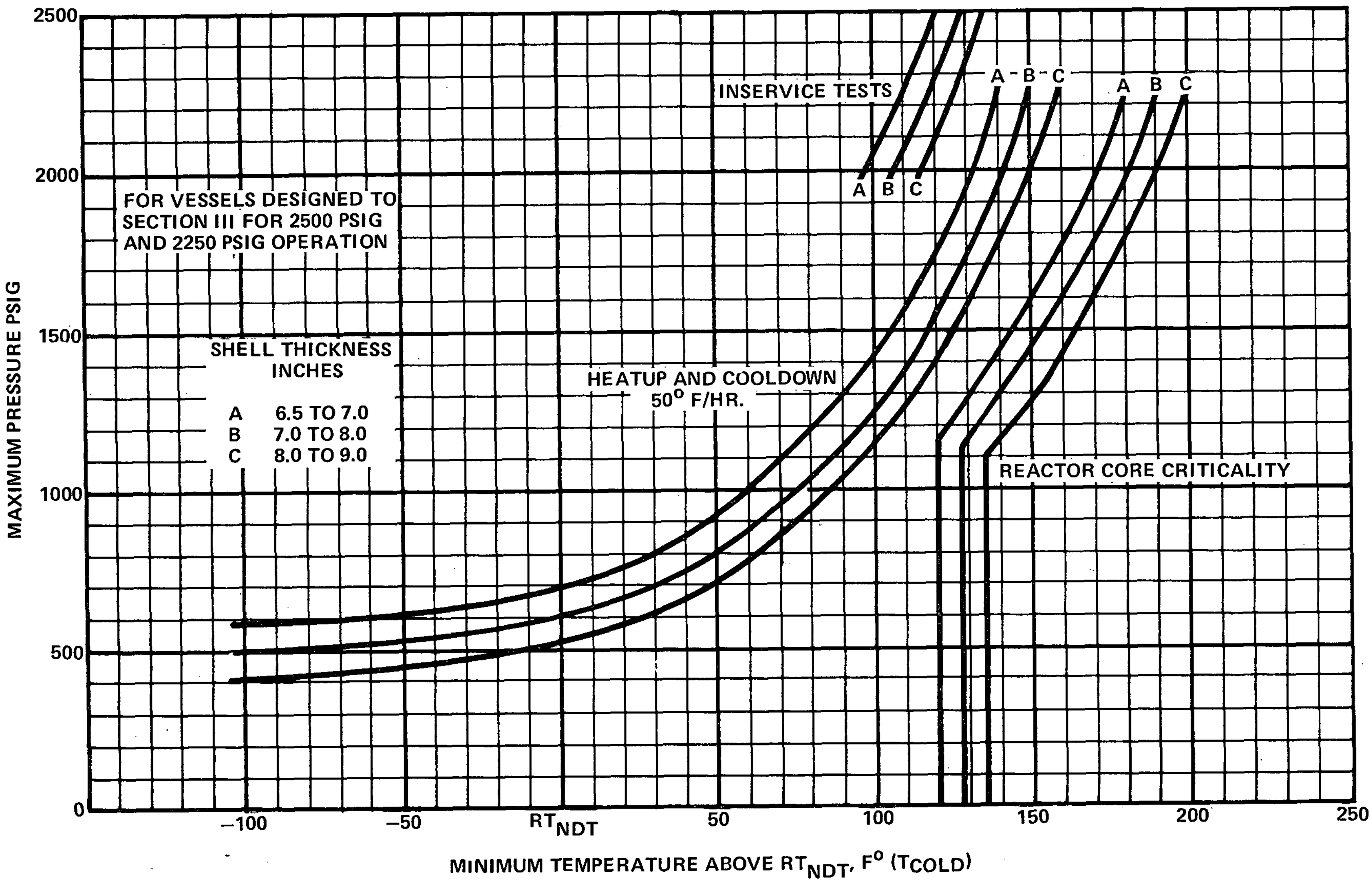


FIGURE 5 PWR REFERENCE LIMIT CURVES

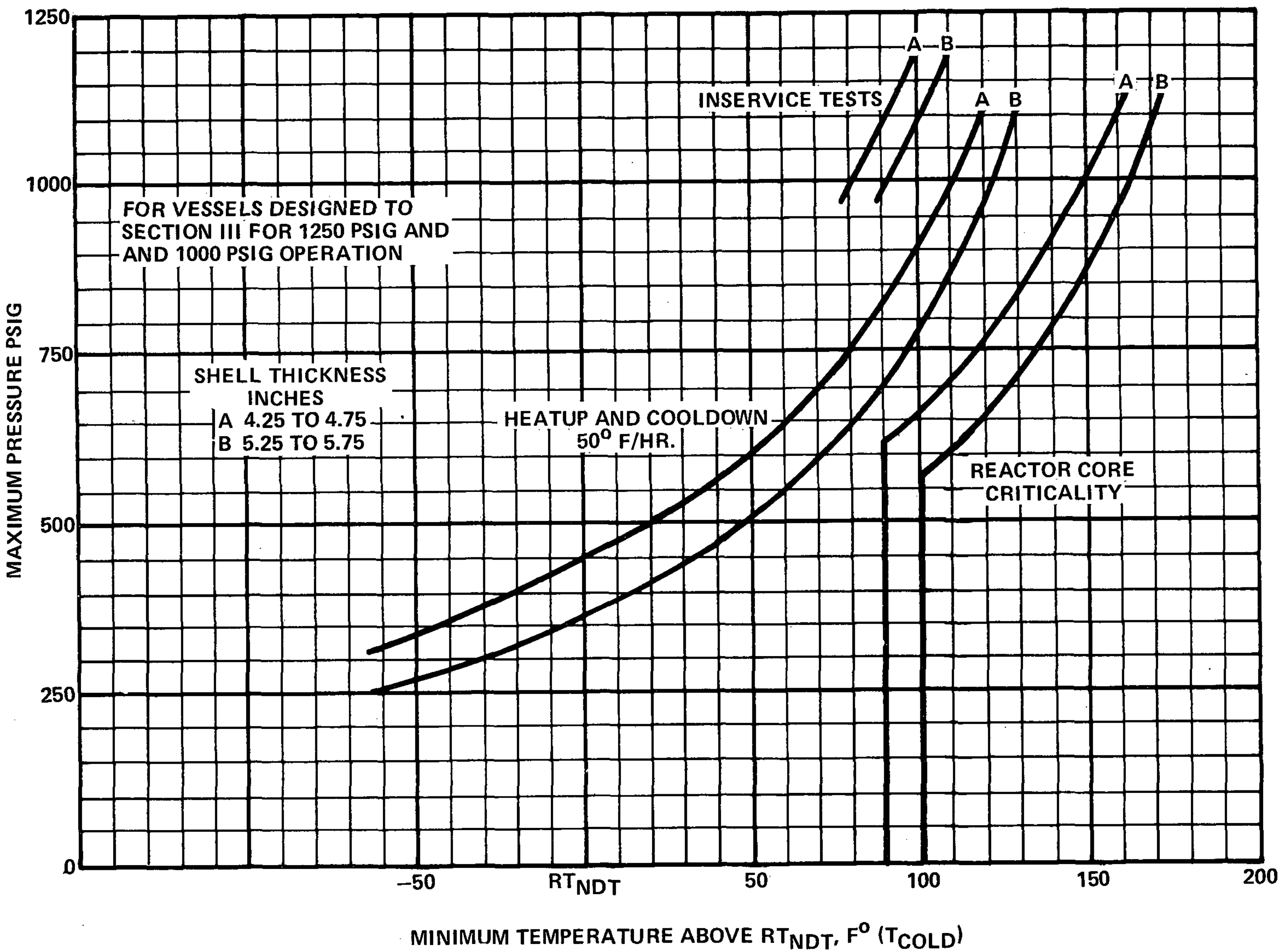


FIGURE 6 BWR REFERENCE LIMIT CURVES

11/24/75

BRANCH TECHNICAL POSITION - MTEB NO. 5-2

FRACTURE TOUGHNESS REQUIREMENTS

A. Background

Current requirements regarding fracture toughness, pressure-temperature limits, and material surveillance are covered by the ASME Code and Appendices A, G, and H to 10 CFR Part 50. The purpose of this branch technical position is to summarize these requirements and provide clarification, as necessary.

Since many of these requirements were not in force when the plant was designed and built, this position also provides guidance for applying these requirements to older plants. Also included is a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with the new requirements.

B. Branch Technical Position

1. Preservice Fracture Toughness Test Requirements.

The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Section III of the ASME Code, as augmented by Appendix G, 10 CFR 50. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in all respects. The fracture toughness of the materials for these plants must be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This must be done because the operating limitations imposed on old plants must provide the same safety margins as are required for new plants.

1.1 Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from results of fracture toughness tests. Both drop weight NDTT tests and Charpy V-notch tests must be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208) is the RT_{NDT} if, at 60°F above the NDTT, at least 50 ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V tests on specimens oriented in the weak direction (traverse to the direction of maximum working).

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests required to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided for guidance.

(1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 30 ft-lbs was obtained in Charpy V-notch tests, or 0°F, whichever was higher.

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:

(a) 60°F.

(b) The temperatures of the Charpy V-notch upper shelf.

(c) The temperature at which 100 ft-lbs was obtained on Charpy V-notch tests if the upper-shelf energy values were above 100 ft-lbs.

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 50 ft-lbs and 35 mils LE would have been obtained on traverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.

(b) Temperatures at which 50 ft-lbs and 35 mils LE were obtained on longitudinally-oriented specimens increased 20°F to provide a conservative estimate of the temperature that would have been required to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 30 ft-lbs was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 45 ft-lbs was obtained if the specimens were longitudinally-oriented. If the minimum value obtained was less than 45 ft-lbs, the RT_{NDT} may be estimated as 20°F above the test temperature.

1.2 Estimation of Charpy V Upper-Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must be adequate to accommodate degradation by neutron radiation. The original minimum shelf energy must be 75 ft-lbs for vessels with an estimated end of life neutron fluence (> 1 MeV) of 1×10^{19} and over. A value of 70 ft-lbs is considered adequate for material for vessels that will be subjected to lower fluences.

If upper-shelf Charpy energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

1.3 Reporting Requirements

Fracture toughness information required by the Code and by Appendix G, 10 CFR 50, must be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.

2. Operating Limitations for Fracture Toughness

2.1 Required Pressure-Temperature Operating Limitations

As required by Appendix G, 10 CFR 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR 50, and Appendix G to Section III of the Code.

(1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system.

(2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service.

(3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions.

(4) Maximum pressure-minimum temperature curves for core operation.

2.2 Recommended Bases for Operating Limitations

2.2.1 Leak and Hydrostatic Tests

(1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel.

(2) Tests at pressures less than design pressure should be conducted at temperatures calculated according to Appendix G of Section III of the Code for the beltline region (including conservative estimates of radiation damage, see Section 3.0 below) if the maximum calculated primary stress in no other region of the vessel exceeds 1.25 S_m during the test, and the RT_{NDT} of the beltline is assumed to be at least 30°F above that of the higher stressed regions. If primary stresses are calculated to be over 1.25 S_m in any region during the test, the RT_{NDT} of the vessel must be assumed to be at least 50°F higher than that of any region where the calculated primary stresses are over 1.25 S_m.

(3) Alternatively, a fracture mechanics analysis, with technical justification for all assumptions and bases, may be made to determine the minimum test temperature. In no event shall the minimum temperature be lower than that resulting from calculations for the beltline region in accordance with Appendix G of the Code.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using simple $\frac{pr}{t}$ stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from Figure 4-5 in WRC Bulletin 175.

Calculations need only be performed for the beltline region, if the assumed RT_{NDT} of the beltline is at least 50°F above the RT_{NDT} for all higher stressed regions.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.

2.2.3 Core Operation Limits

To provide added margins during actual core operation, Appendix G, 10 CFR 50 requires a minimum temperature during core operation, and a 40°F margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above.

2.2.4 Upset Conditions

The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures must permit variations from intended operation, including all upset conditions, without exceeding the limit curves.

2.2.5 Emergency and Faulted Conditions

It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator.

For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications.

The SAR must present evaluations of the continued integrity of all vital components during postulated faulted conditions. It is recommended that such evaluations be made in a realistic manner, avoiding grossly overconservative assumptions and procedures, and clearly show that margins against loss of integrity are calculable and adequate.

2.3 Reporting Requirements

The Technical Specifications must include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications must also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.

3. Inservice Surveillance of Fracture Toughness

The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR 50, must be carried out.

3.1 Surveillance Program Requirements

The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR 50. It is strongly recommended that consideration by given to the desirability of additional surveillance methods, such as the inclusion of CT, DWT, DT, or other specimens to provide the capability of redundant test methods and analytical procedures, particularly if the estimated neutron fluence is over 2×10^{19}, or the toughness of the vessel material is marginal.

The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-73, unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RT_{NDT} and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RT_{NDT}, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and phosphorous content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Requirements

The adequacy of the surveillance program cannot be evaluated unless all pertinent information is included in the SAR. Information requested for beltline materials includes the following:

(1) Tensile properties.
(2) DWT and Charpy V test results used to determine RT_{NDT}.
(3) Charpy V test results to determine the upper shelf toughness.
(4) Composition, specifically the copper and phosphorous content.
(5) Estimated maximum fluence for each beltline material.
(6) List of materials included in the surveillance program, with basis used for their selection.

3.3 Surveillance Test Procedures

Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR 50. The proposed removal and test schedule shall be included in the Technical Specifications.

3.4 Reporting Requirements

All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include:

(1) Original properties and compositions of the materials.
(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall.
(3) Test results on surveillance specimens.
(4) Basis for evaluation of changes in RT_{NDT} and upper shelf toughness.
(5) Updated prediction of vessel properties.

3.5 Technical Specification Changes

Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Directorate of Licensing for approval.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.3.3 REACTOR VESSEL INTEGRITY

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

The portions of the applicant's safety analysis report (SAR) listed below are reviewed. These portions are all related to the integrity of the reactor vessel. Although most of these areas are reviewed separately in accordance with other review plans, the integrity of the reactor vessel is of such importance that a special summary review of all factors relating to the integrity of the reactor vessel is warranted. The information in each area is reviewed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

1. Design

The basic design of the reactor vessel is reviewed.

2. Materials of Construction

The materials of construction are each taken into consideration.

3. Fabrication Methods

The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are reviewed.

4. Inspection Requirements

The inspection test methods and requirements are reviewed.

5. Shipment and Installation

Protective measures taken during shipment of the reactor vessel and its installation at the site are reviewed.

6. Operating Conditions

All the operating conditions as they relate to the integrity of the reactor vessel are reviewed.

7. Inservice Surveillance

Plans and provisions for inservice surveillance of the reactor vessel are reviewed.

II. ACCEPTANCE CRITERIA

The basic acceptance criteria for each review area are covered by other standard review plans, so they will be discussed here only in general terms. References are made to the review plans that include detailed criteria. Interrelationships among review areas, and criteria for consistency, compatability, and technical coherence among review areas, are emphasized in the following discussion.

1. Design

The basic acceptance criteria for the design of the vessel are detailed in the standard review plans for SAR Sections 3 and 5.2. These cover the requirements of the General Design Criteria and Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"). The design of the reactor vessel must be compatible with the properties of the materials used, and must permit construction by the use of standard and well proven fabrication methods. The design details should not include new or novel concepts unless they are substantiated by a comprehensive justification showing that no aspects of the design will compromise the overall integrity of the vessel in any manner.

The design details must be adequate to permit all required inspections and to provide required access to all areas requiring inservice inspection in conformance with Section XI of the Code, as detailed in Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

If the neutron radiation exposure of the reactor vessel becomes high enough that the predicted value of the adjusted reference temperature of the material exceeds 200°F, the design must be adequate to permit in-place annealing of the vessel to restore ductility and toughness, in accordance with Appendix G, 10 CFR Part 50, and as detailed in the review plan for SAR Section 5.3.1, "Reactor Vessel Materials."

2. Materials of Construction

The basic acceptance criteria for the materials used in the construction of the reactor vessel are the requirements of Section III of the Code, as augmented by Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements." These criteria are detailed in Standard Review Plans 5.2.3, "Reactor Coolant Pressure Boundary Materials," and 5.3.1, "Reactor Vessel Materials."

The materials must be compatible with the design requirements. Acceptability is based on standard practice and engineering judgement, with consideration being given to such factors as material form, size-related variations in properties, and nonisotropic characteristics.

Although many materials are acceptable for reactor vessels according to Section III of the Code, the special considerations relating to fracture toughness and radiation effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr B Cl 1, SA 508 Cl 2, and SA 508 Cl 3. Acceptability criteria for other grades will have to be developed before they can be used.

The relationships among material compositions, expected neutron fluence, and requirements for the material surveillance program must be compatible. The reviewer uses published data to ensure that the predicted shift in toughness properties (RT_{NDT} and upper shelf energy) is conservative, based on actual material composition and predicted fluence. Acceptability of the material surveillance program, as specified in Appendix H, 10 CFR Part 50, depends on these relationships.

3. Fabrication Methods

Acceptability criteria for the basic fabrication processes and their qualification and control requirements are given in Sections III and IX of the Code, and detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

Although a particular fabrication process (such as multiple wire-high heat input welding) may be generally acceptable, it may not be suitable for reactor vessel fabrication for some materials without further justification or qualification. The reviewer uses "state-of-the-art" criteria and past practice to evaluate the acceptability of materials-process combinations.

Because fabrication methods, materials, and the effectiveness of non-destructive evaluation methods are interrelated, the reviewer must rely on state-of-the-art knowledge and past practice to determine whether the proposed combinations are compatible and acceptable.

4. Inspection Requirements

The basic requirements for performing nondestructive inspections and the quality assurance criteria for the reactor vessel are contained in Sections III and V of the Code. These are detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

Acceptance criteria for compatibility with materials and fabrication areas are discussed in previous sections.

Very important relationships are those among in-process and final shop inspections, and the inservice inspection requirements of Section XI of the Code. The reviewer must determine that the methods of inspection, the sensitivity levels, and flaw evaluation criteria are compatible with Section XI, and that the results of the preservice baseline inspection can be correlated with the results of later inservice inspections.

5. Shipment and Installation

The basic acceptance criteria for procedures and care used in shipping, storage, and installation of the vessel are given in Regulatory Guides 1.37, "Quality Assurance

Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Reactor Plants," 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," and 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

The purpose of this area of review is to verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. Acceptability in these areas is assured for current designs and materials by compliance with the basic acceptance criteria. If nonstandard materials or designs are used, the reviewer must determine that these criteria will be adequate, based on current technology.

If the basic criteria are not followed, either intentionally or through error, the reviewer must evaluate, on a case basis, whether the integrity of the reactor vessel is compromised, using current technology, past practice, and experience as applicable.

6. Operating Conditions

Acceptance criteria for operating limits for the reactor vessel are given in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," and are detailed in Standard Review Plan 5.3.2, "Pressure-Temperature Limits." In addition, Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" provides acceptable criteria for other phases of operational procedures.

Abnormal operational occurrences must not result in loss of reactor vessel integrity. The most severe postulated transient is the thermal shock to the vessel caused by emergency core cooling system operation after a loss-of-coolant accident. The criterion for acceptable behavior is that the vessel must remain leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the basis for acceptance of analyses that support conformance with this criterion.

7. Inservice Surveillance

The acceptance criteria for adequacy of the reactor vessel materials surveillance program are based on the requirements of Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," and detailed in Standard Review Plan 5.3.1, "Reactor Vessel Materials."

The SAR also provides information regarding the inservice inspections to be performed on the reactor vessel. The acceptance criteria for accessibility and inspection plan details are those of Section XI of the Code, and are detailed in Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case. The reviewer initially determines that the basic criteria are met in each review area covered by this plan. Although he will not normally be responsible for the basic reviews of all of these areas, he will consult with those responsible for basic review of the other areas to determine that all areas are individually acceptable.

He then reviews each area again, considering the information presented in other areas that interrelate with it, as discussed in II above.

Because the reviewer is familiar with the specific procedures used by the reactor vendor, he can readily pick out any differences from past practice. He will evaluate these in detail, consulting with other MTEB members as appropriate.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided to satisfy the requirements of this review plan, and that the completeness and technical adequacy of his evaluation will support conclusions of the following type, to be included in the staff's safety evaluation report:

"We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for this plant. The bases for our conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable AEC Regulations and Regulatory Guides, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III. The stringent fracture toughness requirements of the Regulations and ASME Code Section III will be met, including requirements for surveillance of vessel material properties throughout service life. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile Failure," of ASME Code Section III, and Appendix G, 10 CFR Part 50.

"The integrity of the reactor vessel is assured because the vessel

(1) will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and any pertinent Code Cases;
(2) will be made from materials of controlled and demonstrated high quality;
(3) will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies;
(4) will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation, and that the vessel will not fail under the conditions of any of the postulated accidents;
(5) will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions; and
(6) may be annealed to restore the material toughness properties if this becomes necessary."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

3. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

4. ASME Boiler and Pressure Vessel Code, Section III, especially Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

5. ASME Boiler and Pressure Vessel Code, Sections II, V, IX, and XI, American Society of Mechanical Engineers.

6. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

7. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

8. Regulatory Guide 1.38, "Quality Assurance Requirements in Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Plants."

9. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

10. Standard Review Plan 5.2.3, "RCPB Materials."

11. Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing."

12. Standard Review Plan 5.3.1, "Reactor Vessel Materials."

13. Standard Review Plan 5.3.2, "Pressure-Temperature Limits."

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PREFACE TO SECTION 5.4

The Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants, Revision 2, under 5.4, "Component and Subsystem Design," contains several paragraphs that provide examples of principal components and subsystems within or allied with the reactor coolant system. The technical review of many of these principal items is conducted under the cognizance of Standard Review Plans which have been prepared for other sections of the Standard Format.

An outline of the material to be reviewed and the Standard Review Plan (SRP) under which it is covered is presented below for those items reviewed in other plans.

5.4.1 Reactor Coolant Pumps

The RSB under SRP 4.4 reviews the process design parameters. Flow coastdown and startup characteristics are reviewed under the Chapter 15 SRP.

The MEB under SRP 3.9.1-3.9.3 and 3.9.6 reviews the structural integrity and operability, the methods of analysis, and the dynamic testing of pumps.

The MTEB under SRP 5.2.4 reviews the inservice inspection and functional testing and under SRP 5.2.3 the materials of fabrication.

5.4.2 Steam Generators (PWR)

The RSB under SRP 4.4 reviews the configuration and process design parameters. The response to various anticipated transients and accidents is reviewed under various SRP for Chapter 15.

The MEB under SRP 3.9.1-3.9.3 reviews the structural adequacy, the methods of analysis, and the dynamic testing of the steam generator.

The MTEB reviews the materials and the inservice inspection program under SRP 5.4.2.1 and 5.4.2.2.

5.4.3 Reactor Coolant Piping

The RSB under SRP 4.4 reviews the piping and instrumentation diagrams, process flow features, and equipment arrangements.

The MEB under SRP 3.9.1-3.9.3 reviews the structural integrity, the methods of analysis, and the dynamic testing of the reactor coolant piping.

The MTEB reviews the materials and inservice inspection program under SRP 5.2.3 and 5.2.4.

5.4.4 Main Steam Line Flow Restrictions

The RSB reviews the functional requirements under the Chapter 15 SRP.

The MEB reviews the mechanical design bases, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

5.4.5 Main Steam Line Isolation System

This is covered in SRP 9.5.9. The APCSB has primary review responsibility.

5.4.6 Reactor Core Isolation Cooling System (BWR)

A review plan is provided on this subject. The RSB has primary responsibility.

5.4.7 Residual Heat Removal (RHR) System

A review plan is provided on this subject. The RSB has primary responsibility.

5.4.8 Reactor Coolant Cleanup System (BWR)

A review plan is provided on this subject. The ETSB has primary responsibility.

5.4.9 Main Steam Line and Feedwater Piping

The APCSB reviews the functional and related requirements under SRP 10.3 and 10.4.9.

The MEB reviews the structural integrity, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

The MTEB reviews the materials of fabrication under SRP 10.3.6.

5.4.10 Pressurizer

The RSB reviews the configuration and process design parameters under SRP 4.4. Related safety and relief valve capacities are reviewed under SRP 5.2.2. Performance under system transients is reviewed under appropriate SRP of Chapter 15.

The MEB reviews the structural integrity, methods of analysis, and dynamic testing under SRP 3.9.1-3.9.3.

The MTEB reviews the inservice testing and inspection under SRP 5.2.4, and the materials under SRP 5.2.3.

5.4.11 Pressurizer Relief Tank

A review plan is provided on this subject. The APCSB has primary responsibility.

5.4.12 Valves

The RSB reviews the functional aspects of valves within and connected to the reactor coolant pressure boundary under the cognizant review plan. These review plans include SRP 5.4.6, 5.4.7, and 6.3.

The MEB reviews the structural integrity and operability, the methods of analysis, the dynamic testing, and functional testing of valves under SRP 3.9.1-3.9.3 and 3.9.6.

The MTEB reviews valve materials under SRP 5.2.3, 6.1.1, and 10.3.6, and inservice inspection under SRP 5.2.4 and 6.6.

5.5.13 Safety and Relief Valves

The RSB reviews set points and capacities of primary coolant system safety and relief valves under SRP 5.2.2.

The MEB reviews the structural integrity, methods of analysis, dynamic testing, and functional testing under SRP 3.9.1-3.9.3 and 3.9.6.

The MTEB reviews the inservice inspection program under SRP 5.2.4 and 6.6, and materials under SRP 5.2.3, 6.1.1, and 10.3.6.

5.4.14 Component Supports

The MEB reviews the structural integrity, methods of analysis, and dynamic testing of component supports under SRP 3.9.1-3.9.3.

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SECTION 5.4.1.1 PUMP FLYWHEEL INTEGRITY (PWR)

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 4 requires that structures, systems, and components of nuclear power plants important to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of 900 rpm or 1200 rpm during normal reactor operation, a loss of integrity could result in high energy missiles and excessive vibration of the reactor coolant pump assembly. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment, or the engineered safety features.

The following areas relating to reactor coolant pump flywheel integrity are reviewed:

1. Materials Selection

Reactor coolant pump flywheels are of a simple geometric shape, and are made of ductile material. Their quality can be closely controlled and their service conditions are not severe; therefore, the use of suitable material, coupled with adequate design and inservice inspection can provide a sufficiently small probability of a flywheel failure that the consequences of failure need not be protected against.

Information in the applicant's safety analysis report (SAR) on materials selection and the procedures used to minimize flaws and improve mechanical properties is reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the flywheel materials.

2. Fracture Toughness

The fracture toughness of the materials, including materials tests, correlation of Charpy specimens to fracture toughness parameters, or the alternate use of a nil-ductility transition reference temperature (RT_{NDT}), are reviewed to establish that the flywheel materials will exhibit adequate fracture toughness at normal operating temperature.

3. Preservice Inspection

The descriptive information is reviewed to verify that the bore of the flywheel is machined to final dimensions if it is flame cut, and that ultrasonic and surface inspections are performed on all finished machined surfaces.

4. Flywheel Design

The flywheel design information including allowable stresses, design overspeed considerations, and shaft and bearing design adequacy, is reviewed.

5. Overspeed Test

The applicant's overspeed test procedures are reviewed to establish their adequacy.

6. Inservice Inspection

A description of the preservice and postoperational phases of the inservice inspection program, including types of inspections, areas inspected, frequencies of inspection, and flaw acceptance criteria, is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are provided in Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and are as follows:

1. Materials Selection

The applicant's selection of flywheel material is acceptable if it is in accordance with the following criteria:

The flywheel material must be produced by a process (such as vacuum melting or degassing) that minimizes flaws in the material and improves its fracture toughness properties. The material must be examined and tested to meet the following criteria:

a. The nil-ductility transition (NDT) temperature of the flywheel material, as obtained from drop-weight tests (DWT) performed in accordance with the specification ASTM E-208 (Ref. 3), should be no higher than 10°F.

b. The Charpy V-notch (C_v) upper-shelf energy level in the "weak" direction (WR orientation in plates) of the flywheel material should be at least 50 ft-lbs. A minimum of three C_v specimens should be tested from each plate or forging, in accordance with ASTM A-370 (Ref. 4).

2. Fracture Toughness

The following fracture toughness criteria are derived from Regulatory Guide 1.14, C.1.c, and the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Appendix G. The pump flywheel fracture toughness properties are acceptable if they are in compliance with the following criteria:

The minimum static fracture toughness of the material at the normal operating temperature of the flywheel should be equivalent to a critical stress intensity factor, K_{Ic}, of at least 150 ksi $\sqrt{\text{in.}}$ Compliance can be demonstrated by either of the following:

a. Testing of the actual material to establish the K_{Ic} value at the normal operating temperature.

b. Determining that the normal operating temperature is at least 100°F above the RT_{NDT}.

3. Preservice Inspection

The following preservice inspection criteria are derived from Regulatory Guide 1.14, C.1.d, C.1.e, and C.1.f. The applicant's preservice inspection program including finish machining and ultrasonic and surface inspections is acceptable if in compliance with the following criteria:

a. Each finished flywheel should be subjected to a 100% volumetric examination by ultrasonic methods using procedures and acceptance criteria specified in Code Section III, NB-2530 for plates, and NB-2540 for forgings.

b. If the flywheel is flame cut from a plate or forging, at least 1/2 inch of material should be left on the outer and bore radii for machining to final dimensions.

c. Finish machined bores, keyways, splines, and drilled holes should be subjected to magnetic particle or liquid penetrant examination.

4. Flywheel Design

The following flywheel design criteria are derived from Regulatory Guide 1.14, C.2. The applicant's flywheel design is acceptable if in compliance with the following criteria:

The flywheel should be designed to withstand normal conditions, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake without loss of structural integrity.

The design of the pump flywheel should meet the following criteria:

a. The combined stresses at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, should not exceed 1/3 of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel.

b. The design overspeed of a flywheel should be at least 10% above the highest anticipated overspeed. The anticipated overspeed should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump. The basis for the assumed design overspeed should be submitted to the staff for review.

c. The combined stresses at the design overspeed, due to centrifugal forces and the interference fit, should not exceed 2/3 of the minimum specified yield strength, or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel.

d. The shaft and the bearings supporting the flywheel should be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake.

5. Overspeed Test

The following overspeed test criterion is taken from Regulatory Guide 1.14, C.3. The applicant's commitment to perform an overspeed test is acceptable if each flywheel assembly is to be tested at the design overspeed of the flywheel.

6. Inservice Inspection (ISI)

The following inservice inspection program criteria are derived from Regulatory Guide 1.14, C.4. The applicant's ISI program is acceptable if in compliance with the following:

a. A volumetric examination by ultrasonic methods of the areas of higher stress concentration at the bore and keyway at approximately 3-1/3 year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the Code, Section XI. Removal of the flywheel is not required.
b. A surface examination by liquid penetrant or magnetic particle methods of all exposed surfaces, and 100% volumetric examination by ultrasonic methods at approximately ten-year intervals, during the plant shutdown coinciding with the inservice inspection schedule as required by the Code, Section XI. Removal of the flywheel is not required.
c. A preservice baseline inspection incorporating all the procedures of a and b above, which should establish initial flywheel conditions, accessibility, and practicality of the program.
d. Examination procedures and acceptance criteria should be in conformance with the requirements specified in II.3.a.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedure is followed:

1. Materials Selection

The materials selection, including the procedures to minimize flaws and improve mechanical properties described by the applicant, are reviewed and compared with the requirements of Section II.1 of this plan. If it is a new material not used in prior licensing cases, the materials selection is reviewed and evaluated to establish its acceptability. Based on past evaluations, the following materials are suitable for pump flywheels provided that they meet all the criteria listed in II. 1 and II. 2 of this plan: ASME SA-533-B Class 1, ASME SA-508 Class 2, and ASME SA-516 Grade 65 (Ref. 2).

2. Fracture Toughness

The fracture toughness properties of the flywheel materials, including test data where applicable, are reviewed and compared with the requirements of Section II.2 of this plan. Two alternative methods for deriving the fracture toughness of the flywheel materials are acceptable. The value of the critical stress intensity factor is based on fracture mechanics testing, while the use of the reference temperature approach is based on the stated normal operating temperature of the flywheel and the actual reference nil-ductility transition temperature of the materials, if an operating license review, or as specified, if a construction permit review.

3. Preservice Inspection

The preservice inspection program, including finish machining, and ultrasonic and surface inspections described by the applicant is reviewed and compared with the requirements of Section II.3 of this plan. The extent to which the ultrasonic inspections proposed and the acceptance criteria in the SAR agree with Code Section III, NB-2530 for plate materials or NB-2540 for forgings, are reviewed.

4. Flywheel Design

The design and stress analysis procedures used for the flywheel are reviewed, including the following areas:

a. Load combinations at normal operating speed and allowable stresses.
b. Design overspeed and basis for selection of design overspeed.
c. Load combinations or design overspeed and allowable stresses.
d. Shaft and bearing load combinations.

The information given in the SAR is compared and evaluated against Section II.4 of this plan.

5. Overspeed Test

The applicant should confirm that an overspeed test will be run in compliance with Section II.5 of this plan.

6. Inservice Inspection

The inservice inspection program described by the applicant in the plant technical specifications, including areas to be inspected, methods of inspection, frequency of inspection, and acceptance criteria, is reviewed and compared with the requirements of Section II.6 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, preservice spin testing, and inservice inspection.

"The applicant's selection of materials, fracture toughness tests, design procedures, preservice overspeed spin testing program, and inservice inspection program for reactor coolant pump flywheels have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and established industry codes and standards.

"The use of suitable materials with adequate fracture toughness, conservative design procedures, preservice testing, and inservice inspection for flywheels of reactor coolant pump motors provide reasonable assurance of the structural integrity of the flywheels in the event of design overspeed transients or postulated accidents. Conformance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying in part the requirements of General Design Criterion 4."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

2. ASME Boiler and Pressure Vessel Code, Sections II, III, and XI, American Society of Mechanical Engineers.

3. ASTM E-208-69, "Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

4. ASTM A-370-72, "Methods and Definitions for Mechanical Testing of Steel Products," Annual Book of ASTM Standards, Part 31, American Society for Testing and Materials.

5. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity" (originally Safety Guide 14).

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SECTION 5.4.2.1 STEAM GENERATOR MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 14, 15, and 31 of Appendix A of 10 CFR Part 50 require that the reactor coolant pressure boundary (RCPB) must have an extremely low probability of abnormal leakage and must be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation and anticipated operational occurrences, and that the probability of rapidly propagating failure of the RCPB is minimized.

A review is made of the following areas, reported in the applicant's safety analysis report (SAR). These are all related to the ASME Boiler and Pressure Vessel Code (hereafter "the Code") Class 1 and 2 materials of pressurized water reactor (PWR) steam generators, including all components that constitute part of the reactor coolant pressure boundary.

1. Selection and Fabrication of Materials

The materials selected for the steam generator are reviewed.

Materials for components of the steam generator are divided into two classes: Class 1, which includes material for those parts exposed to the primary reactor coolant, and Class 2, which includes materials for parts exposed to the secondary coolant water.

Class 1 component materials include the following:

Inconel 600 Tubing	-	ASME SB-163, Ni-Cr-Fe, Annealed
Carbon Steel Tube Sheet	-	ASME SA-533, Grade A, weld-clad with Inconel 600 on the primary coolant side
Channel Head Casting or	-	ASME-SA-216, Grade WCC, Class 1, weld-clad with austenitic stainless steel
Channel Head Plate	-	ASME - SA-533, Grade A, B, or C
Forged Nozzles	-	ASME SA-503, Class 2

Class 2 component materials include the following:

Shell Pressure Plates	-	ASME - SA-533, Grade A, B, or C, Class 2
Bolting	-	ASME SA-193, Grade B-7
		ASME SA-540, Grade B 23 or B 24

The Inconel-600 tubes are commonly welded to the tube-sheet cladding and expanded into the tube sheet by rolling or explosive-expanding (explanding). Full depth expansion is the preferred design, especially for "U"-tubed steam generators.

The adequacy and suitability of the ferritic materials are reviewed. The fracture toughness properties and requirements for Class 1 and Class 2 ferritic components are reviewed.

2. <u>Steam Generator Design</u>
   The extent of crevice areas in the design of the steam generator is reviewed.

3. <u>Compatibility of the Steam Generator Components with the Primary and Secondary Coolant</u>
   The possibility of stress-corrosion cracking and wastage of the tubes as determined by the chemistry of both the primary and secondary coolants are reviewed. The methods to be used in monitoring and maintaining the chemistry of the secondary coolant within the specified ranges are reviewed. The compatibility of ferritic low alloy steels and carbon steels with the primary and secondary coolants is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. <u>Selection and Fabrication of Materials</u>
   The mechanical properties of the materials selected for the steam generator components must meet the Code requirements given in Appendix I of Section III and Parts A, B, and C of Section II. The corrosion-resistant weld-deposited cladding on the tube sheet must be made and inspected according to the requirements given in Article QW-214 of Section IX of the Code. The tubes in U-tubed steam generators must be rolled or "explanded" for the full depth of the tube sheet to avoid the presence of a deep crevice between the tube and tube sheet, as recommended in the Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators" (Ref. 7), appended. Onsite cleaning and cleanliness control should be in accordance with the position given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (Ref. 4), and in ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants" (Ref. 3). The welds between the tubes and the tube sheet must meet the requirements of Section III and Section IX of the Code. Any materials designed to code-case requirements must meet the requirements given in Regulatory Guide 1.85, "Code-Case Acceptability-Materials" (Ref. 5).

The fracture toughness of ferritic materials used for Class 1 components in the steam generator must meet the requirements of Article NB-2300 of Code Section III, and Appendix G, Paragraph G-2000.

The fracture toughness properties of the ferritic materials selected for Class 2 components in the steam generator must meet the requirements of paragraph NC-2310 of the Summer 1972 Addenda to Section III of the Code, which state that the test requirements and acceptance standards for Class 2 components must be the same as specified for Class 1 components. Paragraph NB-2332(b) states that for Class 2 components greater than 2 1/2 in. wall thickness, the lowest service temperature must not be less than the nil-ductility transition reference temperature, RT_{NDT}, plus 100°F, unless a lower temperature is justified by methods similar to those contained in Article G-2000 of the Summer 1972 Addenda to Section III.

2. Steam Generator Design
The steam generators must be designed to avoid extensive crevice areas where the tubes pass through the tube sheet, and where the tubes pass through tubing supports, as indicated in Branch Technical Position MTEB 5-3 (Ref. 7).

3. Compatibility of The Steam Generator Tubing with the Primary and Secondary Coolant
The acceptance criteria for primary coolant chemistry are given in Standard Review Plan 5.2.3, "RCPB Materials." The secondary coolant purity should be monitored as described in Reference 7.

III. REVIEW PROCEDURES

The reviewer wwill select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review, the following review procedure is followed:

1. Selection and Fabrication of Materials
The reviewer examines the materials and fabrication procedures as given in the SAR for Class 1 and Class 2 components of the steam generators, to determine the degree of conformance with the acceptance criteria stated in Section II.1. The reviewer verifies that the tubes are properly welded and expanded into the tube sheet, that proper care is taken to maintain cleanliness during fabrication, assembly, and installation of the unit, and that information relative to impact tests is in conformance with the acceptance criteria stated in Section II.1.

2. Steam Generator Design
The reviewer examines the design of the steam generators to verify that tight crevice areas where tubes pass through the tube supports are minimized, as discussed in Section II.2.

3. <u>Compatibility of the Steam Generator Tubing with the Primary and Secondary Coolant</u>
The reviewer examines the controls to be placed on the composition of the primary and secondary coolants to determine that they meet the acceptance criteria cited in Section II.3.

4. <u>General</u>
If the information contained in the safety analysis reports or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information, but also the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The materials used in Class I and Class 2 components of the steam generators were selected and fabricated according to codes, standards, and specifications acceptable to the staff. The onsite cleaning and cleanliness controls during fabrication conform to the recommendations of Regulatory Guide 1.37, "Cleaning of Fluid Systems and Associated Components during the Construction Phase of Nuclear Power Plants." The controls placed on secondary coolant chemistry are in agreement with established staff technical positions. Conformance with applicable codes, standards, staff positions, and Regulatory Guides constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 14, 15, and 31."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary," Criterion 15, "Reactor Coolant System Design," and Criterion 31, "Fracture Prevention of The Reactor Coolant Pressure Boundary."

2. ASME Boiler and Pressure Vessel Code, Part A of Section II, Appendix I of Section III, and Section IX, American Society of Mechanical Engineers.

3. ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants," Draft 2, Revision 0, November 15, 1973, American National Standards Institute.

4. Regulatory Guide 1.37, "Quality Assurance Requirements of Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.85, "Code Case Applicability-Material."

6. Standard Review Plan 5.2.3, "RCPB Materials."

7. Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," appended.

BRANCH TECHNICAL POSITION MTEB 5-3

MONITORING OF SECONDARY SIDE WATER CHEMISTRY IN PWR STEAM GENERATORS

A. BACKGROUND

In view of the extensive history of stress corrosion or wastage of steam generator tubing in operating PWR's, we recommend the following criteria.

B. Branch Technical Position

1. Crevices between the tubing and the tube sheets or tubing supports should be minimized to prevent concentration of impurities or solids in these areas. Steam generators incorporating Inconel 600 tubes should be designed and built to achieve this goal.

2. The methods utilized for control of secondary side water chemistry should be described. In plants having more than one steam generator, additives to each steam generator should be controlled separately. Records should be made of the following items, and summaries of the data should be available for report as requested by the Commission.

   a. For plants utilizing volatile chemistry:

      (1) The composition, quantities, and rates of addition of additives should be recorded initially and whenever a change is made.

      (2) The electrical conductivity and the pH of the bulk steam generator water and feedwater should be measured continuously.

      (3) For once-through steam generators, the pH and electrical conductivity at the coolant inlet should be measured continuously.

      (4) Free hydroxide concentration and impurities (particularly chloride, ammonia and silica) in the steam generator water should be measured at least three times per week (daily if serious condenser leakage is occurring).

      (5) The electrical conductivity of the condensate should be measured at least once weekly (daily if serious condenser leakage is occurring).

      (6) The condenser leakage should be measured at least daily in freshwater-cooled plants, and continuously in seawater-cooled plants.

   b. For older plants still utilizing phosphate treatment:

      (1) The composition, quantity, and rate of addition of each additive should be recorded initially and whenever a change is made.

      (2) The Na/PO_4 molar ratio of the secondary coolant should be recorded initially and whenever a change is made.

      (3) The electrical conductivity and pH of the bulk steam generator water and feedwater should be measured continuously.

      (4) The concentration of suspended/dissolved solids and impurities (particularly free caustic, chloride, silica, and sodium) in the steam generator water should be measured daily.

(5) The concentration of dissolved solids (particularly sodium and phosphate) in the blowdown liquid should be measured once each week.

(6) The rate of blowdown should be recorded initially and whenever a change in rate is made.

(7) The hideout and reverse hideout of phosphate should be recorded. The phosphate concentration in each steam generator (or in one steam generator if this is shown to be representative of all) and in the blowdown liquid should be measured before and after each planned power level change of 10% or greater, and should be measured after each unplanned power level change of 20% or greater.

(8) The condenser leakage should be measured at least daily in freshwater-cooled plants and continuously in seawater-cooled plants.

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SECTION 5.4.2.2 STEAM GENERATOR INSERVICE INSPECTION

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 1 and 32 of Appendix A of 10 CFR Part 50 require that components which are part of the reactor coolant pressure boundary (RCPB) or other components important to safety be designed to permit periodic inspection and testing of critical areas for structural and leaktight integrity. The design of the steam generators as described in the preliminary safety analysis report (PSAR) is reviewed to establish that use of the specified inspection techniques is feasible. The provisions made for baseline inspection prior to startup, the methods to be used for the inspections, and the inservice inspection program are reviewed in the final safety analysis report (FSAR) and plant Technical Specifications.

II. ACCEPTANCE CRITERIA

The design of the steam generators to provide access for an inservice inspection (ISI) program, and the proposed ISI program should follow the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section XI, and the recommendations given in Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes." Specifically, the steam generators should be designed to permit inspection of any component, including individual tubes, and the repair or replacement of any component. The tube examination equipment and procedures should be capable of detecting and locating defects with a penetration of 20% or more of the wall thickness. A permanent record of test data should be provided. A baseline tube inspection should be scheduled prior to startup. The sample selection and testing of tubes, the inspection intervals, and the actions to be taken if defects are identified should follow the recommendations of Regulatory Guide 1.83.

Section XI provides for the volumetric inspection of the following components: longitudinal and circumferential welds, including tube-sheet-to-head or shell welds on the primary side; primary nozzle-to-vessel head welds and nozzle-to-head inside radiused section; primary nozzle to safe-end welds; primary side pressure-retaining bolting; integrally-welded primary vessel supports; circumferential butt welds on the secondary shell side; and nozzle-to-vessel welds on the secondary side. The volumetric inspections are supplemented, as detailed in the Code, by visual and surface inspections for both primary and secondary steam generator Code Class 1 and 2 components.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case. He determines that the design of the steam generators, as described in the PSAR, will permit access for the specified inspection techniques of all steam generator components including tubes. He also evaluates the design of the steam generator as described in the FSAR and the Technical Specification inservice inspection program to determine the degree to which the recommendations of Regulatory Guide 1.83 and Code Section XI have been followed. The reviewer determines that the techniques to be used for inservice inspection are those listed in Section XI. He determines that the inspection technique for the tubes, the selected number of tube samples, the inspection intervals, and the actions to be taken in the event defects are observed are in accordance with the positions stated in the guide. He determines that a baseline inspection will be made prior to startup of the plant. Standard Review Plan 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," covers the ISI of Class 1 components of the steam generators. Standard Review Plan 6.6, "Inservice Inspection of Class 2 and 3 Components," is applicable to ISI of the balance of the components of the steam generator, which are all Class 2 components.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> "The steam generators have been designed to permit inservice inspection of all Code Class 1 and 2 components including individual tubes as recommended in Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes," and ASME Code Section XI. [The inservice inspection program for the steam generators is in accordance with the recommendations of the above-cited Regulatory Guide and ASME Code Section XI with respect to the inspection methods to be used, provisions for a baseline inspection, selection and sampling of tubes, inspection interval, and actions to be taken in the event defects are identified.*] Conformance with Regulatory Guide 1.83 and ASME Code Section XI constitutes an acceptable basis for meeting in part the requirements of General Design Criteria 1 and 32."

V. REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

*The statement enclosed in brackets should only be included in the staff's safety evaluation report for an FSAR.

3. ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

4. Regulatory Guide 1.83, "Inservice Inspection of Steam Generator Tubes."

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SECTION 5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The reactor core isolation cooling (RCIC) system in a boiling water reactor (BWR) is a safety system which serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. Abnormal events which could cause such a situation to arise include an inadvertent isolation of all main steam lines, loss of condenser vacuum, pressure regulator failure, loss of feedwater, the loss of offsite power, and total loss of all a-c power (both offsite and diesel generators). Each of these transients is analyzed in Chapter 15 of the applicant's safety analysis report (SAR). For each of these events, the high pressure part of the emergency core cooling system (ECCS) provides a backup function to the RCIC system.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater system, supplemented by in-leakage from the control rod drive system. If the feedwater system is inoperable, the RCIC turbine-pump unit starts automatically or is started by the operator from the control room. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool.

The review of the RCIC system includes the system design bases, design criteria, description, and the points noted below.

1. The piping and instrumentation diagram is reviewed to determine that the system is capable of performing its intended function and of being preoperationally and operationally tested.

2. The degree of separation from the high pressure core spray (HPCS) system, or high pressure core injection (HPCI) system for 1967 product line or earlier BWR's, and protection against common mode failures of both redundant systems (e.g., from flooding, fire, pipe whip, or high temperature, pressure, and humidity) are reviewed.

3. The process flow diagram is reviewed to confirm that the RCIC system design parameters are consistent with expected pressures, temperatures, and flow rates.

4. The complete sequence of operation is reviewed to determine that the system can function as intended and that the system is capable of manual operation.

5. The proposed preoperational and initial startup test programs are reviewed to determine their adequacy.

6. The proposed technical specifications are evaluated to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

7. The RCIC system is reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as part of the effort described in Standard Review Plans (SRP) 3.2.1 and 3.2.2. The RCIC system is to be enclosed in a structure having the proper seismic classification. The review of the building seismic category is also accomplished as a portion of the effort described in SRP 3.2.2.

   The RCIC is to be located in a structure that provides adequate protection against wind, tornadoes, floods, and missiles (as appropriate). The review of the building adequacy is performed as described in other sections of the standard review plans.

8. The CSB reviews the RCIC system, as described in SRP 6.2.4, to confirm that the design is compatible with the containment system and can be isolated.

9. The EICSB, as described in SRP 7.4, evaluates the adequacy of controls and instrumentation of the RCIC system with regard to the required features of automatic actuation, remote sensing and indication, remote control, emergency onsite power, sufficient battery capacity, and use of d-c power only.

10. The MEB, as described in SRP 3.9.3, ensures that the design and installation of the RCIC system meet applicable codes and are adequate for its proper functioning.

11. The CPB, on request, reviews the core decay energy output on which the design is based to see that it is applicable and suitably conservative.

12. The MTEB reviews the materials and the inservice inspection program for the RCIC system.

II. ACCEPTANCE CRITERIA

The general objective of the review is to determine that the RCIC system, in conjunction with the HPCS (or HPCI) system meets the requirements of General Design Criteria 34, (Ref. 3) by providing the capability for decay heat removal to allow complete shutdown of the reactor under conditions requiring its use. It must maintain the reactor water inventory until the reactor is depressurized sufficiently to permit operation of the low pressure cooling systems. The RCIC system, in conjunction with the HPCS (or HPCI) system, must be capable of removing fission product decay heat and other residual heat from the reactor core following shutdown so as to preclude fuel damage or reactor coolant pressure boundary over-pressurization. For the purposes of this plan, the minimum critical heat flux ratio (MCHFR) should be greater than 1.0, based on Reference 7, or the minimum critical power ratio (MCPR) should be greater than X*, based on Reference 8, to preclude fuel damage. The maximum reactor pressure should be less than 110% of design pressure (Ref. 9).

Historically, credit has been taken for RCIC system capability to mitigate the consequences of certain abnormal events; however, since the cooling function is redundant to the HPCI or HPCS system, the RCIC system does not have to meet the single failure criterion. However, the system is to perform its function without the availability of any a-c power. As a system which must respond to certain abnormal events, the RCIC system must be designed to seismic Category I standards, as defined in Regulatory Guide 1.29 (Ref. 10).

The RCIC and HPCS (or HPCI) systems must be protected against natural phenomena, external or internal missiles, pipe whip, and jet impingement forces so that such events cannot fail both systems simultaneously. Jointly, the two systems must meet General Design Criterion 2 (Ref. 1); General Design Criterion 4 (Ref.2); Regulatory Guide 1.46 (Ref. 11); and the staff positions on protection for pipe failures outside containment (Ref. 13).

The RCIC system must meet the requirements of General Design Criteria 55, 56, and 57 (Refs. 4, 5, and 6) with regard to isolation provisions for lines passing through the primary containment.

If the RCIC system is used to control or mitigate the consequences of an accident, either by itself or as a backup to another system, it must meet the requirements of an engineered safety feature.

The preoperational and initial startup test programs for the RCIC system should meet the intent of Regulatory Guide 1.68 (Ref. 12).

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*The value of MCPR will vary for different product lines. The value of MCPR used for a particular case review is to be consistent with the value specified in the plant technical specifications as the fuel integrity safety limit.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For the operating license (OL) review, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the reviewer to determine that the acceptance criteria of Section II have been met. The steps are adapted to CP or OL reviews as appropriate.

1. Using the RCIC operating requirements specified in SAR Section 5.4.6 and Chapter 15, the reviewer confirms that the RCIC can function when required so as to prevent the MCHFR from decreasing below 1.0 or the critical power ratio from decreasing below X* (based on Reference 7 or Reference 8) and prevent the reactor pressure from exceeding 110% of design pressure. This determination is based on engineering judgment and independent calculations (where deemed necessary), using information as specified in steps 2 and 3 below. The reviewer consults with the CPB to assure that the decay heat loads used in the RCIC analyses are applicable and suitably conservative. The reviewer also determines that the RCIC system maintains sufficient coolant inventory in the reactor vessel to keep the core covered and assure clad integrity.

2. Using the description given in Section 5.4.6 of the SAR, including component lists and performance specifications, the reviewer determines that the RCIC system piping and instrumentation are such as to allow the system to operate as intended. This is accomplished by reviewing the piping and instrumentation diagrams to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information.

3. Using the comparison tables of SAR Section 1.3, the RCIC system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

4. The reviewer checks the piping and instrumentation diagrams and equipment layout drawings for the RCIC and HPCS (or HPCI) systems to see that the systems are physically separated and can function independently and that they jointly conform to the requirements of General Design Criteria 2 and 4 and the recommendations of Regulatory Guide 1.46 and staff positions on piping failures outside containment (Refs. 1, 2, 11, and 13).

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*IBID. Page 5.4.6-3

5. Based on the description in SAR Section 5.4.6, the reviewer judges whether adequate control and monitoring information is available to allow the operator to actuate the system manually or to realign the RCIC system manually within the time allowed (i.e., change the RCIC system suction from the condensate storage tank to the suppression pool or residual heat removal system).

6. The reviewer contacts EICSB to confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. As part of their review, the EICSB is to ascertain that the RCIC system operation is not dependent on a-c power sources, and that there is sufficient battery capacity to permit operation of the RCIC for a period of two hours without the availability of a-c power. The instrumentation and controls of the RCIC system, in conjunction with the HPCS (or HPCI) system, are to have sufficient redundancy to satisfy the single failure criterion.

7. The reviewer checks the piping and instrumentation diagrams to see that essential RCIC system components are designated seismic Category I.

8. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 (Ref. 12). At the OL stage, the reviewer assures that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68).

   The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the RCIC system will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the RCIC system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the RCIC system. If the RCIC system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

9. The proposed plant technical specifications are reviewed to:

   a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

   b. Verify that the frequency and scope of periodic surveillance testing is adequate.

10. The reviewer confirms that the RCIC is housed in a structure whose design and design criteria have been reviewed by other branches to assure that it provides adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

"The reactor core isolation cooling (RCIC) system includes the piping, valves, pumps, turbines, instrumentation, and controls used to maintain water inventory in the reactor vessel whenever it is isolated from the main feedwater system. Certain engineered safety features (HPCS or HPCI) provide a redundant backup for this function. The scope of review of the RCIC system for the ______ plant included piping and instrumentation diagrams, equipment layout drawings, and functional specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RCIC system, his analysis of the adequacy of the criteria and bases, and the conformance of the design to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses have been reviewed and have been found to conform to Commission regulations as set forth in General Design Criteria 2, 4, 34, 55, 56, and 57, and to applicable regulatory guides and staff technical positions. The RCIC system has been found to conform to Regulatory Guide 1.29. The RCIC system and HPCS (or HPCI) system jointly conform to General Design Criteria 2, 4, 34 and Regulatory Guide 1.46. The two systems have been found capable of removing core decay heat following feedwater system isolation and reactor shutdown so that the core minimum critical heat flux ratio (MCHFR) does not decrease below 1.0 (or the critical power ratio does not decrease below ______), and the pressure within the reactor coolant pressure boundary does not exceed 110% of design pressure. This capability has been found to be available even with a loss of offsite power and with a single active failure. The staff concludes that the design of the reactor core isolation cooling system conforms to the Commission's regulations and to applicable regulatory guides and staff positions, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

4. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

5. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

6. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

7. J. M. Healzer, et al, "Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors," APED-5286, General Electric Company, September 1966.

8. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company, November 1973.

9. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

10. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

11. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

12. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

13. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP 3.6.2.

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SECTION 5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor System Branch (RSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Materials Engineering Branch (MTEB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The residual heat removal (RHR) system is used in conjunction with the main steam and feedwater systems (main condenser), or the reactor core isolation cooling (RCIC) system in a boiling water reactor (BWR), or auxiliary feedwater system in a pressurized water reactor (PWR) to cool down the reactor coolant system following shutdown. Parts of the RHR system also act to provide low pressure emergency core cooling and are reviewed as described in Standard Review Plan (SRP) 6.3. Some parts of the RHR system also provide containment heat removal capability and are reviewed as described in SRP 6.2.2.

Both PWR's and BWR's have RHR systems which provide long term cooling once the initial decay heat load is removed by the main condenser, RCIC, or auxiliary feedwater systems. In both types of plants, the RHR is a low pressure system which takes over the shutdown cooling function when the reactor coolant system (RCS) temperature is reduced to about 300°F. Although the RHR system function is similar for the two types of plants, the system designs are different.

The RHR system in PWR's is composed of piping, pumps, valves, heat exchangers, monitors, and controls which take water from the RCS hot legs, cool it, and pump it back to the cold legs or core flooding tank nozzles. The suction and discharge lines for the RHR pumps have appropriate valving to assure that the low pressure RHR system is always isolated from the RCS when the reactor coolant pressure is greater than the RHR design pressure. The heat removed in the heat exchangers is transported to the ultimate heat sink by the component cooling water or service water system. In PWR's, the RHR system is also used to fill, drain, and remove heat from the refueling canal during refueling operations; to provide an auxiliary pressurizer spray; and to circulate coolant through the core during plant startup prior to RCS pump operation.

The RHR system in BWR's is typically composed of four subsystems. The containment heat removal and low pressure emergency core cooling subsystems are discussed in SRP 6.2.2 and 6.3. The shutdown cooling and steam condensing (via RCIC) subsystems are covered by this plan. These subsystems make use of the same hardware, consisting of pumps, piping, heat exchangers, valves, monitors, and controls. In the shutdown cooling mode, the BWR RHR system can also be used to supplement spent fuel pool cooling. As in the PWR, the low pressure RHR piping is protected from high RCS pressure by isolation valves.

The steam condensing mode of RCIC operation in BWR's (when included in the plant design) provides an alternative to the main condenser or normal RCIC mode of operation during the initial cooldown. Steam from the reactor is transferred to the RHR heat exchangers where it is condensed. The condensate is piped to the suction side of the RCIC pump. The RCIC pump returns the condensate to the reactor vessel via the feedwater line. The heat removed in the heat exchangers is transported to the ultimate heat sink by the service water system.

The RSB reviews the design and operating characteristics of the RHR system with respect to its shutdown and long term cooling function. Where the RHR system interfaces with other systems (e.g., RCIC system, component cooling water system) the effect of these systems on the RHR system is reviewed. Overpressure protection provided by the valving between the RCS and RHR system is also reviewed.

The proposed preoperational and initial startup test programs are reviewed and the proposed technical specifications are evaluated in regard to limiting conditions of operation and periodic surveillance testing.

The RHR system is reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as a portion of the effort described in SRP 3.2.1 and 3.2.2. The RHR system is to be enclosed in a structure having the proper seismic classification. The review is done as a part of the effort described in SRP 3.2.2.

The RHR system is to be housed in a structure that provides adequate protection against wind, tornadoes, floods, and missiles (as appropriate). The review of the adequacy of this enclosure is performed as described in other standard review plans.

The APCSB reviews the component cooling or service water systems as described in SRP 9.2.1 and 9.2.2.

The CSB, as described in SRP 6.2.4, reviews the design of the RHR system to see that it is compatible with the function of the containment and that adequate isolation capabilities are provided.

The EICSB, as described in SRP 7.4, reviews motor-operated valve controls, interlocks, sensors for interlocks, position indicators, and power sources. EICSB determines that the interlocks on motor-operated valves used as barriers between the high and low pressure RHR piping are suitable independent and diverse and that trip signals close the valves when the pressure is too high.

The MEB, as described in SRP 3.9.3, reviews the design and installation of the RHR system to see that applicable code requirements are met.

The MTEB reviews the materials and inservice inspection program for the RHR system, as described in SRP 6.1.1 and 6.6.

The CPB reviews the core decay energy output on which the design is based to see that it is applicable and suitably conservative.

The MEB and APCSB review the effects of pipe breaks both in and outside containment on reactor shutdown systems. This review includes the effects of pipe whip, jet impingement forces, and any environmental conditions created. The effect of missiles on the RHR system is also reviewed by these branches.

II. ACCEPTANCE CRITERIA

The general objective of the review is to determine that the RHR system meets the requirements of General Design Criterion (GDC) 34 (Ref. 4) concerning shutdown and long term cooling and GDC 61 (Ref. 11) concerning cooling during refueling. The RHR system must be capable of removing decay and residual heat from the core after the initial phase of cooldown so as to preclude fuel damage.

The integrated design of the RHR system including pumps, heat exchangers, valves, tanks, piping, and system enclosure must be in accordance with GDC 2 (Ref. 1) and GDC 4 (Ref. 2), and should conform to the recommendations of Regulatory Guide 1.29 (Ref. 12), Regulatory Guide 1.46 (Ref. 13), and the staff positions on protection against piping failures outside containment (Ref. 15). The RHR system should meet the single failure criterion.

Interfaces between the RHR system and RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5 (Ref. 3). Component cooling and service water systems removing heat from the RHR heat exchangers must conform to GDC 44, 45, and 46 (Refs. 5, 6, and 7). Containment isolation provisions for the RHR system must conform to GDC 55, 56, and 57 (Refs. 8, 9, and 10).

It must be shown that adequate equipment, control, and sensing information is available to allow the operator to properly execute any required manual operations during operation or test.

The preoperational and initial startup test programs should meet the intent of Regulatory Guide 1.68 (Ref. 14).

All connections between the RCS and RHR systems should be blocked by two independent and redundant barriers whenever the RCS pressure is above the RHR design pressure. The acceptance criteria concerning this feature are as follows:

1. At least two valves in series shall be provided to isolate the RHR system whenever the primary system pressure is above the pressure rating of the RHR system.

2. For systems where both valves are motor operated, the valves should have independent and diverse interlocks to prevent the valves from being accidentally opened unless the primary system pressure is below the RHR system design pressure. The valves should also receive a signal to close automatically whenever the primary system pressure exceeds the RHR system design pressure.

3. For those systems where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent valve opening whenever the primary pressure is above the RHR system design pressure, and to close automatically whenever the primary system pressure exceeds the RHR system design pressure.

4. For those systems where two check valves are provided, continuous or frequent periodic (e.g., annual) checking should be done to assure that neither check valve allows backflow leakage.

5. Suitable valve position indication should be provided for the above valves in the control room.

In addition to the above criteria, the acceptability of the RHR system may be based on the degree of design similarity with previously approved plants.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the reviewer to determine that the acceptance criteria of Section II have been met. These steps should be adapted to CP or OL reviews as appropriate.

1. Using the description given in Section 5.4.7 of the applicant's safety analysis report (SAR), including component lists and performance specifications, the reviewer determines that the RHR system piping and instrumentation are such as to allow the system to operate as intended, with or without offsite power and given any single active component failure. This is accomplished by reviewing the piping and instrumentation diagrams (PID's) to confirm that piping arrangements permit the required flow paths to be achieved and that sufficient process sensors are available to measure and transmit required information. A failure modes and effects analysis (or similar system safety analysis) provided in the SAR is used to determine conformance to the single failure criterion.

2. Using the comparison tables of SAR Section 1.3, the RHR system is compared to designs and capacities of such systems in similar plants to see that there are no unexplained departures from previously reviewed plants. Where possible, comparisons should be made with actual performance data from similar systems in operating plants.

3. Using the system process diagrams, PID's, failure modes and effects analysis, and component performance specifications, the reviewer determines that the RHR system has the capacity to remove the core decay heat load following the initial cooldown phase, given a single active component failure and with either onsite or offsite electric power available. The reviewer consults with the CPB to confirm that the proper core decay energy output was assumed for the analysis.

4. The reviewer checks the PID's to see that essential RHR system components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III). Based on statements made in SAR Section 5.4.7 or on the reviews made by other branches the RSB reviewer confirms that the RHR system meets the requirements of GDC 2 and 4, and conforms to the recommendations of Guides 1.29 and 1.46 and the staff positions on piping failures outside containment.

5. By reviewing the piping arrangement and system description of the RHR system, the reviewer confirms that the RHR system meets the requirements of GDC 5 concerning shared systems.

6. The RSB reviewer contacts the APCSB reviewer in conjunction with his review of the RHR system heat sink and refueling system interaction to interchange information and assure that the reviews are consistent in regard to the interfacing parameters. For example, the APCSB review determines the maximum service or component cooling water temperature. The RSB reviewer then reviews the RHR system description to determine that this maximum temperature has been allowed for in the RHR system design.

7. From the system description and PID's, the reviewer determines that the overpressure protection provided for the RHR system meets the acceptance criteria as to valve placement, function, and testing. The review must also show that adequate overpressure protection (e.g., relief valves) is afforded so that any single misoperation (e.g., inadvertent startup of a makeup pump) or failure will not overpressurize the RHR system. EICSB is contacted to confirm that independent and diverse interlocks and trips are provided on any motor-operated valve used for overpressure protection and that valve position indication is adequate.

8. The RSB reviewer contacts his counterpart in the EICSB to obtain any needed information from their review. Specifically, EICSB confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. The instrumentation and controls of the RHR system are to have sufficient redundancy to satisfy the single failure criterion.

9. The RSB engineer contacts his counterpart in CSB so that the information needed concerning their reviews will be interchanged.

10. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68. At the OL stage, the reviewer assures that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68).

    The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the RHR system will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the RHR system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the RHR system. If the RHR system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

11. The proposed plant technical specifications are reviewed to:

    a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

    b. Verify that the frequency and scope of periodic surveillance testing is adequate.

12. The reviewer confirms that the RHR system is housed in a structure whose design and design criteria have been reviewed by other branches to assure that it provides adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

13. The RSB reviewer provides information to other branches in those areas where the RSB has a secondary review responsibility that is not explicitly covered in steps 1-11 above. These additional areas of secondary review responsibility include:

    a. Identification of engineered safety features (ESF) and safe shutdown electrical loads, and verification that the minimum time intervals for the connection of the ESF to the standby power systems are satisfactory.

    b. Identification of vital auxiliary systems associated with the RHR system and determination of cooling load functional requirements and minimum time intervals.

    c. Identification of essential components associated with the main steam supply and the auxiliary feedwater system that are required to operate during and following shutdown.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

"The residual heat removal (RHR) system includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to remove core decay heat and provide long term core cooling following the initial phase of reactor cooldown. The scope of review of the RHR system for the _______ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHR system, his analysis of the adequacy of the criteria and bases, and the conformance of the design to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses associated with the RHR system have been reviewed and have been found to conform to Commission regulations and to applicable regulatory guides and staff technical positions. The RHR system has been found to conform to General Design Criteria 2, 4, 5, 34, 55, 56, 57 and to Regulatory Guides 1.29, 1.46, and 1.68. The system was found capable of performing its shutdown cooling functions with only onsite or offsite electrical power available, assuming the most restrictive single active component failure. It was also found that two independent and redundant barriers are always in place between the reactor coolant systems (RCS) and RHR system whenever the RCS pressure is higher than the RHR design pressure.

"The staff concludes that the design of the residual heat removal system conforms to all applicable regulations, guides, and staff positions, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. 10 CFR Part 50, Appendix A, General Design Criterion 34, "Residual Heat Removal."

5. 10 CFR Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

6. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

7. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

8. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

9. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

10. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

11. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

12. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

13. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

14. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

15. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP 3.6.2.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.4.8 REACTOR WATER CLEANUP SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary Effluent Treatment Systems Branch (ETSB)

Secondary - Auxiliary Power and Conversion Systems Branch (APCSB)
Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

At the construction permit (CP) stage of review, ETSB reviews the information in the applicant's safety analysis report (SAR) in the specific areas that follow. At the operating license (OL) stage of review, the ETSB review consists of confirming the design accepted at the CP stage and evaluating the adequacy of the applicant's technical specifications in these areas.

1. The design of components, design features which influence system availability and reliability, and interconnections with the reactor primary coolant and radwaste systems are reviewed. Fission product removal by the reactor water cleanup system (RWCS) is considered under Standard Review Plan (SRP) 11.2. The provisions for isolating the RWCS from the reactor system following liquid poison injection, holding filter and demineralizer beds in place if system flow is decreased, straining resins from return flows to the primary system, component venting, and resin transfer are reviewed.

2. The component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability to assure the system capacity will meet the reactor coolant specifications are reviewed.

3. The quality group and seismic design criteria are reviewed.

4. The instrumentation and process controls provided to ensure proper system operation and system isolation when necessary, including instrumentation for (a) automatic system isolation to prevent removal of liquid poison in the event of standby liquid control system actuation and to prevent damage to the filter demineralizer resins, and (b) monitoring impurity removal (conductivity measurements), differential pressure across pressure-sensitive components, and temperature control prior to demineralization, are reviewed. In addition, the process controls responding to these measurements to maintain operation within the established system parameters are reviewed.

Secondary reviews are provided as follows: (a) APCSB reviews the system design for pipe breaks that could incapacitate safety-related equipment. APCSB also reviews the physical separation which is provided between system components and piping to protect essential portions of the system from missiles, pipe whip, and jet impingement that may result from piping breaks. This system is normally inside containment, but the criteria for line breaks outside containment are applied to it (SRP 3.5.1); (b) MTEB reviews the material properties and the inservice inspection requirements of the portions of the system that comprise the reactor coolant pressure boundary (SRP 5.2.3); (c) EICSB reviews instrumentation, components, and power sources with respect to their capacities, capabilities, reliability, and conformance to acceptance standards (SRP 7.6); (d) RSB reviews the system drawings of portions of the RWCS that are part of the reactor coolant pressure boundary for correct identification and for capability for isolation from the remainder of the system. RSB evaluates isolation valve performance and verifies that two automatically operated isolation valves in series, or one automatically operated isolation valve and one check valve in series, or one automatically operated isolation valve and one check valve in series, physically separate essential from nonessential portions of the system. RSB also verifies that sufficient instrumentation and controls have been provided to permit plant operators to diagnose and correct system failures that could impair the condition of engineered safety features.

II. ACCEPTANCE CRITERIA

The ETSB will accept the reactor water cleanup system design if the following criteria are met:

1. The reactor water cleanup system should include the following:
   a. Provisions for automatically isolating the RWCS from the reactor primary coolant system in the event the liquid poison system is actuated for reactor shutdown.
   b. Provisions for automatically isolating the RWCS in the event the nonregenerative heat exchanger effluent temperature exceeds the prescribed resin operating temperature for the cleanup demineralizer resins.
   c. Means for automatically maintaining flow through filter demineralizer beds in the event of low process flow or loss of process flow through the system, to prevent bed loss. The recirculation loop and holding pump subsystem provided for precoating can serve this purpose if it is activated on loss of flow or low flow conditions.
   d. Means of transferring resins. Sight glass provisions (bull's eyes) are acceptable for monitoring resin transfers. Systems should be designed to prevent "resin traps" in sluice lines. A statement indicating that consideration will be given in the design to avoid resin traps, e.g., a statement that resin transfer lines will be designed to avoid resins collecting in valves, low points, or stagnant areas, will be acceptable for transfer line designs.
   e. Provisions for venting RWCS components through a closed system, i.e., not to the immediate atmosphere. The SAR should state that vent lines run to a ventilation duct exhausting from the plant.
   f. Provision, in return lines to the reactor system or condensate system, of resin strainers capable of removing resin particles contained in demineralizer effluents.

2. The system should be capable of maintaining acceptable reactor water purity in normal operation and during anticipated operational occurrences, e.g., reactor startup, refueling, and condensate demineralizer breakthrough. The following points should be included in the system design:
   a. The system should be designed to maintain reactor water purity within the guide lines of Regulatory Guide 1.56. The system should provide demineralization of reactor water through mixed bed nuclear grade resins (beads or powdered) at approximately 1% of the main steam flow rate.
   b. The non-regenerative heat exchangers should be designed to reduce the cleanup flow temperature to the demineralizer operating temperature without the aid of the regenerative heat exchangers.
   c. The RWCS capacity should be sufficient to permit processing of surplus refueling water prior to storage in refueling water storage tanks or condensate storage tanks. Interconnections between the reactor water cleanup and liquid water systems to share the processing burden are acceptable.
   d. The RWCS should be designed to permit processing of reactor water during periods of single component failures or equipment downtime.

3. To meet the guidelines of Regulatory Guides 1.26 and 1.29, the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost drywell isolation valves should be designed to seismic Category I and Quality Group A. The remainder of the system should be designed to Quality Group C and need not be seismic Category I.

4. The RWCS should include provisions for monitoring:
   a. System effluent conductivity. Instrumentation should be consistent with Regulatory Guide 1.56.
   b. Temperature upstream of the demineralizer, to assure the ion exchange resin temperature limits are not exceeded.
   c. Differential pressure, to assure the design limits on filter/demineralizer septums and resin strainers are not exceeded.

III. <u>REVIEW PROCEDURES</u>

The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. ETSB reviews the system description and piping and instrumentation diagrams (P&IDs) to determine the processing sequence, interconnections with other systems, and similarity to systems previously evaluated, and establishes that the following are considered in the applicant's design:
   a. Provisions to automatically terminate flow to the RWCS following liquid poison injection into the reactor water.
   b. Provisions to automatically terminate flow to the cleanup demineralizers if the non-regenerative heat exchanger effluent temperature exceeds the resin operating temperature limits.

c. Provisions for automatically maintaining flow through filter/demineralizer units in the event system flow decreases to a point where the bed may drop from the septum.
d. Provisions for monitoring resin transfers to assure transfers are complete and design considerations are incorporated to eliminate resin traps.
e. Provisions for venting cleanup system components during drain, fill, and air mixing operations.
f. Provisions for removing resin particles from cleanup system product water to prevent resins from entering the reactor system.

2. ETSB reviews the system capacity and processing flexibility and considers the following:
   a. The process equipment, resin types, and bed volumes compared to those for similar reactors and the RWCS capability compared to the guidelines of Regulatory Guide 1.56.
   b. The design flows and temperatures through the system to assure the criteria for outlet temperature relative to resin temperature are met.
   c. The RWCS capability to process surplus refueling water prior to storage in the refueling water storage tanks or the condensate storage tanks. The system flow rate, surplus capacity in the liquid radwaste system if interconnections exist, and the volume of water to be processed to assure the wastes could be processed in a time which is consistent with the plant requirements, are considered.
   d. Redundant or parallel components which will permit cleanup, if required, during periods of equipment downtime or single component failures.

3. In the review of the quality group and seismic design classification of the system, ETSB compares the design to the guidelines of Regulatory Guides 1.26 and 1.29. In particular, the portion of the RWCS extending from the reactor vessel and recirculation loops to the outermost drywell isolation valves is reviewed to assure conformance to seismic Category I and Quality Group A.

4. ETSB reviews the instrumentation and controls for the reactor water cleanup system to assure that monitors are provided for:
   a. Conductivity of demineralizer effluent.
   b. Temperature and conductivity of demineralizer influent.
   c. Differential pressure across the demineralizer and across the resin strainers.

   ETSB assures that system controls are responsive to the monitor indications to maintain the required temperature and flow and that conductivity meters cover the entire range up to mandatory shutdown as delineated in the plant technical specifications in the final safety analysis report (FSAR).

IV. EVALUATION FINDINGS

ETSB verifies that sufficient information has been provided and that the review is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The reactor water cleanup system (RWCS) will be used to aid in maintaining the reactor water purity and to reduce the reactor water inventory as required by plant operations. The scope of the review of the RWCS includes the system capability to meet the anticipated needs of the plant, the capability of the instrumentation and process controls to ensure operation within limits defined in Regulatory Guide 1.56, and the seismic design and quality group classifications contained in Regulatory Guides 1.26 and 1.29. Our review has included piping and instrumentation diagrams and process diagrams along with descriptive information concerning the system design and operation.

"The basis for acceptance in our review has been conformance of the applicant's designs and design criteria to the Commission's regulations and to applicable regulatory guides, as referenced above, as well as to staff technical positions and industry standards.

"Based on the foregoing evaluation, we conclude that the proposed reactor water cleanup system is acceptable."

V. REFERENCES

1. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 2.

2. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

3. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 5.4.11 PRESSURIZER RELIEF TANK SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The pressurizer relief tank is a pressure vessel provided in typical pressurized water reactor (PWR) primary systems to condense and cool the discharge from the pressurizer safety and relief valves. Discharges from small relief valves located inside the containment may also be piped to the tank. Tank capacity is based on a requirement to absorb the pressurizer discharge during a specified step load decrease.

The review of the pressurizer relief tank, as described in the applicant's safety analysis report (SAR), includes the tank, the piping connections from the tank to the pressurizer relief and safety valves, the tank spray system and associated piping, the nitrogen supply piping, and piping leaving the tank to the cover gas analyzer and to the reactor coolant drain tank.

The review covers the following specific areas:

1. The seismic design classification of the pressurizer relief tank system.

2. The quality standards to which the system will be designed, fabricated, erected, and tested.

3. The ability of the system to withstand a single active component failure without loss of function.

4. The measures taken in the design to prevent system performance degradation below acceptable levels as a result of failures of other nearby systems.

5. The steam condensing capacity of the tank compared to the largest anticipated plant step load decrease.

6. The instrumentation provided to measure and indicate pressurizer relief tank pressure, temperature, and liquid level, and to signal the operator in the event of high or low parameter levels.

7. The tank rupture disk relief capacity compared to the capacity of the pressurizer relief and safety valves.

8. The proposed technical specificions, for operating license applications, as they relate to areas covered in this plan.

The review of the pressurizer relief tank system will involve secondary reviews performed by other branches. The results of these reviews are used by APCSB to complete overall evaluation of the system. The evaluations performed by others are as follows: the RSB will determine that the anticipated and maximum pressurizer relief and safety valve discharge rates are acceptable based on a review of the limiting transient and will determine that the piping between the valves and the tank is adequately sized. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatability of the materials of construction with service conditions. The MEB will review the transient fluid-induced load applied to the piping downstream of safety or relief valves. In addition, MEB will review the seismic qualification testing and operability of components and confirm that the system is designed in accordance with applicable codes and standards. The CSB will evaluate the blowdown, vent clearing, and condensing capabilities of the system for normal, anticipated, and maximum system flow rates upon request from APCSB. In addition, CSB will, upon request, determine the containment pressure response in the event the rupture disks are blown. The EICSB will determine the adequacy of the design, installation, inspection, and testing of essential electrical components.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the pressurizer relief tank system as described in the SAR, including related sections of SAR Chapters 2 and 3, is based on specific general design criteria and regulatory guides. An additional basis for determining the acceptability of the system is the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience. Listed below are specific criteria related to the pressurizer relief tank system.

The design of the pressurizer relief tank system is acceptable if the integrated system design is in accordance with the following criteria:

1. The rupture disks have a relief capacity at least equal to the combined capacity of the pressurizer relief and safety valves.

2. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or anticipated abnormal occurrences. The initial temperature of water inside tank should be assumed to be no lower than 120°F.

3. The pressurizer relief tank system is designed for pressures ranging from full vacuum to the disk rupture pressure setting and its corresponding saturation temperature with for rupture disc tolerance.

4. The pressurizer relief tank system may be classified as non-Seismic and Quality Group D.

5. High temperature, high pressure, high and low liquid level alarms for the pressurizer relief tank have been provided.

III. REVIEW PROCEDURES

The procedures below are used in the construction permit (CP) review to determine that the design criteria and bases and the preliminary design described in the SAR meet the acceptance criteria given in Section II of this plan. For operating license (OL) reviews, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design.

The review for OL's includes a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements developed as a result of the staff's review. The reviewer will select and emphasize material from the paragraphs below, as appropriate for a particular case.

1. The SAR is reviewed to establish that the pressurizer relief tank system description and related diagrams clearly delineate system operation and the system capability to accept the steam flow released from the pressurizer for step load decreases. The reviewer examines the adequacy of the design in terms of the seismic design classification (Regulatory Guide 1.29), quality group classification (Regulatory Guide 1.26), and conformance with industry standards. Where necessary, the review will include the requirements for system testing, minimum performance, and surveillance requirements.

2. The SAR is reviewed to determine that the rupture disks on the relief tank have a relief capacity at least equal to the combined capacity of the pressurizer relief and safety valves. The reviewer determines that the tank design pressure provides a conservative margin above the calculated pressure resulting from the maximum design relief and safety valve discharge, i.e., the maximum surge resulting from complete loss of load. The tank and rupture disks should be designed for full vacuum, so as to prevent tank collapse if the contents are cooled following a discharge without nitrogen being added.

3. The pressure suppression capability of the system is reviewed to assure proper system operation. This aspect of the review is similar to the evaluation of the vent clearing and vent flow model for pressure suppression containment systems. The review includes such effects as dynamic loadings and oscillatory behavior of the steam slug in the discharge line. The RSB will verify the mass and energy blowdown data to evaluate the above effects.

4. The piping and instrumentation diagrams are reviewed to verify that high temperature and pressure alarms and high and low liquid level alarms have been provided for the pressurizer relief tank.

5. The reviewer verifies that the system has been designed so that the system function can be maintained as required in the event of a loss of offsite power.

6. The reviewer verifies that the system will function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the failure modes and effects analysis presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR information contains verification that minimum system flow and heat transfer requirements are met for each degraded situation over the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

7. The reviewer determines that failure of portions of the system or of other systems not designed to seismic Category I standards and located close to the system, or of non-seismic Category I structures that house, support, or are close to the pressurizer relief tank system will not preclude essential operations. Reference to the general arrangement and layout drawings for structures and systems will be necessary.

8. The reviewer determines that the system is protected from the effects of high energy line breaks and moderate energy leakage cracks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to the pressurizer relief system, or that protection from the effects of failure will be provided. The means of providing such protection will be described in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding review plans.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The pressurizer relief tank system includes components and piping such as the pressurizer relief and safety valve connections to the tank, the relief tank spray system piping, the nitrogen supply piping, and piping leaving the tank to the cover gas analyzer and reactor coolant drain tank. The scope of review of the pressurizer relief tank system for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and for supporting systems essential to its safe operation. [The review has included the applicant's proposed design criteria and design bases for the pressurizer relief tank system, the adequacy of those criteria and bases, and the requirements for performance of safety-related functions of the system during normal, abnormal, and accident conditions. (CP)] [The review has included the applicant's analysis of the manner in which the design of the pressurizer relief tank and supporting systems conform to the proposed design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the pressurizer relief tank and supporting systems to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry standards.

"The staff concludes that the design of the pressurizer relief tank system conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

2. Regulatory Guide 1.29, "Seismic Design Classification."

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SECTION 6.1.1 ENGINEERED SAFETY FEATURES METALLIC MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criteria 35, 38, and 41 of Appendix A of 10 CFR Part 50 require that emergency core cooling systems, containment heat removal systems, and containment atmosphere cleanup systems shall be provided as engineered safety features.

The following areas relating to general materials considerations in the design of these engineered safety features (ESF) and the chemistry of ESF coolants are reviewed:

1. Materials Selection and Fabrication

The materials selection and fabrication procedures used in the engineered safety features are reviewed. These systems include containment heat removal systems, containment air purification and cleanup systems, emergency core cooling systems (ECCS), and other ESF specific to an individual plant, as described in Section 6 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. The specific areas of review and review procedures are similar to those in Standard Review Plan 5.2.3, "Reactor Coolant Pressure Boundary Materials." The basis of the review is to assure compliance with Section III of the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), control of welding procedures, control of the use of sensitized stainless steels, and compatibility of materials with the specific coolants used.

2. Composition and Compatibility of Containment and Core Spray Coolants

The composition of the containment and core spray coolants must be controlled to ensure their compatibility with materials in the containment building, including the reactor vessel, reactor internals, primary piping, and structural and insulating materials. These controls must be selected to maintain the integrity of the reactor coolant pressure boundary by preventing stress-corrosion cracking of safety-related components, and to prevent evolution of excessive amounts of hydrogen within the containment in the unlikely event of a loss-of-coolant accident.

Containment and core spray solutions containing boron for reactivity control and other additives (such as thiosulfates) for reacting with gaseous fission products must be stable under long term storage conditions and during prolonged operation of the sprays.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Materials Selection and Fabrication

   Materials for use in ESF must be selected for their compatibility with core and containment spray solutions, as described in Section III of the Code, Articles NB-2160, and NB-3120. Mechanical properties must be as given in Appendix I to Section III of the Code, or parts A, B, and C of Section II of the Code, except that cold-worked austenitic stainless steels must have a maximum 0.2% offset yield strength of 90,000 psi, to minimize the probability of stress-corrosion cracking in these systems, as described in Reference 9.

   Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptable criteria for preventing intergranular corrosion of stainless steel components of the ESF. Furnace-sensitized material should not be allowed in the ESF, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling and storage, for testing materials prior to fabrication, and for ensuring that no deleterious sensitization occurs during welding.

   Regulatory Guide 1.31, "Control of Stainless Steel Welding," describes acceptable criteria for assuring the integrity of welds in stainless steel components of the ESF. The control of delta ferrite content of weld filler metal as specified in this guide is modified by the Branch Technical Position MTEB 5-1 (Ref. 10), which sets forth an acceptable basis for delta ferrite content of weld filler metal.

   The composition of nonmetallic thermal insulation for austenitic stainless steel components of ESF (if thermal insulation is used) should be controlled as described in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Concentrations of leachable contaminants and added inhibitors should be controlled as specified in position C.2.b and Figure 1 of this guide to minimize the probability of stress-corrosion cracking of these components.

2. Composition and Compatibility of Containment and Core Spray Coolants

   The compositions of containment spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as given in the Branch Technical Position MTEB 6-1, Reference 11, attached. Experience has shown that maintaining the pH of borated solutions at this level will inhibit initiation of stress-corrosion cracking of austenitic stainless steel components for periods of more than seven months.

Hydrogen release within the containment because of corrosion of materials by the sprays in the event of a loss-of-coolant accident (LOCA) should be controlled as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." As the pH increases over 7.5, the rate of corrosion of aluminum increases. The amount of aluminum within the containment should therefore be controlled, and the amount of hydrogen that could be generated within the containment should be calculated as recommended in Regulatory Guide 1.7.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

To ascertain that the acceptance criteria given in Section II are met, the reviewer examines each of the review areas given in Section I for the required information, using the following procedure:

1. Materials Selection and Fabrication

   The reviewer compares the mechanical properties of the materials proposed for the ESF for their compliance with Appendix I of Section III of the Code, or with parts A, B, and C of Section II of the Code. He verifies that cold-worked austenitic stainless steels used in fabrication of the ESF are in conformance with Section II.1.

   The methods of controlling sensitized stainless steel in the ESF systems are examined by the reviewer and compared with the positions listed in Regulatory Guide 1.44, especially regarding cleaning and protection from contamination during handling and storage, verification of nonsensitization of the materials, and qualification of welding procedures using ASTM A-262-70 (Ref. 3). If alternative methods of testing the qualification welds for degree of sensitization are proposed by the applicant, the reviewer determines if these are satisfactory, based on the degree to which the alternate methods provide the needed results and on MTEB positions taken on previous applications. If necessary, the reviewer asks the applicant to justify technically his departures from the MTEB positions. Alternative tests that have been accepted by the MTEB include the use of ASTM A-393 (Ref. 4), for determining the degree of sensitization of the heat-affected zones (HAZ) of qualification welds, and the use of ASTM A-262-70 as amended by Westinghouse Process Specification 84201 MW (Ref. 5), for qualifying welds and testing raw materials for sensitization.

   The methods for controlling and measuring the amount of delta ferrite in stainless steel weld deposits are examined by the reviewer and compared with Regulatory Guide 1.31, "Control of Stainless Steel Welding," and the Branch Technical Position MTEB 5-1, especially regarding the acceptance procedures for delta ferrite content of filler metal and the examination of production welds. If alternative positions are proposed by the applicant, the reviewer determines if these are satisfactory, taking into consideration positions taken on previous applications. If necessary, the reviewer asks the applicant to justify technically his departures from the acceptance criteria stated in Section II.2.

The reviewer determines whether nonmetallic thermal insulation will be used on austenitic stainless steel components of the ESF, and if it is, he verifies that the amount of leachable impurities in the specified insulation lie within the "acceptable analysis" area of Figure 1 of Regulatory Guide 1.36, as discussed in the acceptance criteria, Section II.1.

The reviewer examines the information on the compatibility of the ESF materials of construction with the proposed ESF coolants to verify that all materials used are compatible with the coolants, as required by Articles NB-2160 and NB-3120 of Section III of the Code. The reviewer considers the composition of the sprays and any mixing processes that might occur during operation of the sprays.

2. Composition and Compatibility of Containment and Core Spray Coolants

The reviewer determines that the coolant sprays will have a minimum pH of 7.0 and reviews the methods of ascertaining that the pH will remain above this minimum during the operation of the sprays. In many instances, the ESF coolant solutions are stored in more than one form (such as a boric acid solution and a sodium hydroxide solution) and mixed only when the ESF are called upon to operate during an emergency. In some plants, the coolant is stored as a boric acid solution which is neutralized by (dry) sodium phosphates mounted in baskets inside the containment after the ESF sprays are activated. Consequently, the reviewer must examine the control of pH of such coolants, to evaluate the short-term (during the mixing process) compatibility and long-term compatibility of these sprays with all safety-related components within the containment.

The applicant's estimate of the amount of hydrogen generated within the containment by corrosion of materials is evaluated by the reviewer for conformance with Section II.2. He pays particular attention to the hydrogen generated by the corrosion of aluminum if the pH of the coolant is above 7.5. The review verifies that this estimate is realistic and conservative using the calculation methods outlined in the guide.

The reviewer examines the methods of storing the ESF coolants to determine whether deterioration will occur either by chemical instability or corrosive attack on the storage vessel. The reviewer determines what effects such deterioration could have on the compatibility of these ESF coolants with both the ESF materials of construction and the other materials within the containment.

3. General

If the information contained in the safety analysis reports or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information but also the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the acceptance criteria.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The mechanical properties of materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Code, or Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi.

"The controls on the pH of the reactor containment sprays and the emergency core cooling water following a postulated loss-of-coolant accident are adequate to ensure freedom from stress-corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling systems throughout the duration of the postulated accident to completion of cleanup. The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress-corrosion cracking will not occur during the postulated accident time interval. The control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution by corrosion of containment metal or cause serious deterioration of the containment. (The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the engineered safety features are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel.")* Conformance with the Codes and Regulatory Guides mentioned above, and with the staff positions on the allowable maximum yield strength of cold-worked austenitic stainless steel, and the minimum level of pH of containment sprays and emergency core cooling water constitute an acceptable basis for meeting applicable requirements of General Design Criteria 35, 38, and 41."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling," Criterion 38, "Containment Heat Removal," and Criterion 41, "Containment Atmosphere Cleanup."

2. ASME Boiler and Pressure Vessel Code, Section II, Parts A, B, and C, and Section III, Articles NB-2160 and NB-3120, and Appendix I, American Society of Mechanical Engineers.

*The sentence in parenthesis is to be included only if nonmetallic thermal insulation is to be used on ESF piping.

3. ASTM A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

4. ASTM A-393-63, "Recommended Practice for Conducting Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

5. Process Specification 84201 MW, "Corrosion Testing of Wrought Austenitic Stainless Steel Alloy," Westinghouse Electric Corporation.

6. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

7. Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

8. Regulatory Guide 1.44, "Control of the Use of Sensitized Steel."

9. Standard Review Plan 5.2.3,"Reactor Coolant Pressure Boundary Materials."

10. Branch Technical Position MTEB 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'," appended to Standard Review Plan 5.2.3.

11. Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water," appended.

BRANCH TECHNICAL POSITION MTEB 6-1

pH FOR EMERGENCY COOLANT WATER

A. Background

In response to a Technical Assistance Request, dated April 20, 1972, needed to establish the minimum value of pH in post-accident containment sprays for the Fort Calhoun Station, the Materials Engineering Branch, reviewed the available information and recommended the criteria listed in the Branch Technical Position, below.

The minimum pH value of 7.0 follows from the Westinghouse report (Ref. 1) conclusion that in ECCS solutions adjusted with NaOH to pH 7.0 or greater, no cracking should be observed at chloride concentrations up to 1000 ppm during the time of interest. Figure 7 of the Westinghouse report shows that time for initiation of cracking of sensitized and nonsensitized U-bend specimens of Type 304 austenitic stainless steel in solutions of 7.0 pH having 100 ppm chloride was seven and one half months and ten months, respectively. These time periods are more than ample time to allow cleanup after the hypothetical design basis accident.

The great majority of tests reported in the Oak Ridge report, Reference 2, were performed with pH of 4.5, and only two tests were conducted with pH's other than 4.5. Some cracking was observed at pH 7.5 in the sensitized 304 stainless steel U-bend specimens after two months exposure to pH = 7.5 and chloride concentration of 200 ppm. All of the 316 stainless steel specimens showed no evidence of cracking. Considering the fact that in U-bend specimens the material was sensitized, stressed beyond yield, and plastically deformed, we conclude that the reported test conditions were much more severe than the stress conditions likely to exist in the post-accident emergency coolant systems.

We agree with the Oak Ridge conclusion that absolute freedom from failure of any complex system such as a spray system can never be guaranteed, but by proper design, fabrication, and control of the corrosive environment, the probability of failure can be significantly reduced. Our recommended minimum pH of 7 is somewhat higher than Oak Ridge recommendation of a minimum of 6.5.

B. Branch Technical Position

MTEB criteria for pH level of post-accident emergency coolant water to minimize the probability of stress-corrosion cracking of austenitic stainless steel components, non-sensitized or sensitized, not stressed or stressed, are as follows:

1. Minimum pH should be 7.0.

2. The higher the pH (in the 7.0 to 9.5 range) the greater the assurance that no stress corrosion cracking will occur.

3. If a pH greater than 7.5 is used, consideration should be given to the hydrogen generation problem from corrosion of aluminum in the containment.

C. References

1. D. D. Whyte and L. F. Picone, "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environment," WCAP-7798-L, Westinghouse Nuclear Energy Systems, November 1971 (NES Proprietary Class 2).

2. J. C. Griess and G. E. Creek, "Design Considerations of Reactor Containment Spray Systems - Part X, The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions," ORNL-TM-2412, Part X, Oak Ridge National Laboratory, May 1971.

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SECTION 6.1.2 ORGANIC MATERIALS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - None

I. AREAS OF REVIEW

1. The coating systems (paints) used inside the containment are evaluated as to suitability for design basis accident (DBA) conditions.

2. The stability of materials (particularly organics) and their decomposition products are examined to determine the potential for interactions with engineered safety features (ESF), such as filters (poisoning). Radiation and chemical effects are considered. (Physical effects are considered by the Containment Systems Branch.)

II. ACCEPTANCE CRITERIA

A coating system is acceptable if:

1. It meets Regulatory Guide 1.54 (Ref. 1) or equivalent; or, the area covered with the system is a negligible fraction of the containment interior surfaces.

2. No adverse interactions with engineered safety features are likely as a result of materials released by radiation decomposition or chemical reaction of the coating system in the containment post-accident environment.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on the areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether or not it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

A detailed evaluation of every paint and organic material found in the containment should not be attempted. The "significant" paints and organic materials are reviewed, where significance is judged by the rad-gram exposure level (i.e., by the product of the estimated DBA unit radiation dose and the mass of the particular coating) for both coatings and plastics, and by the possibility of clogging sump screens for coatings.

1. Review of coating systems:

   a. The reviewer verifies that each significant coating system meets the recommendations of Regulatory Guide 1.54 (Ref. 1).

   b. The reviewer verifies that any information supplied regarding methods of application corresponds to manufacturer's recommendations.

   c. The reviewer confirms that the quantities of unidentified coatings are insignificant.

2. Review of gases and soluble materials released by coating systems:

   a. The radiation levels in the containment are estimated (see Standard Review Plan 3.11.5). The reviewer estimates the quantities and types of materials released due to the radiation exposure, and verifies that the decomposition products cannot adversely affect any engineered safety feature system. The generation of methane or other volatile alkanes to form a source for organic iodides is of specific concern (Ref. 2).

   b. Chemical effects are also considered in the potential for material release. This area is reviewed on a case-by-case basis (Ref. 3).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following types, to be included in the staff's safety evaluation report:

"The containment coating systems have been evaluated as to their suitability to withstand a postulated design basis accident (DBA) environment. The coating systems chosen by the applicant have been qualified under conditions which take into account the postulated DBA conditions. No adverse interactions (under DBA conditions) between the decomposition products and the engineered safety features have been established. The amount of unqualified paint in the containment is not significant. The staff concludes, therefore, that the coating system will remain intact under postulated design basis accident conditions."

V. REFERENCES

1. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

2. ANSI N4.1, "Polymeric Materials for Service in Ionizing Radiation, Classification System for," American National Standards Institute (1973).

3. ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," American National Standards Institute (1972).

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SECTION 6.1.3 POST-ACCIDENT CHEMISTRY

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The methods and procedures used to control the chemical composition of solutions recirculated within containment after design basis accidents (DBA) are reviewed to assure that adverse chemical reactions or inadequate solution mixing will not occur.

II. ACCEPTANCE CRITERIA

The procedures and methods which the applicant proposes to use to raise or maintain the pH of the solutions expected to be recirculated within containment after a DBA should be straightforward and reliable. The chemistry of the post-accident environment in the containment should not result in significant deterioration of engineered safety features.

III. REVIEW PROCEDURES

The purpose of controlling the pH is to reduce the probability of chloride stress corrosion cracking leading to equipment failure or loss of containment integrity, and to ensure low volatility of dissolved radioiodines. These purposes are met by maintaining a high pH, at least 7 (Ref. 1 and 2), but not high enough to cause any substantial attack on aluminum fittings. A number of plants have used NaOH added to the containment spray solution, or solid trisodium phosphate placed in baskets on the containment lower level where it can dissolve in the recirculated water in the event of a DBA.

Guidance as to allowable pH histories should be obtained from the Materials Engineering Branch. At present, available information indicates optimum pH control consists of stabilizing pH between 7 and 8 within four hours (Ref. 3).

The reviewer examines the paths which solutions would follow in the containment from sprays and emergency core cooling systems to the sump, for both injection and recirculation phases, to verify that no areas accumulate very high or low pH solutions and that any assumptions regarding pH in the modeling of containment spray fission product removal are valid (see Standard Review Plan 6.5.2).

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The methods and procedures for controlling the pH of solutions expected to be recirculated in containment following design basis accidents have been found adequate. The proposed control provides assurance that the pH will be maintained at a level which minimizes the possibility of stress corrosion cracking of mechanical systems and components."

V. REFERENCES

1. D. D. Whyte and L. F. Picone, "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Environments," WCAP-7798-L (proprietary), Westinghouse Electric Corporation, November 1971.

2. J. C. Greiss and G. E. Creek, "Design Considerations of Reactor Containment Spray Systems - Part X, The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions," ORNL-TM-2412, Oak Ridge National Laboratory, May 1971.

3. R. Zavadoski, "Stress Corrosion Cracking and pH for the Fort Calhoun Station," regulatory staff memorandum to K. Goller, April 7, 1972.

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SECTION 6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Reactor Systems Branch (RSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Auxiliary and Power Conversion Systems Branch (APCSB)

INTRODUCTION

The CSB reviews information regarding the functional capability of the reactor containment presented in Section 6.2.1 of the applicant's safety analysis report (SAR). The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant and steam or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a loss-of-coolant accident (LOCA). The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the emergency core cooling system cools the reactor core to prevent any extensive fuel melting.

The evaluation of a containment functional design includes calculation of the progress of a LOCA event after an instantaneous rupture is assumed to occur in some section of the primary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy in the primary system are determined from a solution of the incremental space and time-dependent energy, mass, and momentum equations. The basic functional design requirements for containment are given in General Design Criteria 16 and 50 in Appendix A to 10 CFR Part 50. General Design Criterion 50, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the emergency core cooling system, the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters, in assessing containment design margins.

There are a number of different containment types and designs, and several aspects of containment functional design that are within the scope of SAR Section 6.2.1. The various containment types and aspects to be reviewed under this plan have been separated and assigned to a set of "subplans" as follows:

a. Pressurized water reactor (PWR) dry containments, including subatmospheric containments (SRP 6.2.1.1.A).
b. Ice condenser containments (SRP 6.2.1.1.B).
c. Mark I, II, and III boiling water reactor (BWR) pressure-supression type containments (SRP 6.2.1.1.C).
d. Subcompartment analysis (SRP 6.2.1.2).
e. Mass and energy release analysis for postulated loss-of-coolant accidents (SRP 6.2.1.3).
f. Mass and energy release analysis for postulated secondary system pipe ruptures (SRP 6.2.1.4).
g. Minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability studies (SRP 6.2.1.5).

A separate standard review plan (SRP) has been prepared for each of these areas.

Areas related to the evaluation of the containment functional capability are treated in other standard review plans; e.g., containment heat removal (SRP 6.2.2), combustible gas control (SRP 6.2.5), and containment leakage testing (SRP 6.2.6).

I. AREAS OF REVIEW

The items reviewed are described in the "Areas of Review" sections of the seven "subplans" listed above.

II. ACCEPTANCE CRITERIA

The acceptance criteria are given in the "Acceptance Criteria" sections of the seven "subplans" listed above.

III. REVIEW PROCEDURES

Review procedures are given in "Review Procedures" sections of the seven "subplans" listed above.

IV. EVALUATION FINDINGS

The results of the reviews under the seven "subplans" listed above are consolidated into a single set of findings. The reviewer verifies that sufficient information has been provided and that his evaluation is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

"Containment Functional Design

The scope of review of the functional design of the containment for the ABC nuclear power plant has included a review of plant arrangement drawings, system drawings, and descriptive information for the containment building, subcompartments, and associated systems, components, and structures that are essential to the functional capability and integrity of the containment. The review has included the applicant's proposed design bases for the containment building and internal structures, and associated structures and systems upon which the containment function depends, and the applicant's analysis of postulated accidents and operational occurrences which support the adequacy of the design bases.

"The basis for the staff's acceptance has been conformance of designs and design bases for the containment building, internal structures, and associated systems, components, and structures to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of containment structures, systems, and components should be discussed.)

"The staff concludes that the containment functional design conforms to applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design;" Criterion 39, "Inspection of Containment Heat Removal System;" Criterion 40, "Testing of Containment Heat Removal System;" Criterion 50, "Containment Design Basis;" Criterion 54, "Systems Penetrating Containment;" and Criterion 56, "Primary Containment Isolation."

2. 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

3. ASME Boiler and Pressure Vessel Code, Section II, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

4. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

5. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

6. Regulatory Guide 1.29, "Seismic Design Classifications and Standards."

7. C. F. Carmichael and S. A. Marks, "CONTEMPT-PS, A Digital Computer Code for Predicting the Pressure-Temperature History Within a Pressure-Suppression Containment Vessel in Response to a Loss-of-Coolant Accident," IDO-17252, Phillips Petroleum Company, April 1969.

8. L. C. Richardson, L. J. Finnegan, R. J. Wagner, and J. M. Waage, "CONTEMPT, A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

9. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

10. R. I. Miller, "Evaluation of the Predictive Capabilities of the CONTEMPT-PS Computer Code by Comparison of Calculated Results with the Humboldt Bay and Bodega Bay Pressure Suppression Tests," Interim Report 4.2.1.1, Idaho Nuclear Corporation, September 1970.

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12. R. C. Schmitt, G. E. Bingham, and J. A. Norberg, "Simulated Design Basis Accident Test for the Carolina Virginia Tube Reactor Containment - Final Report," IN-1403, Idaho Nuclear Corporation, December 1970.

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STANDARD REVIEW PLAN

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SECTION 6.2.1.1.A PWR DRY CONTAINMENTS, INCLUDING SUBATMOSPHERIC CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Eningeering Branch (SEB)

I. AREAS OF REVIEW

For pressurized water reactor (PWR) plants with dry containments, the CSB review covers the following areas:

1. The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and secondary system steam line and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The minimum containment pressure used in analyses of emergency core cooling system capability.

4. The effectiveness of static and active heat removal mechanisms.

5. The pressure conditions within subcompartments and acting on system components and supports due to high energy line breaks.

6. The instrumentation provided to monitor and record containment atmosphere pressure and temperature and sump water temperature under post-accident conditions.

7. The proposed technical specifications at the operating license stage of review pertaining to the surveillance requirements for spring or weight loaded check valves used in subatmospheric containments, and vacuum relief devices.

The CSB will also review analyses of anticipated transients without scram (ATWS) which discharge fluid to the containment to assure that containment pressure and temperature design conditions are not exceeded.

Future applications for boiling water reactor (BWR) plants may include a dry containment design. When such a proposal is made, the CSB will review the containment design on the basis of the review plan described herein.

II. ACCEPTANCE CRITERIA

The following acceptance criteria complement General Design Criterion 50 and apply to the design and functional capability of PWR dry containments:

1. For plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break.

2. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin.

3. The containment pressure should be reduced to less than 50% of the containment design pressure within 24 hours after the postulated accident, as recommended by Regulatory Guide 1.4.

4. For subatmospheric containments, the containment pressure should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days.

5. Containment response analyses should be based on the assumption of loss of offsite power and the most severe single active failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.

6. The minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See Standard Review Plan 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").

7. Provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems; or for steel containment vessels, vacuum relief devices provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE (Ref. 3), and applicable requirements of General Design Criteria 54 and 56.

8. If the primary containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.

9. Containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See Standard Review Plan 6.2.1.2, "Subcompartment Analysis").

10. Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of PWR dry containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of dry containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

The CSB reviews the containment response analyses to determine the acceptability of the calculated containment design pressure and temperature, and in addition, the containment depressurization time for subatmospheric type containments. The CSB reviews the assumptions made in the analyses to maximize the calcualted containment pressure. The CSB determines the conservatism of the respective containment response analyses by comparing the analytical models, and the assumptions made, with the acceptance criteria in Section II, and by performing appropriate confirmatory analyses. It is not necessary to perform accident pressure calculations for every plant. The CSB will ascertain, however, that the adequacy of the applicant's calculational model has been demonstrated. The CSB determines that the pipe break resulting in the highest containment pressure has been identified. Hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks of the reactor coolant system, and secondary system steam and feedwater line breaks, should be analyzed by the applicant. The CSB reviews the assumptions used to determine that the analyses are acceptably conservative.

The CSB performs confirmatory containment response analyses when necessary using the CONTEMPT-LT computer code (See References 7, 8, and 9 for a description of this code). If the conservatism of certain input data is in question, such as the mass and energy release rate data for the core reflood and post-reflood phases of a loss-of-coolant accident, the CSB uses data calculated using its own analytical models or obtains corrected data from the applicant. This part of the review may include coordination between the CPB and CSB (See Standard Review Plans 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated

Secondary System Pipe Ruptures"). The purpose of these analyses is to confirm the applicant's predictions of the response of the containment to loss-of-coolant accidents and main steam and feedwater line breaks. In general, only the limiting pipe breaks, i.e. the pipe breaks which establish the containment design pressure and containment depressurization time, are analyzed. However, if in the judgment of the CSB the worst break has not been identified, other pipe breaks will be analyzed.

The CSB reviews analyses of the external pressure that the containment structure may be subjected to as a result of pressure and temperature changes inside the containment due to inadvertent operation of containment heat removal systems. The CSB determines whether the most severe condition has been identified, and whether the analysis was done in a conservative manner. The CSB evaluates the acceptability of the provisions made in the plant design to mitigate or withstand the consequences of the above postulated events, and the administrative controls and instrumentation and control provisions to preclude these events.

The CSB determines whether instrumentation capable of withstanding the post-accident environment, and recording equipment, has been provided to monitor and record the course of an accident within the containment. The CSB also determines whether the instrumentation and recording equipment can accomplish the objectives stated in Section II. This review is coordinated with the EICSB. The EICSB, under Standard Review Plan 7.3, has review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of the plant protection system and the post-accident monitoring instrumentation and recording equipment.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.1.B   ICE CONDENSER CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Accident Analysis Branch (AAB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The CSB review of ice condenser containments includes the following areas:

1. The pressure and temperature conditions in the containment due to a spectrum (including break size and location) of loss-of-coolant accidents; i.e., reactor coolant system pipe breaks and steam and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The design and qualification testing of ice condenser components.

4. The pressure conditions within containment internal structures and acting on system components and supports due to high energy line breaks.

5. The maximum allowable operating deck steam bypass area for a full spectrum of reactor coolant system pipe breaks.

6. The design provisions and proposed surveillance program to assure that the ice condenser will remain operable for all plant power operations.

7. The design and qualification testing of the return air fan systems and system components.

8. The effectiveness of static and active heat removal mechanisms.

9. The minimum containment pressure used in the analyses of emergency core cooling system capability.

10. The instrumentation provided to monitor and record containment atmosphere pressure and temperature and sump water temperature under post-accident conditions.

11. The proposed technical specifications, at the operating license stage of review, pertaining to the surveillance requirements for steam bypass area, return air fan system operability, ice condenser operability, and vacuum relief devices.

The CSB will also review analyses of anticipated transients without scram (ATWS) which discharge fluid to the containment to assure that containment pressure and temperature design conditions are not exceeded.

The fission product removal capability of the ice condenser is evaluated by AAB under Standard Review Plan 6.5.4.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the design and functional capability of ice condenser containments:

1. The ice condenser components should be designed, fabricated, erected, and tested in accordance with Group B quality standards, as recommended by Regulatory Guide 1.26.

   The ice condenser components should be designated Category I (seismic); i.e., designed to withstand the effects of the safe shutdown earthquake without loss of function, as recommended by Regulatory Guide 1.29.

   Analyses or qualification tests should be performed for all ice condenser components that are changed in design from that reported in Appendices M and N to the D.C. Cook FSAR (Ref. 27) to assure that the ice condenser will remain operable in the accident environment for as long as accident conditions require. If a component was originally qualified by analytical methods, confirmation of the new design by reanalysis or a test program will be acceptable. For components that were originally qualified by a test program, the redesigned component should be requalified by a test program.

2. The containment accident pressure and temperature response should be calculated using the LOTIC (or an equivalent) computer code (Ref. 27). Conservative assumptions which maximize the energy release to the containment should also be used (See Standard Review Plan 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents").

   For plants being reviewed for construction permits, the containment design pressure should provide at least a 20% margin above the highest calculated accident pressure. For plants being reviewed for operating licenses, the highest calculated accident pressure should not exceed the design pressure of the containment.

3. Ice condenser subcompartment or control volume differential (internal) pressures should be calculated using the Transient Mass Distribution (TMD) computer code (Ref. 28), without the augmented critical flow correlation. Mass and energy releases from postulated pipe breaks should be determined using the SATAN-VI computer code (Ref. 24) and used as input to the TMD code.

For plants being reviewed for construction permits, the design differential pressures for all ice condenser control volumes or subcompartments, and system components (e.g., reactor vessel, pressurizer, steam generators) and supports, should provide at least a 40% margin above the highest calculated differential pressures. For plants being reviewed for operating licenses, the highest calculated differential pressures for all ice condenser control volumes or subcompartments should not exceed the corresponding design differential pressures.

The operating deck should be designed to withstand the maximum calculated differential pressure between the upper and lower compartments. To account for uncertainties in the analysis of reverse differential pressures, an adequate margin should be provided above the maximum calculated reverse differential pressure.

4. The maximum allowable area for steam bypass of the ice condenser should be greater than the identifiable bypass area for the plant (e.g., the drainage provisions to allow containment spray water to return from the upper compartment to the sumps in the lower compartment). The bypass area capability of the plant should be based on analyses of the spectrum of postulated reactor coolant system pipe breaks, and should be about 35 square feet or greater.

5. The design of the ice condenser system should incorporate provisions for periodic inservice inspection and testing of essential system components; e.g., the ice baskets and doors, the ice condenser temperature monitoring system, and the available mass of ice. The inspection and test program should assure the integrity and operability of the ice condenser system and should satisfy the requirements of General Design Criteria 39 and 40.

6. The return air fan system components should be designed, fabricated, erected, and tested in accordance with Group B quality standards, as recommended by Regulatory Guide 1.26. The system should be designated Category I (seismic) as recommended by Regulatory Guide 1.29.

   The inservice inspection and testing program for the return air fan system should satisfy the requirements of General Design Criteria 39 and 40.

   Analyses or tests should be performed for the return air fan system components to demonstrate that the system will remain operable in the accident environment for as long as accident conditions require.

7. Inadvertent operation of engineered safety features (e.g., the return air fan system or the containment spray system) should not cause the external design pressure of the primary containment to be exceeded. This may be accomplished through conservative containment design, use of vacuum relief devices, or electrical interlocks that preclude inadvertent operation of the spray and fan systems. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE (Ref. 3) and should meet applicable requirements of General Design Criteria 54 and 56.

8. Instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature, and the sump water temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.

9. The minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See Standard Review Plan 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").

III. REVIEW PROCEDURES

The procedures described below are followed for the review of ice condenser containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of ice condenser containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

1. The CSB evaluates the design of the ice condenser type containment by comparing it to the design information presented in Appendices M and N to the D.C. Cook FSAR, and discussed in the staff's safety evaluation report on the plant (Ref. 27). The CSB has reviewed the design of the Cook ice condenser as reported in these documents and has found that it satisfies the acceptance criteria stated in Section II. Any differences from the design reported in the Cook documents are evaluated. The CSB determines that all design changes have been justified, and the components have been requalified for use in the ice condenser by the same methods originally used to qualify them, i.e., for simple structures which were qualified by analytical methods, a reanalysis will be accepted; and for components qualified by test programs, the tests should be repeated on the revised design.

   The CSB compares the quality standards applied to the ice condenser to Regulatory Guide 1.26.

   The CSB compares the seismic design classification of the ice condenser to Regulatory Guide 1.29.

2. The CSB reviews the analysis of the containment pressure and temperature response. The CSB and CPB determine that the mass and energy release to the containment for the duration of the accident has been maximized (See Standard Review Plan 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss of Coolant Accidents"). The CSB has reviewed the LOTIC code which is used to determine the containment pressure and temperature response, and has determined that the code is acceptable for containment analysis. The CSB assures that the LOTIC code has been used and that the input assumptions to the code are conservative. Code revisions and improvements will also be considered.

CSB determines from the results of analyses that the peak calculated containment pressure does not exceed the design pressure of the containment, for plants at the operating license stage of review. For plants at the construction permit stage of review, the CSB will ascertain from the results of analyses reported in the safety analysis report that the design pressure provides a margin of at least 20% above the maximum calculated pressure.

Modifications to the CONTEMPT-LT code which will provide improved capability to analyze the response of an ice condenser containment to a loss-of-coolant accident are being made. When the COMTEMPT-LT modifications have been completed, CSB will perform confirmatory analyses using the modified code.

3. The TMD code is used to evaluate the transient pressure responses (internal) of the ice condenser containment subcompartments. The code is described in the proprietary report WCAP-8077 (Ref. 28). The TMD code utilizes ice condenser heat transfer and flow data obtained from full-scale section tests of the ice condenser. As stated in the D.C. Cook Safety Evaluation Report, the CSB has reviewed the assumptions and equations used in the TMD code and with the exception of the critical flow model used to predict subcompartment vent mass flow rates, has concluded that the TMD code conservatively calculates transient pressure response.

   The TMD code calculates the critical flow of a two-component, two-phase fluid (air, steam, and water) assuming a thermal equilibrium condition. However, a correction factor is then applied to the calculated critical flow. The CSB has not accepted the use of this corrected critical flow, referred to as "augmented flow," and has required that the short-term transient responses of subcompartments be determined using the TMD code without applying a correction factor to the critical flow; i.e., without the "augmented flow" correlation.

   Before accepting the containment transient responses calculated by the TMD code, the CSB reviews the mass and energy data input to the TMD code and the modeling of the containment subcompartments, the size and area of assumed vents between nodes, volumes of nodes, the flow loss coefficients for each vent modeled, and the heat transfer coefficients within the ice condenser.

   The CSB will determine from the safety analysis report that the TMD code, without the "augmented flow" correlation, has been utilized to determine the transient pressure response in each subcompartment that contains a high energy line, and in adjoining subcompartments.

The CSB reviews the maximum calculated differential pressures and pressure profiles for each subcompartment. For plants at the construction permit state of review, the CSB will ascertain that it is the applicant's intent to design all internal structures with a margin of 40% between the maximum calculated differential pressure and the design differential pressure of the structure or component. At the operating license stage of review, the CSB will ascertain that an appropriate margin exists. However, changes in technology and calculational methods may affect the margin. The CSB will then determine that the maximum calculated differential pressures do not exceed the design differential pressures for the internal structures. The loads on components or their supports installed within the compartment due to possible pressure gradients will be evaluated by MEB and SEB.

Modifications to the RELAP4 code to include two-phase, two-component mixtures are being made. This will improve the capability of the code so that it may be used for subcompartment analysis of ice condenser plants. When the modifications to the RELAP4 code have been completed, the CSB will use the code to conduct confirmatory analyses.

4. The CSB reviews the methods, input assumptions, and results of the applicant's steam bypass analysis. The applicant's analysis should show considerable margin between the maximum tolerable bypass leakage area and the identifiable bypass area required to allow spray water drainage back to the containment sump. The CSB determines the adequacy of the margins provided for the full spectrum of reactor coolant pipe ruptures. Factors affecting the determination include the proposed inspections and tests to determine bypass leakage area and whether the design of the plant will permit access to seals between the upper and lower compartments for inspection. At the operating license stage, the CSB reviews the proposed technical specifications to assure that adequate surveillance will be maintained for the steam bypass area.

5. The CSB reviews the initial programs for ice loading and subsequent verification of individual ice basket and total ice loads. In addition, it reviews design provisions for monitoring the status of the ice condenser during plant operation to assure that the ice condenser retains its full capability. The CSB also reviews the aspects of the ice condenser design which will allow inspection and functional testing of ice condenser components during various modes of plant operation. Specific areas to be evaluated are the ice condenser temperature instrumentation system, lower inlet door position monitoring system, proposed ice basket inspection programs to determine total ice weight, proposed inspection and testing programs for intermediate and top deck doors, floor drains, lower inlet doors, ice condenser flow passages, divider barrier seals, and access hatches. The CSB determines that the proposed surveillance programs and attendant design provisions fulfill the intent of General Design Criteria 39 and 40. At the operating license stage, the CSB also evaluates the proposed technical specifications that have been established to assure ice condenser operability.

6. The CSB reviews the environmental conditions used in the qualification testing of the return air fan system components. The CSB determines whether the test conditions are representative of post-accident conditions to which the equipment may be subjected. The CSB will ascertain that the equipment can operate in the accident environment for as long as accident conditions require. The CSB reviews analyses demonstrating that where required, the return air fan system and its components are designed to withstand the transient differential pressures to which the system would be subjected following a loss-of-coolant accident.

The CSB reviews the provisions made in the design of the return air fan system and the proposed program for periodic inspection and functional testing of the system and components for compliance with the intent of General Design Criteria 39 and 40. The CSB determines the acceptability of the proposed periodic surveillance program for the return air fan system, taking into account the extent and frequency of testing proposed and the practices established for previous ice condenser plants. At the operating license stage, the CSB also evaluates the technical specifications for the return air fan system that have been proposed to assure system operability.

7. The CSB reviews the analysis of the maximum depressurization transient due to inadvertent operation of the containment sprays or return air fans. The CSB reviews the assumed containment initial conditions, methods of calculation, and spray system efficiency to determine whether the containment depressurization analysis is conservative. If the external design pressure of the containment is shown to be exceeded, the CSB will ascertain that containment vacuum relief devices to mitigate the consequences of inadvertent operation of the sprays or fans have been provided, or administrative controls have been established and interlocks provided to prevent inadvertent operation of the sprays or fans. If containment vacuum relief devices are used, the CSB reviews the analysis provided to demonstrate that the response time of the relief devices is short enough to prevent depressurization of the containment below the external design pressure. The CSB determines that the vacuum relief devices comply with the requirements of Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. CSB reviews the design of the vacuum relief devices and proposed inspection and testing programs to ensure that the intent of General Design Criteria 54 and 56 is fulfilled. If administrative controls are established and electrical interlocks provided to preclude inadvertent operation of the sprays or fans, the CSB in conjunction with the EICSB reviews the acceptability of these provisions from a functional standpoint. At the operating license stage of review, CSB also reviews the proposed technical specifications to assure that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

8. The CSB determines whether instrumentation capable of withstanding post-accident environments, and recording equipment, has been provided to monitor and record the course of an accident within the containment. The CSB also determines that the instrumentation and recording equipment can accomplish the objectives stated in Section II. This review effort is coordinated with the EICSB. The EICSB, in Standard Review Plan 7.3, has review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of the plant protection system, the ice condenser temperature monitoring system, and the post-accident monitoring instrumentation and recording equipment.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.1.C PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)
Mechanical Engineering Branch (MEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

For Mark I, II, and III pressure-suppression type boiling water reactor (BWR) plant containments, the CSB review covers the following areas:

1. The temperature and pressure conditions in the drywell and wetwell due to a spectrum (including break size and location) of postulated loss-of-coolant accidents.

2. The differential pressure across the operating deck of Mark II plants for a spectrum of loss-of-coolant accidents (including break size and location).

3. Suppression pool dynamic effects during a loss-of-coolant accident or following the actuation of one or more reactor coolant system pressure relief valves, including vent clearing, vent interactions, pool swell, pool stratification, and dynamic and asymmetrical loads on suppression pool and other containment structures.

4. The consequences of a loss-of-coolant accident occurring within the containment (wetwell); i.e., outside the drywell (Mark III containments only).

5. The capability of the containment to withstand the effects of steam bypassing the suppression pool.

6. The external pressure capability of the drywell and wetwell, and systems that may be provided to limit external pressures.

7. The effectiveness of static and active heat removal mechanisms.

8. The instrumentation provided to monitor and record containment atmosphere pressure and temperature and pool water temperature under post-accident conditions.

9. The pressure conditions within subcompartments and acting on system components and supports due to high energy line breaks, e.g., the sacrificial shield structure.

10. The proposed technical specifications, at the operating license stage, pertaining to the surveillance requirements for steam bypass area and vacuum relief devices.

The CSB will also review analyses of anticipated transients without scram (ATWS) which discharge fluid to the containment to assure that containment pressure and temperature design conditions are not exceeded.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the design and functional capability of BWR pressure-suppression type containments:

1. For Mark I and II plants at the operating license stage of review, the peak calculated values of pressure and temperature for the drywell and wetwell should not exceed the respective design values. Also, the peak deck differential pressure for Mark II plants should not exceed the design value.

   For Mark III plants, the calculated results for drywell temperature, containment pressure, and differential pressure between the drywell and containment should be based on the General Electric Mark III analytical model (Ref. 30) that was used in the Grand Gulf analysis and evaluated by CSB. The use of this model at the construction permit stage is acceptable if, in the absence of complete large-scale Mark III test results, an appropriate margin (see below) between the calculated and design differential pressures is used. The Mark III analytical model will be verified by the large-scale Mark III test results prior to the operating license stage of review for a Mark III plant. If an analytical model other than the General Electric Mark III analytical model identified above is used, the model should be demonstrated to be physically appropriate and conservative to the extent that the General Electric model has been found acceptable. In addition, it will be necessary to demonstrate its performance with suitable test data in a manner similar to that described above.

   Additional analytical efforts are needed to further confirm the Mark III design. These matters were discussed in the ACRS letters issued on December 12, 1974, following its review of Perry and Allens Creek facilities. The areas of concern relate to vent clearing, vent interaction, pool swell, pool stratification, and dynamic and asymmetric loadings. These would also include an evaluation of oscillatory behavior.

   For Mark III plants at the construction permit stage, the containment design pressure should provide at least a 15% margin above the peak calculated containment pressure, and the drywell design differential pressure should provide at least a 30% margin above the peak calculated drywell differential pressure.

   For Mark III plants at the operating license stage, the peak calculated containment pressure and drywell differential pressure should be less than the design values. In general, it is expected that the peak calculated pressures will be about the same as at the construction permit stage. However, it is possible that the margins may be affected by revised or improved analytical models, test results, or minor changes in the as-built design of the plant.

2. Calculation of dynamic loads on suppression pool retaining structures and structures which may be located directly above the pool, as a result of pool motion during a loss-of-coolant accident or following actuation of one or more primary system pressure relief valves, should be based on appropriate analytical models and supported by applicable test data.

3. High energy lines passing through the containment should be provided with guard pipes or enclosed in other types of protective structures to assure that the suppression pool is not bypassed. If guard pipes are used, they should be designed in accordance with acceptance criteria established by the MEB as set forth in Standard Review Plan 3.6.2.

4. The allowable leakage areas for steam bypass of the suppression pool should be determined for a spectrum of postulated reactor coolant system pipe breaks. The maximum allowable bypass area of the plant should be based on conservative analyses which consider available energy removal mechanisms and the containment design pressure.

5. For Mark I and II containments, the maximum allowable leakage area for steam bypass of the suppression pool should be greater than the technical specification limit for leakage measured in periodic drywell-wetwell leakage tests. "Mark III containments should be designed to accommodate, for a spectrum of postulated reactor coolant system pipe breaks, without exceeding containment design pressure, a minimum bypass leakage area of the order of one square foot in terms of the parameter $A/\sqrt{k}$, where k is the resistance factor of the actual flow area, A." A leakage test of the drywell at about the design pressure should be performed prior to plant operation but as near to startup as feasible. The high pressure test will impose loads on the drywell which are a substantial fraction of the accident loads and will provide the necessary assurance that the drywell, as constructed, conforms to the design bases. Low pressure leakage tests of the drywell should be done periodically thereafter. The acceptance criterion for these tests should be that the measured leakage is less than the leakage corresponding to an equivalent 0.1 ft^2 leakage area (in terms of $A/\sqrt{k}$) at the test pressure. If the test conditions at a given pressure are representative of a substantial fraction of the loss-of-coolant accident loads, the acceptance criterion becomes 10% of the bypass capability at the given pressure. The determination of acceptable test conditions is made by the SEB. Testing methods and procedures should be described at the operating license stage of review.

6. For Mark III containments, justification should be provided for any reduction in the containment leak rate claimed for times less than 30 days after a postulated pipe break accident.

7. Provisions should be made in one of the following ways to protect the drywell and wetwell (or containment) of Mark I, II, and III plants, and the operating deck of Mark II plants, against loss of integrity from negative pressure transients or post-accident atmosphere cooldown:
   a. Structures should be designed to withstand the maximum calculated external pressure.

b. Vacuum relief devices should be provided in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, to assure that the design external pressures of the structures are not exceeded.

In either case, the design external pressures of the structures, including the design upward deck differential pressure for Mark II plants, should provide an adequate margin above the maximum calculated external pressures to account for uncertainties in the analyses.

8. Instrumentation capable of operating in the post-accident environment should be provided to monitor the drywell and wetwell (or containment) atmosphere pressure and temperature, and pool temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked throughout the course of an accident. Recording equipment capable of following the transient should be provided.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of BWR pressure-suppression containments. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis for aspects of functional design common to a class of BWR pressure-suppression type containments or by adopting the results of previous reviews of plants with essentially the same containment functional design.

1. The CSB reviews the analyses of the drywell and wetwell temperature and pressure response for Mark I and II containments. The CSB performs confirmatory analyses, when necessary, using the CONTEMPT-LT computer code. Input data for the code, including mass and energy release data, is generally taken from the safety analysis report; however, the CSB is currently working in conjunction with the CPB to develop a staff model to calculate mass and energy releases.

   The CSB normally analyzes only the design basis loss-of-coolant accident, which has been found from previous reviews to be the recirculation line break for Mark I and II plants. For Mark III plants, the steam line break has been determined to be the design basis loss-of-coolant accident. However, mass and energy releases from the recirculation line break will be evaluated using various flow correlations.

   The CSB evaluates analyses of both the short-term and long-term pressure and temperature responses of Mark III containment plants. For Mark III plants, the peak containment pressure following a loss-of-coolant accident is independent of the postulated pipe break size. The CSB reviews the containment response analysis presented in the safety analysis report to determine that the acceptance criteria in Section II have been satisfied. The CSB evaluates the conservatism of the assumptions, analytical methods, and long-term energy sources used in the analysis. The CSB also reviews the short-term drywell pressure response of Mark III containments. The CSB verifies from the safety analysis report that the mass and energy releases to the drywell during the period of interest (one second) are based on the acceptance criteria in Section II.

The CSB and its consultants have reviewed the General Electric Mark III analytical model and have determined that the code appears to calculate the drywell pressure response in an acceptable manner. The final acceptance of the code is predicated on its verification by the General Electric Mark III test program. Code modifications and improvements will be made as necessary as test data and other information becomes available.

The CSB verifies from the safety analysis report that the General Electric code has been utilized and that the input assumptions to the code are conservative. If analytical methods other than the General Electric model are used, the CSB, in conjunction with its consultants, will initiate a detailed review of the methods. In this case, the CSB reviews the proposed modeling, analytical methods and assumptions, correlation of results with applicable test data, and comparison with other similar analyses, to determine the acceptability of the proposed model.

The CSB reviews analyses of the drywell response to either a recirculation line rupture or a steam line rupture, as presented in the safety analysis report. The CSB determines from the results of these analyses that the "worst" break has been identified in establishing the drywell design differential pressure as well as the design pressure for subcompartments and equipment supports.

Modifications to the CONTEMP-LT computer code are being made which should provide the capability to perform confirmatory analyses of the Mark III drywell pressure response.

2. The CSB reviews analyses of the dynamic loads associated with suppression pool motion during loss-of-coolant accident or following actuation of one or more primary system pressure relief valves. The CSB evaluates the analytical methods, input assumptions, and results and determines the acceptability of the analyses by comparing the analytical results to applicable test data (e.g., from the General Electric large-scale Mark III tests).

3. For Mark III plants, the CSB verifies from the safety analysis report that high energy lines which pass through the containment outside the drywell are provided with guard pipes or enclosed in other types of protective structures. If guard pipes are used, the design must meet the acceptance criteria established in Standard Review Plan 3.6.2. For unguarded lines, the CSB reviews analyses of the consequences of postulated ruptures in these lines. The CSB bases its acceptance of the analyses on the conservatism of the methods and assumptions and on the margin provided to assure against exceeding the design pressure of the containment. If leakage detection and isolation equipment are provided, the CSB evaluates the effectiveness of the detection instrumentation and isolation devices to mitigate the consequences of a pipe rupture. The EICSB reviews the electrical design criteria for these systems.

4. The CSB evaluates analyses of bypass leakage capability. The CSB determines the adequacy of proposed bypass leakage tests and surveillance programs based on the results of previous reviews, operating experience at similar plants, and engineering judgment. At the operating license stage, CSB evaluates the proposed technical specifications pertaining to bypass leakage surveillance.

5. The CSB evaluates the conservatism of potential depressurization transients. If vacuum relief systems are provided, the CSB verifies from the safety analysis report that the design and operating characteristics of the system meet the requirements of Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. In evaluating surveillance and test programs for vacuum relief systems, the CSB uses the results of previous reviews and operating experience with similar systems to determine their adequacy. At the operating license stage, the CSB reviews the proposed technical specifications to assure that adequate surveillance and administrative control will be maintained over the vacuum relief devices.

6. The EICSB in Section 7.3 has review responsibility for the acceptability of and the qualification test program for the sensing and actuation instrumentation of the plant post-accident monitoring system.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are those listed in Standard Review Plan 6.2.1, together with the following:

1a. Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and attached Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.1.2 SUBCOMPARTMENT ANALYSIS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Mechanical Engineering Branch (MEB)
Core Performance Branch (CPB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The CSB reviews the information presented by the applicant in the safety analysis report concerning the determination of the design differential pressure values for containment subcompartments. A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within this volume. A short-term pressure pulse would exist inside a containment subcompartment following a pipe rupture within this volume. This pressure transient produces a pressure differential across the walls of the subcompartment which reaches a maximum value generally within the first second after blowdown begins. The magnitude of the peak value is a function of several parameters, which include blowdown mass and energy release rates, subcompartment volume, vent area, and vent flow behavior. A transient differential pressure response analysis should be provided for each subcompartment or group of subcompartments that meets the above definition.

The CSB review includes the manner in which the mass and energy release rate into the break compartment were determined, nodalization of subcompartments, subcompartment vent flow behavior, and subcompartment design pressure margins. This includes a coordinated review effort with the CPB. The CPB is responsible for the adequacy of the blowdown model.

The CSB review of the mass and energy release rates includes the basis for the selection of the pipe break size and location within each subcompartment containing a high energy line and the analytical procedure for predicting the short-term mass and energy release rates.

The CSB review of the subcompartment model includes the basis for the nodalization within each subcompartment, the initial thermodynamic conditions within each subcompartment, the nature of each vent flow path considered, and the extent of entrainment assumed in the vent flow mixture. The review may also include an analysis of the dynamic characteristics of components, such as doors, blowout panels, or sand plugs, that must open or be removed to provide a vent flow path, and the methods and results of components tests performed to demonstrate the validity of these analyses. The analytical procedure to determine the loss coefficients for each vent flow path and to predict the vent mass flow rates, including flow correlations used to compute sonic and subsonic flow conditions within a vent, is reviewed. The design pressure chosen for each subcompartment is also reviewed. On request from the APCSB, the CSB evaluates or performs pressure response analyses for subcompartments outside containment.

MEB is responsible for reviewing the acceptability of postulated break locations, and the design criteria and methods employed to limit pipe motion for postulated breaks within subcompartments (See Standard Review Plan 3.6.2).

MEB and SEB are responsible for reviewing the capability of components and their supports to withstand the loads associated with high energy pipe breaks postulated to occur within the subcompartments.

II. ACCEPTANCE CRITERIA

1. The subcompartment analysis should incorporate the following assumptions:

   a. Break locations and types should be chosen according to Regulatory Guide 1.46 for subcompartments inside containment and to Branch Technical Position MEB 3-1 (attached to Standard Review Plan 3.6.2) for subcompartments outside containment. An acceptable alternate procedure is to postulate a circumferential double-ended rupture of each high pressure system pipe in the subcompartment.

   b. Of several breaks postulated on the basis of a, above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

   c. The initial plant operating conditions, such as pressure, temperature, water inventory, and power level, should be selected to yield the maximum blowdown conditions. The selected operating conditions will be acceptable if it can be shown that a change of each parameter would result in a less severe blowdown profile.

2. The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-VI (Ref. 24), CRAFT (Ref. 23), CE FLASH-4 (Ref. 25), and RELAP-3 (Ref. 21), when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation. When RELAP-4 is accepted by the staff as an operational ECCS blowdown code, it will be acceptable for subcompartment analyses.

3. The initial atmospheric conditions within a subcompartment should be selected to maximize the resultant differential pressure. An acceptable model would be to assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from these, the selected values should be justified.

   Another model that is also acceptable, for a restricted class of subcompartments, involves simplifying the air model outlined above. For this model, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach should be limited to subcompartments that have choked flow within the vents. However, the adequacy of this simplified model for subcompartments having primarily subsonic flow through the vents has not been established.

4. Subcompartment nodalization schemes should be chosen such that there is no substantial pressure gradient within a node, i.e., the nodalization scheme should be verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes.

5. If vent flow paths are used which are not immediately available at the time of pipe rupture, the following criteria apply:

   a. The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.
   b. The validity of the analysis should be supported by experimental data or a testing program should be proposed at the construction permit stage that will support this analysis.
   c. The effects of missiles that may be generated during the transient should be considered in the safety analysis.

6. The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100% water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures.

7. At the construction permit stage, a factor of 1.4 should be applied to the peak differential pressure calculated in a manner found acceptable to the CSB for the subcompartment. The calculated pressure multiplied by 1.4 should be considered the design pressure. At the operating license stage, the peak calculated differential pressure should not exceed the design pressure. It is expected that the peak calculated differential pressure will not be substantially different from that of the construction permit stage. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.

III. REVIEW PROCEDURES

The procedures described below are followed for the subcompartment analysis review. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by adopting the results of previous reviews of plants with essentially the same subcompartment and high pressure piping design.

The CSB reviews the initial conditions selected for determining the mass and energy release rate to the subcompartments. These values are compared to the spectrum of allowable operating conditions for the plant. The CBS will ascertain the adequacy of the assumed conditions based on this review.

The CSB confirms with the MEB the validity of the applicant's analysis of subcompartments containing high energy lines and postulated pipe break locations, using elevation and plan drawings of the containment showing the routing of lines containing high energy fluids. The CSB determines that an appropriate reference case for subcompartment analysis has been identified. In the event a pipe break other than a double-ended pipe rupture is postulated by the applicant, the MEB will evaluate the applicant's justification for assuming a limited displacement pipe break.

The CSB may perform confirmatory analyses of the blowdown mass and energy profiles within a subcompartment. The analysis is done using the RELAP3 computer program (See Reference 21 for a description of this code). The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in this analysis. The use of RELAP3 will continue until the RELAP4 computer code has been approved by the staff as an acceptable blowdown code. At that time, the CSB will replace RELAP3 with RELAP4 for all subsequent analyses.

The CSB determines the adequacy of the information in the safety analysis report regarding subcompartment volumes, vent areas, and vent resistances. If a subcompartment must rely on doors, blowout panels, or equivalent devices to increase vent areas, the CSB reviews the analyses and testing programs that substantiate their use.

The CSB reviews the nodalization of each subcompartment to determine the adequacy of the calculational model. As necessary, CSB performs iterative nodalization studies for subcompartments to confirm that sufficient nodes have been included in the model.

The CSB compares the initial subcompartment air pressure, temperature, and humidity conditions to the criteria of II, above, to assure that conservative conditions were selected.

The CSB reviews the bases, correlations, and computer codes used to predict subsonic and sonic vent flow behavior and the capability of the code to model compressible and uncompressible flow. The bases should include comparisons of the correlations to both experimental data and recognized alternate correlations that have been accepted by the staff.

Using the nodalization of each subcompartment as specified in the safety analysis report, the CSB performs analyses using one of several available computer programs to determine the adequacy of the calculated peak differential pressure. The computer program used will depend upon the subcompartment under review as well as the flow regime. At the present time, the two programs used by the CSB are RELAP3 (Ref. 21) and CONTEMPT-LT (Refs. 7, 8, and 9). A multi-volume computer code is currently under development.

At the construction permit stage, the CSB will ascertain that the subcompartment design pressures include appropriate margins above the calculated values, as given in II, above.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are those listed in Standard Review Plan 6.2.1, together with the following:

1a. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

2a. Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," and attached Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment."

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SECTION 6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The CSB reviews analyses of the mass and energy released to the containment during loss-of-coolant accidents (LOCA) in conjunction with the review of the functional capability of the containment structure. While the CPB has the primary responsibility for the review of mass and energy release analyses, the CSB reviews this area as it relates to containment functional design. The review includes the following areas:

1. The sources of the energy assumed to be released to the containment.
2. The applicant's mass and energy release rate calculations for the initial blowdown phase of the accident.
3. For pressurized water reactor (PWR) plants, because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

II. ACCEPTANCE CRITERIA

The following acceptance criteria apply to the mass and energy release analysis for postulated loss-of-coolant accidents:

1. Sources of Energy

   The sources of stored and generated energy that should be considered in analyses of loss-of-coolant accidents include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water.

   Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. (Ref. 2). However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA.

The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

2. Calculations

In general, calculations of the mass and energy release rates for a loss-of-coolant accident should be done in a manner that conservatively establishes the containment internal design pressure; i.e., maximizes the post-accident containment pressure. The criteria given below for each phase of the accident indicate the conservatism that should exist. These calculations should be done for a spectrum of possible pipe breaks to assure that the worst case has been identified. This spectrum should include hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks with cross-sectional areas up to and including a double-ended pipe break, and longitudinal splits in the largest pipes with break areas equal to the cross-sectional area of the pipe.

a. Initial Blowdown Phase

The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions existing at 102% of full power (Ref. 2).

Mass release rates should be calculated using the Moody model (Ref. 22), or a model that is demonstrated to be equally conservative.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWR's should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

For Mark I BWR plants, the mass and energy releases from the rupture of a recirculation line should be based on the General Electric blowdown model (Ref. 30). For Mark II and III plants, mass and energy releases from the rupture of a recirculation line or main steam line should be based on a blowdown model which accounts for the short-term (less than one second) pressure response time of the drywell.

The following computer codes and models are acceptable for calculating mass and energy release rates during the blowdown phase, if they incorporate the foregoing criteria: CRAFT (Ref. 23), SATAN VI (Ref. 24), CE FLASH-4 (Ref. 25), and the GE blowdown model (Ref. 30). Other codes will be acceptable for containment analyses if they are approved by CPB and are determined to be conservative for containment analyses.

b. PWR Core Reflood Phase (Cold Leg Breaks Only)

Following initial blowdown of the reactor coolant system, the water remaining in the reactor vessel should be assumed to be saturated and at the level of the bottom of the active core.

Calculations of the core flooding rate should be based on the emergency core cooling system operating condition that maximizes the containment pressure either during the core reflood phase or the post-reflood phase.

Calculations of liquid entrainment; i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR FLECHT experiments (Ref. 26). Liquid entrainment should be assumed to continue until the water level in the core is 2 feet from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 18-inch core level, a linearly increasing CRF to 0.80 at the 24-inch level, and a constant CRF of 0.80 until the water level is 2 feet from the top of the core. Above this level, a CRF of 0.05 may be used.

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increased core inlet water temperature caused by steam quenching.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

c. PWR Post-Reflood Phase (Cold Leg Breaks Only)

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

d. PWR Decay Heat Phase (Cold Leg Breaks Only)

The dissipation of core decay heat should be considered during this phase of the accident.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis for loss-of-coolant accidents. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CPB and the CSB compare the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in Section II, above. The CPB determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a loss-of-coolant accident. The CSB also compares energy inventories at various times during a loss-of-coolant accident to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale. The acceptance of the methods and determination of the degree of conservatism is the responsibility of the CPB. In general, such efforts are accomplished through co-operation between branches.

Mass and energy release data for computer codes that have not been previously reviewed and accepted can be compared to the results of computer codes which have been found acceptable, such as the SATA-VI, CRAFT, and CE FLASH-4 computer codes, and the GE blowdown model. FLOOD 1 or FLOOD 2 computer codes (Refs. 17 and 16) are used, as appropriate, to analyze the mass and energy releases for the PWR core reflood phase. The acceptability of assumptions made in the analyses regarding steam quenching may be determined by comparing the results of the analyses with applicable experimental data.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.4 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The CSB reviews analyses of the mass and energy released to the containment during a steam or feedwater line break accident in conjunction with its review of the functional capability of the containment structure. The CSB review includes the following areas:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations.

II. ACCEPTANCE CRITERIA

In addition to the provisions of General Design Criterion 50, the following acceptance criteria apply to the mass and energy release analysis for postulated PWR secondary system pipe ruptures:

1. Sources of Energy

The sources of energy that should be considered in analyses of steam and feedwater line break accidents include: the stored energy in the affected steam generator metal, including the vessel tubing, feedwater line, and steam line; the stored energy in the water contained within the affected steam generator; the stored energy in the feedwater transferred to the affected steam generator prior to the closure of the isolation valves in the feedwater line; the stored energy in the steam from the unaffected steam generator(s) prior to the closure of the isolation valves in the steam generator crossover lines; and the energy transferred from the primary coolant to the water in the affected steam generator during blowdown.

The steam line break accident should be analyzed for various plant conditions from hot standby to 102% of full power.

2. Mass and Energy Release Rate Calculations

In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be done in a manner that is conservative from a containment response standpoint; i.e., that maximizes the post-accident containment pressure. The following criteria indicate the degree of conservatism that is desired.

Mass release rates should be calculated using the Moody model (Ref. 22), or a model that is demonstrated to be equally conservative.

Calculations of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer.

Calculations of mass release should consider the water in the affected steam generator and feedwater line, the feedwater transferred to the affected steam generator prior to the closure of the isolation valves in the feedwater lines, the steam in the affected steam generator, and the steam coming from the unaffected steam generator(s) as the secondary system is being depressurized prior to the closure of the isolation valves in the steam generator crossover lines.

If liquid entrainment is calculated for steam line breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steam line breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur, to allow selection of the maximum release case. If no liquid entrainment is calculated, a double-ended rupture of the steam line should be assumed.

The single active failure in the steam generator feedwater line isolation provisions or feedwater pumps that optimizes the mass and energy release to the containment, such that the containment peak pressure is maximized, should be assumed to occur in steam and feedwater line break analyses.

III. REVIEW PROCEDURES

The procedures described below are followed for the review of the mass and energy release analysis of secondary coolant system pipe breaks. The reviewer selects and emphasizes material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CSB reviews the secondary coolant system pipe break analysis assumptions to determine whether the "worst" pipe break accident case has been identified by the applicant, and whether the analysis was done in a conservative manner from the standpoint of containment pressure. The CPB reviews the acceptability of the analytical models.

This review may involve coordination between members of the CPB and the CSB on the proposed methods and models used for blowdown analyses. The acceptability of the approach used by the applicant is evaluated based on the acceptance criteria in Section II. The CSB also reviews analyses of postulated single failures of active components in the secondary system, such as steam and feedwater line isolation valves and feedwater pumps, to determine whether the most severe single failure has been selected which allows mass and energy from the unaffected steam generators and the feedwater system to be transferred to the steam generator blowing down.

If liquid entrainment is calculated in the applicant's steam line break model, the CSB and CPB will determine the validity of the experimental data provided to support the entrainment calculation. CSB and CPB will also ascertain that the effect of steam separators located upstream from the postulated steam line break have been taken into account in the analysis. In addition, the CSB reviews the results of a spectrum of steam line breaks, beginning with the double-ended break and decreasing in area until no entrainment occurs, to be sure that the worst steam line break size has been identified.

The CSB performs confirmatory analyses of the containment pressure response to steam and feedwater line breaks inside the containment using the CONTEMPT-LT computer code.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. REFERENCES

The references for this plan are listed in Standard Review Plan 6.2.1.

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SECTION 6.2.1.5 MINIMUM CONTAINMENT PRESSURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM PERFORMANCE CAPABILITY STUDIES

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

Following a loss-of-coolant accident in a pressurized water reactor (PWR) plant, the emergency core cooling system (ECCS) will supply water to the reactor vessel to reflood, and thereby cool, the reactor core. The core flooding rate is governed by the capability of the ECCS water to displace the steam generated in the reactor vessel during the core reflooding period. For PWR plants, there is a direct dependence of core flooding rate on containment pressure; i.e., the core flooding rate will increase with increasing containment pressure. Therefore, as part of the overall evaluation of ECCS performance, the CSB reviews analyses of the minimum containment pressure that could exist during the period of time until the core is reflooded following a loss-of-coolant accident to confirm the validity of the containment pressure used in ECCS performance capability studies. The CSB reviews the assumptions made regarding the operation of engineered safety feature heat removal systems; the effectiveness of structural heat sinks within the containment to remove energy from the containment atmosphere, and other heat removal processes, such as steam in the containment mixing with ECCS water spilling from the break in the reactor coolant system; and in the case of ice condenser containments, mixing with water from melted ice that drains into the lower containment volume. The review is done for all PWR containment types; i.e., dry, subatmospheric, and ice condenser containments.

The CPB is responsible for determining the acceptability of the mass and energy release data used in the minimum containment pressure analysis (See Standard Review Plan 6.3). This information is derived from the applicant's evaluation of ECCS performance capability in accordance with Appendix K to 10 CFR Part 50.

It should be noted that the minimum containment pressure analysis done in connection with ECCS performance evaluation differs from the containment functional performance analysis, in that the conservatisms and margins are taken in opposite directions in the two cases. Thus, the minimum containment pressure analysis required by the regulations for ECCS performance evaluation is not conservative with regard to peak containment pressure in the event of a loss-of-coolant accident and cannot be used to determine the containment design basis.

II. ACCEPTANCE CRITERIA

Paragraph I.D.2 of Appendix K to 10 CFR Part 50 requires that the containment pressure used to evaluate the performance capability of a PWR emergency core cooling system not exceed a pressure calculated conservatively for that purpose.

The guidelines given below indicate the conservatism that analyses of the containment response to loss-of-coolant accidents should have for determining the minimum containment pressure for ECCS performance capability studies:

1. Calculations of the mass and energy released during postulated loss-of-coolant accidents should be based on the requirements of Appendix K to 10 CFR Part 50 (Ref. 2).

2. Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," delineates the calculational approach that should be followed to assure a conservative prediction of the minimum containment pressure.

III. REVIEW PROCEDURES

The review procedures described below are followed for the review of the minimum containment pressure analysis. The reviewer selects and emphases material from these procedures as may be appropriate for a particular case. Portions of the review may be carried out on a generic basis or by applying the results of previous reviews of similar plants.

The CSB reviews the analyses in the safety analysis report of the minimum containment pressure following a loss-of-coolant accident. The CSB, in conjunction with the CPB, confirms the validity of the applicant's mass and energy release data. The CSB evaluates the conservativeness of the assumptions used by the applicant regarding the operation of containment heat removal systems and the effectiveness of structural heat sinks, by comparing the applicant's calculational approach to the method outlined in Branch Technical Position CSB 6-1. In certain cases, the CSB may perform confirmatory containment pressure response analyses using the CONTEMPT-LT computer code. In these cases, the containment pressure calculated by the CSB is compared to that used in the applicant's evaluation of the performance capability of the emergency core cooling system, to ensure that an appropriately conservative value has been used.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this section are presented in Standard Review Plan 6.2.1.

V. The references for this plan are listed in Standard Review Plan 6.2.1.

BRANCH TECHNICAL POSITION CSB 6-1

MINIMUM CONTAINMENT PRESSURE MODEL
FOR PWR ECCS PERFORMANCE EVALUATION

A. BACKGROUND

Paragraph I.D.2 of Appendix K to 10 CFR Part 50 (Ref. 1) requires that the containment pressure used to evaluate the performance capability of a pressurized water reactor (PWR) emergency core cooling system (ECCS) not exceed a pressure calculated conservatively for that purpose. It further requires that the calculation include the effects of operation of all installed pressure-reducing systems and processes. Therefore, the following branch technical position has been developed to provide guidance in the performance of minimum containment pressure analysis. The approach described below applies only to the ECCS-related containment pressure evaluation and not to the containment functional capability evaluation for postulated design basis accidents.

B. BRANCH TECHNICAL POSITION

1. Input Information for Model

a. Initial Containment Internal Conditions

The minimum containment gas temperature, minimum containment pressure, and maximum humidity that may be encountered under limiting normal operating conditions should be used.

b. Initial Outside Containment Ambient Conditions

A reasonably low ambient temperature external to the containment should be used.

c. Containment Volume

The maximum net free containment volume should be used. This maximum free volume should be determined from the gross containment volume minus the volumes of internal structures such as walls and floors, structural steel, major equipment, and piping. The individual volume calculations should reflect the uncertainty in the component volumes.

2. Active Heat Sinks

a. Spray and Fan Cooling Systems

The operation of all engineered safety feature containment heat removal systems operating at maximum heat removal capacity; i.e., with all containment spray trains operating at maximum flow conditions and all emergency fan cooler units operating, should be assumed. In addition, the minimum temperature of the stored water for the spray cooling system and the cooling water supplied to the fan coolers, based on technical specification limits, should be assumed.

Deviations from the foregoing will be accepted if it can be shown that the worst conditions regarding a single active failure, stored water temperature, and cooling water temperature have been selected from the standpoint of the overall ECCS model.

b. Containment Steam Mixing With Spilled ECCS Water

The spillage of subcooled ECCS water into the containment provides an additional heat sink as the subcooled ECCS water mixes with the steam in the containment. The effect of the steam-water mixing should be considered in the containment pressure calculations.

c. Containment Steam Mixing With Water from Ice Melt

The water resulting from ice melting in an ice condenser containment provides an additional heat sink as the subcooled water mixes with the steam while draining from the ice condenser into the lower containment volume. The effect of the steam-water mixing should be considered in the containment pressure calculations.

3. Passive Heat Sinks

a. Identification

The passive heat sinks that should be included in the containment evaluation model should be established by identifying those structures and components within the containment that could influence the pressure response. The kinds of structures and components that should be included are listed in Table 1.

Data on passive heat sinks have been compiled from previous reviews and have been used as a basis for the simplified model outlined below. This model is acceptable for minimum containment pressure analyses for construction permit applications, and until such time (i.e., at the operating license review) that a complete identification of available heat sinks can be made. This simplified approach has also been followed for operating plants by licensees complying with Section 50.46 (a)(2) of 10 CFR Part 50. For such cases, and for construction permit reviews, where a detailed listing of heat sinks within the containment often cannot be provided, the following procedure may be used to model the passive heat sinks within the containment:

(1) Use the surface area and thickness of the primary containment steel shell or steel liner and associated anchors and concrete, as appropriate.

(2) Estimate the exposed surface area of other steel heat sinks in accordance with Figure 1 and assume an average thickness of 3/8 inch.

(3) Model the internal concrete structures as a slab with a thickness of 1 foot and exposed surface of 160,000 ft^2.

The heat sink thermophysical properties that would be acceptable are shown in Table 2.

At the operating license stage, applicants should provide a detailed list of passive heat sinks, with appropriate dimensions and properties.

b. Heat Transfer Coefficients

The following conservative condensing heat transfer coefficients for heat transfer to the exposed passive heat sinks during the blowdown and post-blowdown phases of the loss-of-coolant accident should be used (See Figure 2):

(1) During the blowdown phase, assume a linear increase in the condensing heat transfer coefficient from $h_{initial}$=8 Btu/hr-ft^2-°F, at t = 0, to a peak value four times greater than the maximum calculated condensing heat transfer coefficient at the end of blowdown, using the Tagami correlation (Ref. 2),

$$h_{max} = 72.5 \left[\frac{Q}{Vt_p}\right]^{0.62}$$

where h_{max} = maximum heat transfer coefficient, Btu/hr-ft^2-°F

Q = primary coolant energy, Btu

V = net free containment volume, ft^3

t_p = time interval to end of blowdown, sec.

(2) During the long-term post-blowdown phase of the accident, characterized by low turbulence in the containment atmosphere, assume condensing heat transfer coefficients 1.2 times greater than those predicted by the Uchida data (Ref. 3) and given in Table 3.

(3) During the transition phase of the accident, between the end of blowdown and the long-term post-blowdown phase, a reasonably conservative exponential transition in the condensing heat transfer coefficient should be assumed (see Figure 2).

The calculated condensing heat transfer coefficients based on the above method should be applied to all exposed passive heat sinks, both metal and concrete, and for both painted and unpainted surfaces.

Heat transfer between adjoining materials in passive heat sinks should be based on the assumption of no resistance to heat flow at the material interfaces. An example of this is the containment liner to concrete interface.

C. REFERENCES

1. 10 CFR §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

2. T. Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966 (unpublished work).

3. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

TABLE 1

IDENTIFICATION OF CONTAINMENT HEAT SINKS

1. Containment Building (e.g., liner plate and external concrete walls, floor, and sump, and liner anchors).

2. Containment Internal Structures (e.g., internal separation walls and floors, refueling pool and fuel transfer pit walls, and shielding walls).

3. Supports (e.g., reactor vessel, steam generator, pumps, tanks, major components, pipe supports, and storage racks).

4. Uninsulated Systems and Components (e.g., cold water systems, heating, ventilation, and air conditioning systems, pumps, motors, fan coolers, recombiners, and tanks).

5. Miscellaneous Equipment (e.g., ladders, gratings, electrical cable trays, and cranes).

TABLE 2

HEAT SINK THERMOPHYSICAL PROPERTIES

Material	Density lb/ft^3	Specific Heat Btu/lb-°F	Thermal Conductivity Btu/hr-ft-°F
Concrete	145	0.156	0.92
Steel	490	0.12	27.0

TABLE 3

UCHIDA HEAT TRANSFER COEFFICIENTS

Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)	Mass Ratio (lb air/lb steam)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)
50	2	3	29
20	8	2.3	37
18	9	1.8	46
14	10	1.3	63
10	14	0.8	98
7	17	0.5	140
5	21	0.1	280
4	24		

Figure 1
Area of Steel Heat Sinks Inside Containment

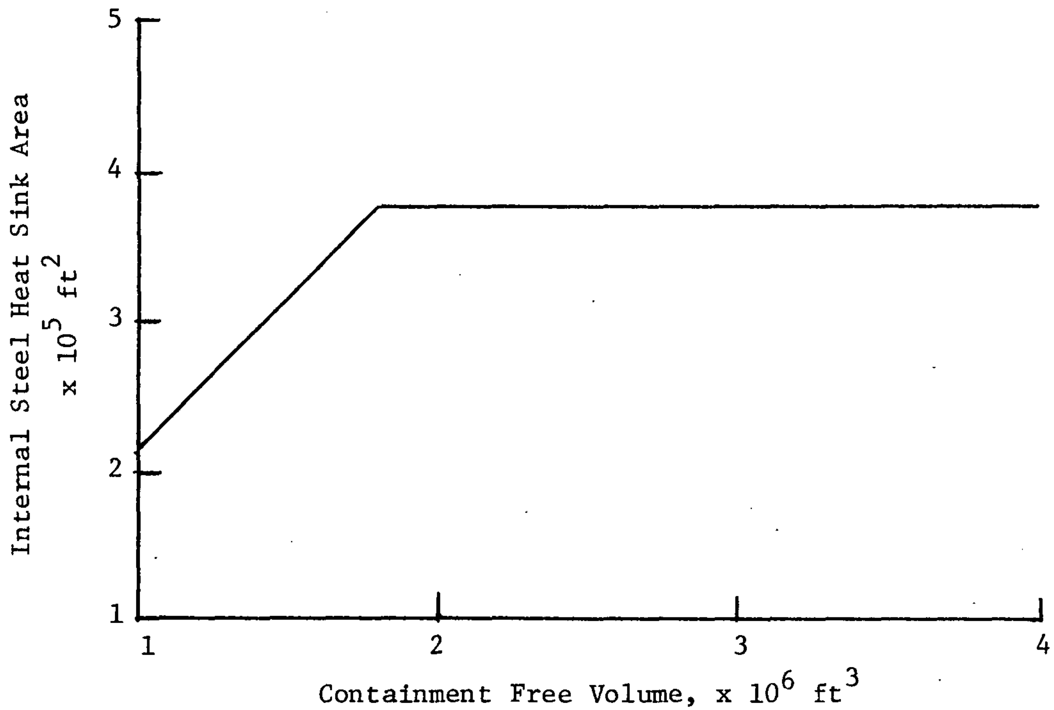
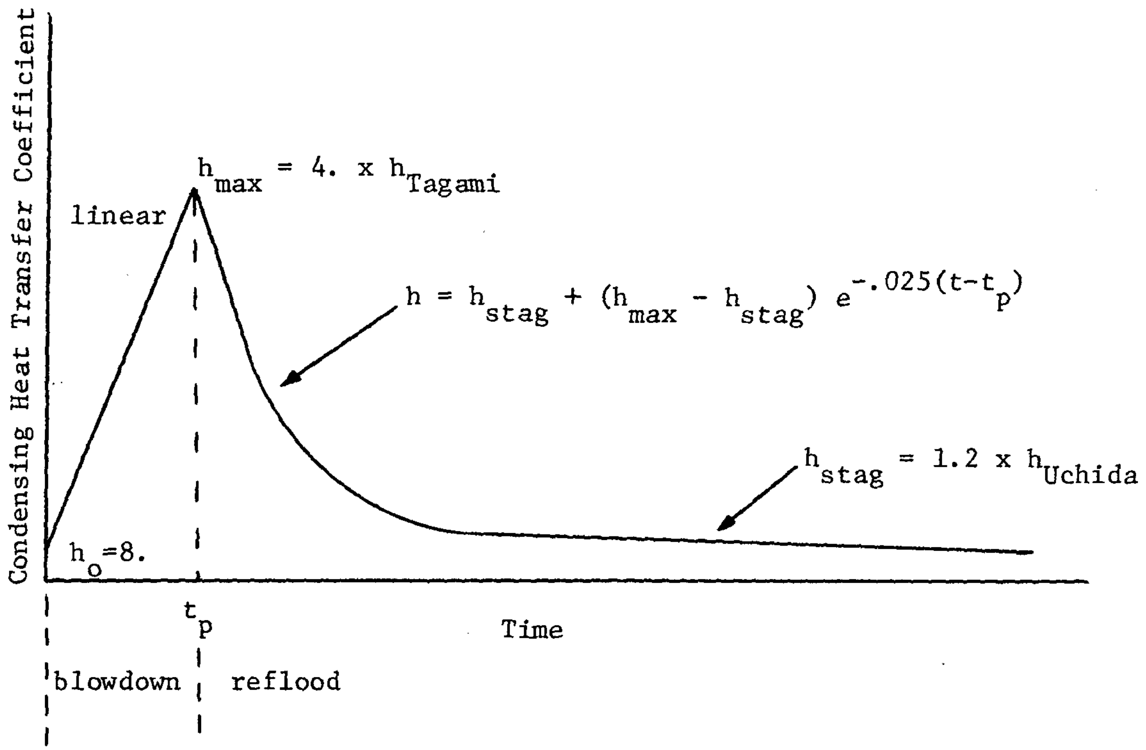


Figure 2

Condensing Heat Transfer Coefficients for Static Heat Sinks



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SECTION 6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The CSB reviews the information in the applicant's safety analysis report (SAR) concerning containment heat removal under post-accident conditions. The information needed for this review is described in Reference 13. The types of systems provided to remove heat from the containment include fan cooler systems, spray systems, and residual heat removal systems. These systems remove heat from the containment atmosphere and the containment sump water, or the water in the containment wetwell. The CSB review includes the following analyses and aspects of containment heat removal system designs:

1. Analyses of the consequences of single component malfunctions in each system.

2. Analyses of the available net positive suction head (NPSH) to the recirculation heat removal pumps.

3. Analyses of the heat removal capability of the spray water system.

4. Analyses of the heat removal capability of fan cooler heat exchangers.

5. The potential for surface fouling of fan cooler, recirculation, and residual heat removal heat exchangers, and the effect on heat exchanger performance.

6. The quality group classification of each system.

7. The seismic design classification of each system.

8. The design provisions and proposed program for periodic inservice inspection and operability testing of each system or component.

9. The proposed technical specifications for each system.

10. The instrumentation provided to monitor system or component performance.

11. The design of sumps for emergency core cooling and containment spray systems.

12. The effects of debris including insulation on recirculating fluid systems.

The APCSB has the review responsibility for the secondary cooling systems which provide for heat removal from the containment systems to the ultimate heat sink. The APCSB is responsible for determining that the systems supplying cooling water to the heat exchangers in the containment heat removal systems meet the design requirements for engineered safety features.

The EICSB has review responsibility for the sensing and actuation instrumentation for the containment heat removal systems (Standard Review Plan 7.3) and for the qualification test programs for their instrumentation.

The AAB reviews fission product control features of containment spray systems (Standard Review Plan 6.5.2).

II. ACCEPTANCE CRITERIA

General Design Criteria 38, 39, 40, and 50 of 10 CFR Part 50, Appendix A, establish requirements for the design, periodic inspection and operability testing, and functional capability of the containment heat removal systems (Refs. 1, 2, 3, and 4). The items listed below amplify these general requirements and form the basis for the staff's detailed review of containment heat removal systems.

1. The containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature; i.e., the systems should be designed to accommodate a single active failure. The results of failure modes and effects analyses of each system should assure that the system is capable of withstanding a single failure without loss of function. (See Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," for the definition of Single Failure.)

2. The recirculation spray system is required to circulate water in the containment in the long term (after about one hour) following a loss-of-coolant accident, and should be designed to accomplish this without pump cavitation occurring. Therefore, the net positive suction head available to the recirculation pumps should be greater than the required NPSH. A supporting analysis should be presented in sufficient detail to permit the staff to determine the adequacy of the analysis and should show that the available NPSH is greater than the required NPSH. The analysis will be acceptable if it is done in accordance with the guidelines of Regulatory Guide 1.1, i.e., is based on maximum expected temperatures of the pumped fluid and generally with atmospheric pressure in the containment.

   The recirculation spray system for a subatmospheric containment is designed to start about five minutes after a loss-of-coolant accident; i.e., during the injection phase of spray system operation. For subatmospheric containments, the guidelines of Regulatory Guide 1.1 with regard to containment pressure will apply after the injection phase has terminated, which occurs about one hour after the accident.

3. Analyses of the heat removal capability of the spray system should be based on the following considerations:

   a. The locations of the spray headers relative to the internal structures.

   b. The arrangement of the spray nozzles on the spray headers and the expected spray pattern.

   c. The type of spray nozzles used and the nozzle atomizing capability, i.e., the spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle.

   d. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.

   The spray systems should be designed to assure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize the overlapping of the sprays.

4. The design heat removal capability (i.e., heat removal rate vs. containment temperature) of fan coolers should be established on the basis of qualification tests on production units or acceptable analyses that take into account the expected post-accident environmental conditions and variations in major operating parameters such as the containment atmosphere steam-air ratio, condensation on finned surfaces, and cooling water temperature and flow rate. The equipment housing and ducting associated with the fan cooler system should be analyzed to determine that the design is adequate to withstand the effects of containment pressure following a loss-of-coolant accident (See Standard Review Plan 6.2.5). Fan cooler system designs that contain components which do not have a post-accident safety function, should be designed such that a failure of non-safety related equipment will not prevent the fan cooler system from accomplishing its safety function.

5. The potential for surface fouling of the secondary sides of fan cooler, recirculation, and residual heat removal heat exchangers by the cooling water over the life of the plant and the effect of surface fouling on the heat removal capacity of the heat exchangers should be analyzed and the results discussed in the SAR. The analysis will be acceptable if it is shown that provisions such as closed cooling water systems are provided to prevent surface fouling or surface fouling has been accounted for in establishing the heat removal capability of the heat exchangers.

6. The containment heat removal systems should be designed, fabricated, erected, and tested to Group B quality standards, as recommended by Regulatory Guide 1.26.

7. The containment heat removal systems should be designated Category I (seismic), as recommended by Regulatory Guide 1.29.

8. Provisions should be made in the design of containment heat removal systems for periodic inspection and operability testing of the systems and system components such

as pumps, valves, duct pressure-relieving devices, and spray nozzles. The inspection and test program will be acceptable if it is judged by the CSB to be consistent with that proposed for other engineered safety features.

9. Instrumentation should be provided to monitor containment heat removal system and system component performance under normal and accident conditions. The instrumentation should be capable of determining whether a system is performing its intended function, or a system train or component is malfunctioning and should be isolated. The instrumentation should be redundant and where practical, diverse, and should have readout and alarm capability in the control room.

10. Provisions should be made to allow drainage of spray and emergency core cooling water to the sumps (recirculation piping suction points). The design of protective screen assemblies around recirculation piping suction points will be acceptable if it is capable of preventing debris from entering the recirculation piping which could impair the performance of system pumps, valves, heat exchangers, or spray nozzles. Regulatory Guide 1.82 (Ref. 8) provides guidance on the design of sumps for emergency core cooling and containment spray systems.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the review of containment heat removal systems. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of heat removal systems common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same systems.

CSB assures that the design and functional capability of the containment heat removal system conform to the requirements of General Design Criteria 38, 39, 40, and 50.

CSB determines the acceptability of a containment heat removal system design by reviewing failure modes and effects analyses of the system to be sure that all potential single failures have been identified and no single failure could incapacitate the entire system; comparing the quality standards applied to the system to Regulatory Guide 1.26; comparing the seismic design classification of the system to Regulatory Guide 1.29; reviewing qualification tests performed on system components such as fan coolers; reviewing the system design provisions for periodic inservice inspection and operability testing to ensure that the system and components are accessible for inspection and all active components can be tested; and reviewing the capability to monitor system performance and control active components from the control room so that the operator can exercise control over system functions or isolate a malfunctioning system component.

For plants at the operating license stage of review, the CSB reviews the proposed technical specifications for containment heat removal systems to assure that limiting conditions for operation and surveillance requirements satisfy the intent of General Design Criteria 39 and 40.

CSB reviews analyses of the net positive suction head available to the recirculation pumps since recirculation system operability is contingent upon adequate NPSH being available to preclude pump cavitation. Calculations of the available NPSH are based on transient values of the containment pressure, the vapor pressure of the pumped fluid, the suction head, and the friction head. Containment pressure and vapor pressure head are addressed in Regulatory Guide 1.1, which recommends that the NPSH analyses be based on maximum sump water temperature and, in general, atmospheric containment pressure. CSB reviews the analyses in accordance with the guidelines of Regulatory Guide 1.1. The analyses should provide justification that the assumed accident conditions lead to a conservative prediction of the sump water temperature by discussing the effects of assuming various combinations of operating modes of emergency core cooling equipment and containment heat removal equipment. The conservatism in determining the water level in the containment and the friction losses in the recirculation system suction piping should be justified. For example, the uncertainty in determining the free volume in the lower part of the containment that may be occupied by water, and the quantity of water that may be trapped by the reactor cavity and the refueling canal should be factored into the calculation of the suction head.

The recommendation in Regulatory Guide 1.1 that the calculation of available NPSH be based on the assumption of atmospheric pressure in the containment does not apply directly to subatmospheric containments. The recirculation system in a subatmospheric containment is designed to become operational within five minutes following a loss-of-coolant accident. CSB permits the short-term elevated containment pressure to be used in the calculation of NPSH. After the containment has been depressurized, the subatmospheric pressure that existed in the containment prior to the accident should be used in the calculation.

If in the judgment of the CSB, the NPSH analyses were not done in a sufficiently conservative manner, confirmatory analyses are performed using the CONTEMPT-LT computer code. See References 10, 11, and 12 for a description of this code.

The CSB also reviews the evaluation of the volume of the containment covered by the sprays and the extent of overlapping of the sprays with respect to heat removal capabilities. A judgment will be made regarding the acceptability of the spray coverage and extent of overlapping; the volume of the containment covered by the sprays should be maximized and the extent of overlapping kept to a minimum. Elevation and plan drawings of the containment showing the spray patterns are used to determine coverage and overlapping.

In general, the design requirements for the spray systems with respect to spray drop size spectrum and mean drop size, spray drop residence time in the containment atmosphere, containment coverage by the sprays, and extent of overlapping of the sprays are more stringent when the acceptability of the system is being considered from an iodine removal capability standpoint rather than from a heat removal capability standpoint. Consequently, when the iodine removal capability of the system is satisfied, the heat removal capability will be found acceptable. The Accident Analysis Branch is responsible for determining the acceptability of the iodine removal effectiveness of the sprays (See Standard Review Plan 6.5.2). Since all plants do not use the containment sprays as a fission product removal system, the CSB reviews the system for cases where the system is used only as a heat removal system.

CSB reviews analyses of the heat removal capability of the spray system. This capability is a function of the degree of thermal equilibrium attained by the spray water and the volume of the containment covered by the spray water. The spray drop size and residence time in the containment atmosphere determine the degree of thermal equilibrium attained by the spray water. The CSB confirms the validity of the degree of thermal equilibrium attained using the following information: an elevation drawing of the containment showing the locations of the spray headers relative to the internal structures, including fall heights, and the results of the spray nozzle test program to determine the spectrum of drop sizes and mean drop size emitted from the nozzles as a function of pressure drop across the nozzles.

Reference 9 contains information regarding the heating of spray drops in air-steam atmospheres which can be used to determine the validity of the degree of thermal equilibrium of the spray water used in the analyses.

CSB reviews the adequacy of provisions made to prevent overpressurization of fan cooler ducting following a loss-of-coolant accident (Standard Review Plan 6.2.5). CSB reviews the heat removal capability of the fan coolers. The test programs and calculation models used to determine the performance capability of fan coolers are reviewed for acceptability. If the secondary side of a fan cooler heat exchanger is not a closed system, the CSB reviews the potential for surface fouling. The CSB determines whether or not surface fouling impairs the heat removal capability of a fan cooler.

CSB reviews the system provided to allow drainage of containment spray water and emergency core cooling water to the recirculation suction points (sumps). CSB reviews the design of the protective screen assemblies around the suction points. CSB reviews potential sources of debris including the types of insulation used inside the containment. CSB reviews plan and elevation drawings of the protective screen assemblies, showing the relative positions and orientations of the trash bars or grating and the stages of screening, to determine that the potential for debris clogging the screening is minimized. CSB also reviews the drawings to determine that suction points do not share the same screened enclosure. The effectiveness of the protective screen assembly will be determined by comparing the smallest mesh size of screening provided to the clogging potential of pumps, heat exchangers, valves, and spray nozzles. The methods of attachment of the trash bars or grating and the screening to the protective screen assembly structure should be discussed in the SAR and shown on drawings. A discussion of the adequacy of the surface area of screening with respect to assuring a low velocity of approach of the water to minimize the potential for debris in the water being sucked against the screening should be presented. Regulatory Guide 1.82 (Ref. 8) presents guidelines for the acceptability of the design of containment sumps.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"6.2.2 Containment Heat Removal Systems

The containment heat removal systems include (identify the systems).

"The scope of review of the containment heat removal systems for the __________ plant has included system drawings and descriptive information. The review has included the applicant's proposed design bases for the containment heat removal systems, and the analyses of the functional capability of the systems.

"The basis for the staff's acceptance has been the conformance of system designs and design bases to the Commission's Regulations as set forth in the general design criteria, and to applicable regulatory guides, staff technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of the containment heat removal systems should be discussed.)

"The staff concludes that the design of the containment heat removal systems conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 38, "Containment Heat Removal."

2. 10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

4. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

5. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

7. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

8. Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

9. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part VI, The Heating of Spray Drops In Air-Steam Atmospheres," ORNL-TM-2412, Oak Ridge National Laboratory, January 1970.

10. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

11. C. F. Carmichael and S. A. Marks, "CONTEMPT-PS, A Digital Computer Code For Predicting The Pressure-Temperature History Within A Pressure-Suppression Containment Vessel In Response To A Loss-of-Coolant Accident," IDO-17252, Phillips Petroleum Company, April 1969.

12. L. C. Richardson, L. J. Finnegan, R. J. Wagner, and J. M. Waage, "CONTEMPT, A Computer Program For Predicting The Containment Pressure-Temperature Response To A Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

13. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

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SECTION 6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

The CSB reviews the information in the applicant's safety analysis report (SAR) concerning the functional capability of the secondary containment system. The secondary containment system includes the outer containment structure of dual containment plants and the associated systems provided to mitigate the radiological consequences of postulated accidents. The secondary containment structure and supporting systems are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. Other plant areas contiguous to the secondary containment may also be served by these or similar systems.

The CSB review of the functional capability of the secondary containment system of dual containment designs includes the following points:

1. Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident within the primary containment.

2. Analyses of the effect of openings in the secondary containment on the capability of the depressurization and filtration system to accomplish its design objective of establishing a negative pressure in a prescribed time.

3. Analyses of the pressure and temperature response of the secondary containment to a high energy line rupture within the secondary containment.

4. The functional design criteria applied to guard pipes surrounding high energy lines within the secondary containment.

5. Analyses of any primary containment leakage paths that bypass the secondary containment.

6. The design provisions for periodic leakage testing of secondary containment bypass leakage paths.

7. The proposed technical specifications pertaining to the functional capability of the secondary containment system and the leakage testing of bypass leakage paths.

The AAB reviews the design requirements and the periodic inspection and operability test program for the depressurization and filtration systems, from the standpoint of assuring that the systems and system components are functionally capable of depressurizing the secondary containment. The fission product removal capability of the secondary containment supporting systems is reviewed by the AAB under Standard Review Plan 6.5.3.

II. ACCEPTANCE CRITERIA

1. Analyses of the pressure and temperature response of the secondary containment to a loss-of-coolant accident occurring in the primary containment should be based on the following guidelines:

   a. Both radiative and convective heat transfer from the primary containment structure to the secondary containment atmosphere should be considered.

   b. Adiabatic conditions should be assumed for the secondary containment structure, i.e., no heat transfer from the secondary containment structure to the environs should be assumed.

   c. The compressive effect of primary containment expansion on the secondary containment atmosphere should be considered.

   d. Secondary containment inleakage should be considered.

   e. No credit should be taken for secondary containment outleakage.

   f. Any delay in actuating the secondary containment depressurization and filtration system should be considered.

2. High energy lines passing through the secondary containment should be provided with guard pipes. Design criteria for guard pipes are given in Standard Review Plan 3.6.2. If guard pipes are not provided, analyses should be provided which demonstrate that the secondary containment structure is capable of withstanding the effects of a high energy pipe rupture occurring inside the secondary containment without loss of integrity.

3. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. Branch Technical Position (BTP) CSB 6-3 (Ref. 7) provides guidance for identifying the leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly

bypass the secondary containment and other contiguous areas served by ventilation and filtration systems should be described.

4. The negative pressure to be maintained in the secondary containment and other contiguous plant areas should be low enough to preclude exfiltration under wind loading conditions characteristic of the plant site. If the leakage rate is in excess of 100% of the volume per day, a special exfiltration analysis should be performed.

5. The containment depressurization and filtration systems should be capable of maintaining a uniform negative pressure throughout the secondary containment, as well as other areas served by the systems.

6. Provisions should be made in the design of the secondary containment system to permit inspection and monitoring of functional capability. The determination of the depressurization time, the uniformity of negative pressure throughout the secondary containment and other contiguous areas, and the potential for exfiltration should be included in the preoperational and periodic test programs.

7. All openings, such as personnel doors and equipment hatches, should be under administrative control. These openings should be provided with position indicators and alarms having readout and alarm capability in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and confirmatory preoperational tests conducted.

Some plants may have only portions of the primary containment enclosed, rather than having a secondary containment structure or shield building that completely encloses the primary containment. These enclosed areas are areas into which the primary containment would most likely leak, and they may be equipped with air filtration systems. Quantitative credit cannot be given for the holdup effect of these enclosed areas or for the air filtration systems, to mitigate the radiological consequences of a postulated accident, unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed on a case-by-case basis.

III. REVIEW PROCEDURES

The procedures described below provide guidance on the review of the secondary containment system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of secondary containment functional design common to a class of plants, or by adopting the results of previous reviews of similar plants.

CSB reviews the analytical models used and the assumptions made in the analyses of the pressure and temperature response of the secondary containment to loss-of-coolant accidents in the primary containment. In general, CSB determines that the analyses conservatively predict the secondary containment pressure response. In so doing, CSB compares the analyses to the guidelines in Section II.

If considered necessary, CSB performs confirmatory analyses of the pressure and temperaturae response of the secondary containment for loss-of-coolant accidents within the primary containment for for high energy line (e.g., steam line and feedwater line) ruptures occurring within the secondary containment. The analyses are done using the CONTEMPT-LT computer code (Ref. 6). It should be noted that for the analysis of the pressure and temperature response in the secondary containment for loss-of-coolant accidents within the primary containmeht, the present version of the CONTEMPT-LT only has the capability of calculating the pressure in the secondary containment up to the time of peak pressure. The code is being improved to permit the calculation of the pressure response for the entire course of an accident.

The analysis will be based on the guidelines given in Section II, and code input data obtained from the SAR. CSB determines that the secondary containment design pressure is not exceeded and that the depressurization time is consistent with that assumed in the AAB analysis of the radiological consequences of the accident. In addition, CSB determines that the primary containment external design pressure is not exceeded.

CSB determines that all direct leakage paths have been properly identified, and from a review of the proposed leakage testing program that provisions have been made in the design of the plant to measure the fraction of total primary containment leakage that bypasses the secondary containment. The acceptability of the leakage testing program is considered in Standard Review Plan 6.2.6. CSB advises AAB of any inadequacies in the applicant's direct leakage assumptions used in the radiological analysis. At the operating license stage of review, CSB reviews technical specifications which specify the surveillance requirements for leakage testing of the secondary containment bypass leakage paths.

CSB reviews analyses of the capability of the secondary containment system to resist exfiltration under post-accident conditions. If the secondary containment leakage rate is in excess of 100% of the volume per day, CSB advises AAB in order that they may perform a special exfiltration analysis. CSB reviews the preoperational and periodic inservice testing programs to assure that testing will be done to verify the extent of exfltration.

CSB reviews the proposed secondary containment system testing program and the surveillance requirements in the technical specifications (operating license stage) to assure that tests will be periodically conducted to verify that the prescribed negative pressure can be uniformly maintained throughout the secondary containment.

CSB reviews the proposed technical specifications to assure that adequate administrative control will be exercised over the secondary containment openings, such as personnel access doors and equipment hatches. CSB determines from the descriptive information in the SAR that all doors and hatches are provided with position indicators having readout and alarm capability inthe main control room. The CSB will ascertain that normally open doors were considered in the analyses of the functional capability of the secondary containment system.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"6.2.3 Secondary Containment Functional Design

The scope of review of the functional design of the secondary containment system for the   has included plan and elevation drawings, system drawings, and descriptive information. This system is provided to control the atmosphere within the secondary containment and contiguous areas. The review has included the applicant's proposed design bases and analyses of the functional capability of the secondary containment system.

"The basis for the staff's acceptance has been the conformance of the functional design and design bases to the Commission's regulations as set forth in the general design criteria, and to applicable guides, staff technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of structures or systems should be discussed.)

"The staff concludes that the secondary containment system design conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

2. 10 CFR Part 50, Appendix A, General Design Criterion 42, "Inspection of Containment Atmosphere Cleanip Systems."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems."

4. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

5. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

6. R. J. Wagner and L. L. West, "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear Company, August 1973.

7. Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," attached to this plan.

BRANCH TECHNICAL POSITION CSB 6-3

DETERMINATION OF BYPASS LEAKAGE PATHS IN DUAL CONTAINMENT PLANTS

A. BACKGROUND

The purpose of this branch position is to provide guidance in the determination of that portion of the primary containment leakage that will not be collected and processed by the secondary containment. Bypass leakage is defined as that leakage from the primary containment which can circumvent the secondary containment boundary and escape directly to the environment, i.e., bypasses the leakage collection and filtration systems of the secondary containment. This leakage component must be considered in the radiological analysis of a loss-of-coolant accident.

The secondary containment consists of a structure which completely encloses the primary containment and can be maintained at a pressure lower than atmospheric so that primary containment leakage can be collected or processed before release to the environment. The secondary containment may include an enclosure building which forms an annular volume around the primary containment, the auxiliary building where it completely encloses the primary containment, and other regions of the plant that are provided with leakage collection and filtration systems. Depressurization systems are provided as part of the secondary containment to decrease or maintain the secondary containment volume at a negative pressure.

All primary containment leakage may not be collected because (1) direct primary containment leakage can occur while the secondary containment is being depressurized and (2) primary containment leakage can bypass the secondary containment through containment penetrations and seals which do not terminate in the secondary containment.

Direct leakage from the secondary containment to the environment can occur whenever an outward positive differential pressure exists across the secondary containment boundary. The secondary containment can experience a positive pressure transient following a postulated loss-of-coolant accident in the primary containment as a result of thermal loading and infiltration from the environment and the primary containment that will occur until the depressurization systems become effective. An outward positive differential on the secondary containment wall can also be created by wind loads. In this regard, a "positive" pressure is defined as any pressure greater than -0.25 in. w.g. (water gauge), to account for wind loads and the uncertainty in the pressure measurements. Whenever the pressure in the secondary containment volume exceeds -0.25 in. w.g., the leakage-prevention function of the secondary containment is assumed to be negated. Since leakage from the secondary containment during positive pressure periods cannot be determined, the conservative assumption is made that, all primary containment leakage is released directly to the environment during these time periods. Therefore, it becomes necessary to determine the time periods during which these threshold conditions exist.

The existence and duration of periods of positive pressure within the secondary containment should be based on analyses of the secondary containment pressure response to postulated loss-of-coolant accidents within the primary containment and the effectiveness of the depressurization systems.

The evaluation of bypass leakage involves both the identification of bypass leakage paths and the determination of leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries. Penetrations that pass through both the primary and secondary containment may include a number of barriers to leakage (e.g., isolation valves, seals, gaskets, and welded joints). While each of these barriers aid in the reduction of leakage, they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths, each of these penetrations should be considered, together with the capability to test them for leakage in a manner similar to the containment leakage tests required by Appendix J to 10 CFR Part 50.

B. <u>BRANCH TECHNICAL POSITION</u>

1. A secondary containment structure should completely enclose the primary containment structure, with the exception of those parts of the primary containment that are imbedded in the soil, such as the base mat of the containment structure. For partial dual containment concepts, leak rates less than the design leak rate of the primary containment should not be used in the calculation of the radiological consequences of a loss-of-coolant accident, unless the magnitude of unprocessed leakage can be adequately demonstrated. Quantitative credit for leakage collection in a partial-dual containment will be reviewed on a case-by-case basis.

2. Direct leakage from the primary containment to the environment, equivalent to the design leak rate of the primary containment, should be assumed to occur following a postulated loss-of-coolant accident whenever the secondary containment volume is at a "positive" pressure; i.e., a pressure greater than -0.25 in. w.g. Positive pressure periods should be determined by a pressure response analysis of the secondary containment volume that includes thermal loads from the primary containment and infiltration leakage.

3. The secondary containment depressurization and filtration systems should be designed in accordance with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants." Preoperational and periodic inservice inspection and test programs should be proposed for these systems and should include means for determining the secondary containment infiltration rate, and the capability of the systems to draw down the secondary containment to the prescribed negative pressure in a prescribed time.

4. For secondary containments with design leakage rates greater than 100 volume percent per day, an exfiltration analysis should be provided.

5. The following leakage barriers in paths which do not terminate within the secondary containment should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment:

a. Isolation valves in piping which penetrates both the primary and secondary containment barriers.

b. Seals and gaskets on penetrations which pass through both the primary and secondary containment barriers.

c. Welded joints on penetrations (e.g., guard pipes) which pass through both the primary and secondary containment barriers.

6. The total leakage rate for all potential bypass leakage paths, as identified in item 5 above, should be determined in a realistic manner, considering equipment design limitations and test sensitivities. This value should be used in calculating the offsite radiological consequences of postulated loss-of-coolant accidents and in setting technical specification limits with margin for bypass leakage.

7. Provisions should be made to permit preoperational and periodic leakage rate testing in a manner similar to the Type B or C tests of Appendix J to 10 CFR Part 50 for each bypass leakage path listed in item 5 above. An acceptable alternate for local leakage rate testing for welded joints would be to conduct a soap bubble test of the welds concurrently with the integrated (Type A) leakage test of the primary containment required by Appendix J. Any detectable leakage determined in this manner would require repair of the joint.

8. If air or water sealing systems or leakage control systems are proposed to process or eliminate leakage through valves, these systems should be designed, to the extent practical, using the guidelines for leakage control systems given in Branch Technical Position APCSB 6-1 (Ref. 3).

9. If a closed system is proposed as a leakage boundary to preclude bypass leakage, then the system should:

a. Either (1) not directly communicate with the containment atmosphere, or (2) not directly communicate with the environment, following a loss-of-coolant accident.

b. Be designed in accordance with Quality Group B standards, as defined by Regulatory Guide 1.26. (Systems designed to Quality Group C or D standards that qualify as closed systems to preclude bypass leakage will be considered on a case-by-case basis.)

c. Meet seismic Category I design requirements.

d. Be designed to at least the primary containment pressure and temperature design conditions.

e. Be designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features.

f. Be tested for leakage, unless it can be shown that during normal plant operations the system integrity is maintained.

C. REFERENCES

1. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

3. Branch Technical Position APCSB 6-1, "Main Steam Isolation Valve Leakage Control Systems," attached to Standard Review Plan 6.7.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)
Electrical, Instrumentation and Control System Branch (EICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products from postulated accidents. This plan, therefore, is concerned with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. Isolation barriers include valves, closed piping systems, and blind flanges.

The CSB reviews the information presented in the applicant's safety analysis report (SAR) regarding containment isolation provisions. The CSB review covers the following aspects of containment isolation:

1. The design of containment isolation provisions, including:

   a. The number and location of isolation valves, i.e., the isolation valve arrangements and the physical location of isolation valves with respect to the containment.

   b. The actuation and control features for isolation valves.

   c. The positions of isolation valves for normal plant operating conditions (including shutdown), post-accident conditions, and in the event of valve operator power failures.

   d. The valve actuation signals.

   e. The basis for selection of closure times of isolation valves.

   f. The mechanical redundancy of isolation devices.

g. The acceptability of closed piping systems inside containment as isolation barriers.

2. The protection provided for containment isolation provisions against loss of function from missiles, pipe whip, and earthquakes.

3. The environmental conditions inside and outside the containment that were considered in the design of isolation barriers.

4. The design criteria applied to isolation barriers and piping.

5. The provisions for detecting a possible need to isolate remote-manual-controlled systems, such as engineered safety features systems.

6. The design provisions for and technical specifications pertaining to operability and leakage rate testing of the isolation barriers.

EICSB has review responsibility for the qualification test program for electric valve operators, and the sensing and actuation instrumentation of the plant protection system that is located both inside and outside of containment; The MEB has review responsibility for the qualifications test program to demonstrate the performance and reliability of containment isolation valves; The MEB and SEB have review responsibility for the structural design of the containment isolation provisions to ensure adequate protection against missiles, pipe whip, and earthquakes.

II. ACCEPTANCE CRITERIA

The general design criteria establish requirements for isolation barriers in lines penetrating the primary containment boundary. In general, two isolation barriers in series are required to assure that the isolation function is satisfied assuming any single active failure in the containment isolation provisions.

The design of the containment isolation provisions will be acceptable to CSB if the following criteria are satisfied:

1. General Design Criteria 55 and 56 require that lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or connect directly to the containment atmosphere should be provided with isolation valves as follows:

   a. One locked closed isolation valve[1/] inside and one locked closed isolation valve outside containment; or

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1/ Locked closed isolation valves are defined as sealed closed barriers (see item II.3.f).

b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

c. One locked closed isolation valve inside and one automatic isolation valve[2] outside containment; or

d. One automatic isolation valve inside and one automatic isolation valve[2] outside containment.

2. General Design Criterion 57 requires that lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere should be provided with at least one locked closed, remote-manual, or automatic isolation valve[2] outside containment.

3. The general design criteria permit containment isolation provisions for lines penetrating the primary containment boundary that differ from the explicit requirements of General Design Criteria 55 and 56 if the basis for acceptability is defined. Following are guidelines for acceptable alternate containment isolation provisions for certain classes of lines:

   a. Regulatory Guide 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines that are closed both inside and outside containment, are designed to withstand the pressure and temperature conditions following a loss-of-coolant accident, and are designed to withstand dynamic effects, are acceptable without isolation valves.

   b. Containment isolation provisions for lines in engineered safety features or engineered safety feature-related systems may include remote-manual valves, but provisions should be made to detect possible leakage from these lines outside containment.

   c. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but provision should be made to detect possible leakage from these lines outside containment.

   d. Containment isolation provisions for lines in the systems identified in items b and c normally consist of one isolation valve inside and one isolation valve outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearest the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled leakage housing, or additional conservatism should be used in the design of this section of piping.

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[2] A simple check valve is not normally an acceptable automatic isolation valve for this application.

e. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve outside containment will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I standards, classified Safety Class 2 (Ref. 5), and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested, unless it can be shown that the system is being maintained during normal plant operations.

f. Sealed closed barriers may be used in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

g. Relief valves may be used as isolation valves provided the relief set point is greater than 1.5 times the containment design pressure.

4. Isolation valves outside containment should be located as close to the containment as practical, as required by General Design Criteria 55, 56, and 57.

5. The position of an isolation valve for normal and shutdown plant operating conditions and post-accident conditions depends on the fluid system function. If a fluid system does not have a post-accident function, the isolation valves in the lines should be automatically closed. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. The position of an isolation valve in the event of power failure to the valve operator should be the "safe" position. Normally this position would be the post-accident valve position. All power-operated isolation valves should have position indication in the main control room.

6. There should be diversity in the parameters sensed for the initiation of containment isolation.

7. Containment isolation valve closure times should be selected to assure rapid isolation of the containment following postulated accidents. System design capabilities should be considered in establishing valve closure times. For lines which provide an open path from the containment to the environs; e.g., the containment purge and vent lines,

isolation valve closure times on the order of 5 seconds or less may be necessary. The closure times of these valves should be established on the basis of minimizing the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and assure that emergency core cooling system (ECCS) effectiveness is not degraded by a reduction in the containment backpressure. Analyses of the radiological consequences and the effect on the containment backpressure due to the release of containment atmosphere should be provided to justify the selected valve closure time. Additional guidance on the design and use of containment purge systems is provided in Branch Technical Position CSB 6-4 (Ref. 9).

8. The use of a closed system inside containment as one of the isolation barriers will be acceptable if the design of the closed system satisfies the following requirements:

   a. The system does not communicate with either the reactor coolant system or the containment atmosphere.

   b. The system is protected against missiles and pipe whip.

   c. The system is designated seismic Category I.

   d. The system is classified Safety Class 2 (Ref. 5).

   e. The system is designed to withstand temperatures at least equal to the containment design temperature.

   f. The system is designed to withstand the external pressure from the containment structural acceptance test.

   g. The system is designed to withstand the loss-of-coolant accident transient and environment.

   Insofar as CSB is concerned with the structural design of containment internal structures and piping systems, the protection of isolation barriers against loss of function from missiles, pipe whip, and earthquakes will be acceptable if isolation barriers are located behind missile barriers, pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment, and the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake, as recommended by Regulatory Guide 1.29.

9. The design criteria applied to components performing a containment isolation function, including the isolation barriers and the piping between them, or the piping between the containment and the outermost isolation barrier, is acceptable if:

   a. Group B quality standards, as defined in Regulatory Guide 1.26, are applied to the components, unless the service function dictates that Group A quality standards be applied.

b. The components are designated seismic Category I, in accordance with Regulatory Guide 1.29.

10. The design of the containment isolation system is acceptable if provisions are made to allow the operator in the main control room to know when to isolate by remote-manual means fluid systems that have a post-accident safety function. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

11. Provisions should be made in the design of the containment isolation system for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. The acceptance criteria for the leakage rate testing program for containment isolation barriers are presented in Standard Review Plan 6.2.6.

III. REVIEW PROCEDURES

The procedures described below provide guidance on review of the containment isolation system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of containment isolation common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same containment isolation provisions.

The CSB determines the acceptability of the containment isolation system by comparing the system design criteria to the design requirements for an engineered safety feature. The quality standards and the seismic design classification of the containment isolation provisions, including the piping penetrating the containment, are compared to Regulatory Guides 1.26 and 1.29, respectively.

The CSB also ascertains that no single fault can prevent isolation of the containment. This is accomplished by reviewing the containment isolation provisions for each line penetrating the containment to determine that two isolation barriers in series are provided, and in conjunction with the EICSB by reviewing the power sources to the valve operators.

The CSB reviews the information in the SAR justifying containment isolation provisions which differ from the explicit requirements of General Design Criteria 55, 56 and 57. The CSB judges the acceptability of these containment isolation provisions based on a comparison with the acceptance criteria given in Section II.

The CSB reviews the position of isolation valves for normal and shutdown plant operating conditions, post-accident conditions, and valve operator power failure conditions as listed in the SAR. The position of an isolation valve for each of the above conditions depends on the system function. In general, power-operated valves in fluid systems which do not have a post-accident safety function should close automatically. In the event of power failure to a valve operator, the valve position should be the position of greater safety, which is normally the post-accident position. However, special cases may arise and these will be considered on an individual basis in determining the acceptability of the prescribed valve positions. The CSB also ascertains from the SAR that all power-operated isolation valves have position indication capability in the main control room.

The CSB reviews the signals obtained from the plant protection system to initiate containment isolation. In general, there should be a diversity of parameters sensed; e.g., abnormal conditions in the reactor coolant system, the secondary coolant system, and the containment, which generate containment isolation signals. Since plant designs differ in this regard and many different combinations of signals from the plant protection system are used to initiate containment isolation, the CSB considers the arrangement proposed on an individual basis in determining the overall acceptability of the containment isolation signals.

The CSB reviews isolation valve closure times. In general, valve closure times should be less than one minute, regardless of valve size. (See the acceptance criteria for valve closure times in Section II.) Valves in lines that provide a direct path to the environs, e.g., the containment purge and ventilation system lines and main steam lines for direct cycle plants, may have to close in times much shorter than one minute. Closure times for these valves may be dictated by radiological dose analyses or ECCS performance considerations. Supporting analyses justifying valve closure times for these lines should be provided in the safety analysis report for the CSB and AAB review.

The CSB determines the acceptability of the use of closed systems inside containment as isolation barriers by comparing the system designs to the acceptance criteria specified in Section II.

The MEB and SEB have review responsibility for the structural design of the containment internal structures and piping systems, including restraints, to assure that the containment isolation provisions are adequately protected against missiles, pipe whip, and earthquakes. The CSB determines that for all containment isolation provisions, missile protection and protection against loss of function from pipe whip and earthquakes were design considerations. The CSB reviews the system drawings (which should show the locations of missile barriers relative to the containment isolation provisions) to determine that the isolation provisions are protected from missiles. The CSB also reviews the design criteria applied to the containment isolation provisions to determine that protection against dynamic effects, such as pipe whip and earthquakes, was considered in the design.

Systems having a post-accident safety function may have remote-manual isolation valves in the lines penetrating the containment. The CSB reviews the provisions made to detect leakage from these lines outside containment and to allow the operator in the main control room to isolate the system train should leakage occur. Leakage detection provisions may include instrumentation for measuring system flow rates, or the pressure, temperature, radiation, or water level in areas outside the containment such as valve rooms or engineered safeguards areas. The CSB bases its acceptance of the leakage detection provisions described in the SAR on the capability to detect leakage and identify the lines that should be isolated.

The CSB determines that the containment isolation provisions are designed to allow the isolation barriers to be individually leak-tested. This information should be tabulated in the safety analysis report to facilitate the CSB review.

The CSB determines from the descriptive information in the SAR that provisions have been made in the design of the containment isolation system to allow periodic operability testing of the power-operated isolation valves and the containment isolation system. At the operating license stage of review, the CSB determines that the content and intent of proposed technical specifications pertaining to operability and leak testing of containment isolation equipment is in agreement with requirements developed by the staff.

IV. EVALUATION FINDINGS

The information provided and the CSB review should support concluding statements similar to the following, to be included in the staff's safety evaluation report:

"6.2.4 Containment Isolation System

The scope of review of the containment isolation system for the (plant name) has included schematic drawings and descriptive information for the isolation provisions for fluid systems which penetrate the containment boundary. The review has also included the applicant's proposed design bases for the containment isolation provisions, and analyses of the functional capability of the containment isolation system.

"The basis for the staff's acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, staff technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to specific containment isolation provisions or the functional capability of the containment isolation system should be discussed.)

"The staff concludes that the containment isolation system design conforms to all applicable regulations, guides, staff positions, and industry codes and standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

2. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

3. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

4. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

5. ANSI N271 (Draft, November 1974), "Containment Isolation Provisions," American National Standards Institute.

6. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

8. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

9. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," attached to this plan.

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the purge and vent lines of the containment purge system. It supplements the position taken in Standard Review Plan 6.2.4.

While the containment purge system provides plant operational flexibility, its design must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on its use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously.

The containment purge system has been used in a variety of ways, for example, to alleviate certain operational probelms, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. However, the purge and vent lines provide an open path from the containment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR 100 guideline values.

The sizing of the purge and vent lines in most plants has been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on non-routine basis during normal plant operation can be achieved by providing additional purge and vent lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge and vent valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge and vent valves would be closed before the onset of fuel failures following a LOCA.

The size of the purge and vent lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR or HTGR.

B. BRANCH TECHNICAL POSITION

The system used to purge the containment for the reactor operational modes of power operation, startup, and hot standby; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of hot shutdown, cold shutdown, and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. The performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in MEB Branch Technical Position MEB-2, Pump and Valve Operability Assurance Program. (Also see Standard Review Plan 3.9.3.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the purge line and vent line flows as a function of time up to and during valve closure.

   b. The number of purge and vent lines that may be used should be limited to one purge line and one vent line.

   c. The size of the purge and vent lines should not exceed about eight inches in diameter unless detailed justification for larger line sizes is provided.

   d. The containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria.

   e. Instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment pressure, safety injection actuation, and containment radiation level. If energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can affect the isolation function.

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves, individually, during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:

   a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the vent and purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR 100 guideline values.

   b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

   c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The allowable leak rates of the purge and vent isolation valves should be specified for the spectrum of design basis pressures and flows against which the valves must close.

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U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

CSB reviews the information presented in the applicant's safety analysis report (SAR) concerning the control of combustible gases in the containment following a loss-of-coolant accident. Following a loss-of-coolant accident, hydrogen and oxygen may accumulate inside the containment. The major sources of hydrogen and oxygen are: a chemical reaction between the fuel rod cladding and steam, the corrosion of aluminium and other materials by an alkaline spray solution, and the radiolytic decomposition of the water in the reactor core and the containment sump. If excessive hydrogen is generated, it may combine with oxygen in the containment atmosphere. For inerted containments, the potential exists for hydrogen to combine with oxygen generated following the accident. The CSB review includes the following general areas:

1. The production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident.

2. The capability to mix the combustible gases with the containment atmosphere and prevent high concentrations of combustible gases in local areas.

3. The capability to monitor combustible gas concentrations within containment.

4. The capability to reduce combustible gas concentrations within containment by suitable means, such as recombination, dilution, or purging.

The CSB review specifically covers the following analyses and aspects of combustible gas control system designs:

1. An analysis of combustible gas (i. e., hydrogen and oxygen) production and accumulation within the containment following a loss-of-coolant accident.

2. An analysis of the functional capability of the systems provided to mix the combustible gas within the containment.

3. An analysis of the functional capability of the systems provided to reduce combustible gas concentrations within the containment.

4. Analyses of the capability of systems or system components to withstand dynamic effects, such as transient differential pressures that would occur early in the blowdown phase of a loss-of-coolant accident.

5. Analyses of the consequences of single active component malfunctions in each system.

6. The quality classification of each system.

7. The seismic design classification of each system.

8. The results of qualification tests performed on system components to demonstrate functional capability and operability in the accident environment.

9. The design provisions and proposed program (including technical specifications at the operating license stage of review) for periodic inservice inspection and operability testing of each system or component.

10. The functional aspects of instrumentation provided to monitor system or system component performance.

11. The extent of sharing of system components between sites or between units at a multi-unit site.

AAB is responsible for determining, from a radiological dose standpoint, the acceptability of purge systems provided to control combustible gas concentrations within the containment following a loss-of-coolant accident. In order to compute the purge doses, AAB will need the elapsed time (in days) following a loss-of-coolant accident before purge system operation becomes necessary and the purge rate (in scfm). CSB provides AAB with this information.

At the construction permit (CP) stage of review, the design of the systems provided for monitoring and reducing the concentrations of combustible gases within the containment may not be completely determined. In such cases, CSB reviews the applicant's preliminary designs and statements of intent to comply with the acceptance criteria for such systems. At the operating license (OL) stage, CSB reviews the final designs of these systems to verify that they meet the acceptance criteria detailed below.

II. ACCEPTANCE CRITERIA

1. The analysis of hydrogen and oxygen production in the containment following postulated accidents, for the purpose of establishing the design basis for combustible gas control systems, should be based on the parameters listed in Table 1 of Branch Technical Position CSB 6-2. Branch Technical Position (BTP) CSB 6-2 is an acceptable interim alternative to Regulatory Guide 1.7, pending completion of the rulemaking proceeding on inerting ordered by the Commission in connection with the Vermont Yankee matter, Docket No. 50-271, Memorandum and Order, November 7, 1974, and subsequent revision of Regulatory Guide 1.7. BTP CSB 6-2 supplements and amends Regulatory Guide 1.7 as necessary to take account of the progress in engineered safety feature designs and standards since the guide was written and various features of recent containment designs.

2. The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis of the emergency core cooling water and sump water is acceptable if it is equal to or more conservative than the decay energy model given in Branch Technical Position APCSB 9-2 in Standard Review Plan 9.2.5.

3. A system should be provided to mix the combustible gases within the containment. The functional design of this system will depend on the type of containment. This system may consist of a fan, a fan cooler, or containment spray. An analysis should be presented which shows that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. For containments which rely on convective mixing in conjunction with system operation to mix the combustible gases, the containment internal structures must have design features which promote the free circulation of the atmosphere. An analysis of the effectiveness of these features for convective mixing should be presented. This analysis is acceptable if it can be shown that combustible gases will not accumulate within a compartment or cubicle to form an explosive mixture.

4. The systems provided to reduce the concentration of hydrogen or oxygen in the containment will be accepted, from a functional standpoint, if analyses indicate that a single system train is capable of maintaining the concentration of hydrogen or oxygen below the concentration limits specified in Table 1 of BTP CSB 6-2. Acceptance of the functional capability of the systems is based on confirmatory analyses performed by CSB using system operating parameters presented in the safety analysis report. The proposed operation of the combustible gas control equipment, excluding containment atmosphere dilution (CAD) systems, is acceptable if there is an approprate margin, e.g., on the order of 0.5 v/o, between the limiting hydrogen concentration limit and the hydrogen concentration at which the equipment would be actuated. The proposed operation of CAD systems will be acceptable if there is a margin of 1 v/o between the limiting hydrogen or oxygen concentration limit, depending on which gas being controlled, and the concentration at which the system would be actuated. This additional margin is needed to allow time for the CAD system to become operational. Repressurization of the containment should be limited to less than 50% of the containment design pressure.

Under loss-of-coolant accident conditions, system components such as ductwork and equipment housings, e.g., for fans, fan-coolers, filters, and recombiners, would be subjected to external transient differential pressures and internal pressure surges. These components should be capable of withstanding all related environmental conditions imposed on them, including steam-laden atmosphere differential pressures and pressure surges, without loss of function. A description of the design provisions, such as pressure relief devices or conservative structural design, supporting analyses, and results of tests should be provided to support the conservatism of design.

5. Combustible gas control systems should meet the redundancy and power source requirements for engineered safety features and should be designed to withstand a single active component failure. Supporting failure mode and effects analyses of each system should be provided in the safety analysis report.

6. Combustible gas control systems should be designed, fabricated, erected, and tested to Group B quality standards, as recommended in Regulatory Guide 1.26.

7. Combustible gas control systems, including foundations and supports, should be designated as seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake without loss of function, as recommended in Regulatory Guide 1.29.

8. Qualification tests should be performed on system components, such as hydrogen recombiners, combustible gas analyzers, air moving equipment motors, and valve operators. The tests should support the analyses of the functional capability of the equipment and demonstrate that the equipment will remain operable in the accident environment for as long as accident conditions require.

9. Combustible gas control systems should be designed with provisions for periodic inservice inspection and operability testing of the systems or components. The inspection and test program is acceptable if it is judged to be consistent with that proposed for other engineered safety features.

10. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room.

11. The sharing of system equipment between nuclear power units at a multi-unit site or between sites is acceptable provided (a) the availability of the shared equipment meets the redundancy requirements for an engineered safety feature, (b) the shared equipment is designed to seismic Category I criteria, (c) the shared equipment is mounted in a seismic Category I structure, and (d) adequate design, installation, and procedural provisions have been made.

12. BTP CSB 6-2 recommends that a backup purge system be provided. The backup purge system is not required to be designed to engineered safety feature requirements with regard to single failure protection since it is not the primary method for controlling combustible gas concentrations in the containment. The backup purge system is acceptable if purge doses are within the guidelines established in BTP CSB 6-2.

13. If the designs of the combustible gas control systems have not been completed at the construction permit stage of review, they will be acceptable if the preliminary system designs and statements of intent in the SAR conform to BTP CSB 6-2.

III. REVIEW PROCEDURES

The procedures described below provide guidance for the detailed review of the combustible gas control systems. The reviewer selects and emphasizes material from this plan, as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of combustible gas control systems design common to a class of plants or by adopting the results of previous reviews of similar plants.

1. CSB reviews the applicant's analyses of the production and accumulation of oxygen and hydrogen in the containment following postulated loss-of-coolant accidents, to see that the recommendations and guidelines of BTP CSB 6-2 have been followed. With regard to the extent of metal-water reaction to be considered, the combustible gas control system designs of some boiling water reactor plants with BWR6/Mark III containments have been evaluated and accepted on the basis of an assumed metal-water reaction involving one percent of the cladding mass. Since this assumption is conservative with respect to BTP CSB 6-2 (the BTP would indicate about 0.7% reaction of the cladding mass in these cases), it will continue to be an acceptable basis for these plants, at the option of the applicants. As necessary, the CSB will make confirmatory analyses of combustible gas production and accumulation. These analyses are done using the COGAP computer code, a description of which is attached as Appendix A to this plan. The safety analysis report should contain the required code input data. The purposes of the analyses are:

   a. To confirm the predictions of hydrogen and oxygen generation appearing in the safety analysis report.
   b. To verify that the systems provided for combustible gas control are capable of maintaining the concentrations of hydrogen and oxygen below the concentration limits specified in Table 1 of BTP CSB 6-2.
   c. To confirm the elapsed time before purge system operation becomes necessary.
   d. To confirm that the assumed purge rate will maintain combustible gas concentrations within acceptable limits.

   The above analyses should be done early in the plant review, since this information is needed by AAB to perform the purge dose computations upon which the acceptability of the purge system is based.

2. The combustible gas control systems include systems for mixing the combustible gases, monitoring combustible gas concentrations, and reducing the combustible gas concentrations. In general, all of the combustible gas control systems should meet the design requirements for engineered safety features, as outlined in Section II. The system description and schematic drawings presented in the safety analysis report should be sufficiently detailed to permit judgments to be made regarding system acceptability.

   CSB determines that all potential, single active mechanical failures and passive electrical failures have been identified and that no single failure would incapacitate the entire system. Passive mechanical failures, beyond those possible from missile impact, need not be considered in view of the design and construction standards for the systems.

   CSB compares the quality standards applied to the systems to Regulatory Guide 1.26.

   CSB compares the seismic design classifications of the systems to Regulatory Guide 1.29.

3. CSB reviews the environmental conditions and duration of tests used for the qualification of system components. CSB determines whether the test conditions and duration are representative of post-accident conditions to which the equipment may be subjected. CSB will ascertain that the equipment can operate in the accident environment for as long as accident conditions require.

4. CSB reviews the provisions made in the design of the systems and the program for periodic inservice inspection and operability testing of the systems or components. The inspections are reviewed with regard to the purpose of each inspection. The operability tests that will be conducted are reviewed with regard to what each test is intended to accomplish. Judgment and experience from previous reviews are used to determine the acceptability of the inspection and test program.

   For plants at the operating license stage of review, CSB reviews the proposed technical specifications for the systems used to control combustible gas concentrations in the containment to assure that the intent of General Design Criteria 41, 42, and 43 are met.

5. CSB reviews the capability to monitor system performance and control active components to be sure that control can be exercised over a system and that a malfunctioning system train or component can be isolated. The instrumentation provided for this purpose should be redundant and should enable the operator to identify the malfunctioning system train or component.

6. CSB reviews the extent of sharing of system equipment between plants at multi-unit sites or between sites to assure that system redundancy requirements are satisfied and that adequate procedural provisions have been made to assure the availability of the shared equipment on a timely basis. The results of CSB analyses of combustible gas production and accumulation are used to confirm the time available following postulated loss-of-coolant accidents to transport the shared equipment to the plant and put it into operation.

7. CSB reviews analyses of the functional capability of the systems provided to mix combustible gases within the containment. CSB reviews the supporting information in the safety analysis report which should include elevation drawings of the containment showing the routing of ductwork and the circulation patterns caused by fans, sprays, or thermal convection. Special attention is paid to interior compartments to assure that combustible gases cannot collect in them without mixing with the bulk containment atmosphere. CSB ensures that interior compartments are identified in the safety analysis report and the provisions made to assure circulation within them are discussed.

   Systems provided to mix the combustible gases within the containment may also be used for containment heat removal, e.g., the fan cooler and spray systems. The acceptability of the design of these systems is considered in the review of the containment heat removal systems in Standard Review Plan 6.2.2.

8. CSB reviews the manner in which the systems provided to reduce combustible gas concentrations will be operated. The concentration at which the system is actuated (the control point) will be determined from the safety analysis report. The margin between the control point and the hydrogen or oxygen concentration limits specified in Table 1 of BTP CSB 6-2 is checked. CSB determines whether the uncertainty in measuring combustible gas concentrations and the time lag in making the system operational after reaching the control point have been covered by the minimum allowable margin specified in the acceptance criteria.

9. At the construction permit stage of review, the design of the combustible gas control systems may not be complete. In such cases, CSB reviews the preliminary design information and the design criteria that have been established.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The scope of review of the design and functional capability of the combustible gas control systems for the ______________ plant has included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentrations, and reduce combustible gas concentrations within the containment following the design basis accident. The review has also included the applicant's proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the systems provided to support the adequacy of the design bases.

"The basis for the staff's acceptance has been the conformance of system designs and design bases to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of the combustible gas control systems should be discussed.)

"The staff concludes that the design of the combustible gas control systems conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

2. 10 CFR Part 50, Appendix A, General Design Criterion 42, "Inspection of Containment Cleanup System."

3. 10 CFR Part 50, Appendix A, General Design Criterion 43, "Testing of Containment Atmosphere Cleanup System."

4. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

5. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and the Supplement to Regulatory Guide 1.7.

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards."

7. Regulatory Guide 1.29, "Seismic Design Classification."

8. L. Baker, Jr., and L. C. Just, "Studies of Metal-Water Reaction at High Temperature, III Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, Argonne National Laboratory, May 1962.

9. J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962.

10. H. F. Coward and G. W. Jones, "Limits of Flammability of Gases and Vapors," Bulletin 503, Bureau of Mines (1952).

11. A. O. Allen, "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Co., New York (1961).

12. Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," attached to Standard Review Plan 9.2.5.

13. Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations In Containment Following a Loss of Coolant Accident," attached to this plan.

APPENDIX A

STANDARD REVIEW PLAN 6.2.5

DESCRIPTION OF COGAP

INTRODUCTION

A digital computer program, COGAP (Combustible Gas Analyzer Program), has been developed by the Containment Systems Branch to provide in-house capability for determining hydrogen-oxygen concentrations within reactor containments following loss-of-coolant accidents. The program can also evaluate the performance of a number of combustible control systems. They are the containment atmosphere dilution system (CAD), the recombiner system, and the backup purge system.

DISCUSSION

In the event of a loss-of-coolant accident (LOCA), hydrogen and oxygen gases will be generated within the reactor containment by several reactions. They are:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant, producing free hydrogen.

2. Radiolytic decomposition of the post-accident emergency cooling solutions, producing both oxygen and hydrogen.

3. Aluminum corrosion by water solutions, producing hydrogen.

4. Zirc corrosion by water solutions, producing hydrogen.

If a sufficient amount of hydrogen is generated, it may react with the O_2 present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following a LOCA.

The extent of zirc-water reaction and associated hydrogen production depends strongly on the course of events assumed for the accident. Analytically the reaction can be described by:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

$$1 \text{ lb } Zr \rightarrow 0.043956 \text{ lb } H_2$$

$$1 \text{ lb } Zr \rightarrow 0.021978 \text{ lb-mole } H_2.$$

Therefore, one pound of reacted zirconium will produce 0.021978 pound-moles of free hydrogen. Assuming the perfect gas relationship, this is equivalent to 8.4866 scf/lb Zr:

$$V = \frac{MRT}{P}$$

$$V = \frac{0.021978(10.71)(530)}{14.7}$$

$$V = 8.4866 \text{ scf/lb Zr.}$$

The total amount of hydrogen produced is based on the amount of reacted zirconium, as determined by the assumptions given in Branch Technical Position CSB 6-2. The computer program, to maintain a degree of generality, allows the reaction percentage to be specified as an input quantity. The expression used is:

$$WG = (.022)(WZR)(f_{MW})$$

where

WG = pound moles of hydrogen generated

WZr = weight of zirconium fuel element clad

f_{MW} = zirconium-water reaction fraction.

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. The radiolytic hydrogen production rate at time (t) is given by:

$$S_H(t) = \frac{P}{(B)\ (N)} \quad \frac{G_cE_c(t) + G_sE_s(t)}{100}$$

where

$S_H(t)$ = hydrogen production rate, lb-mole/sec

P = operating reactor power level, MWt

B = conversion factor, 454 gm-mole/lb-mole

N = Avogadro's number, 6.023×10^{23} molecules/gm-mole

G_c = radiolytic hydrogen yield in core, molecules/100 ev

$E_c(t)$ = gamma ray fission product energy absorbed by core coolant, ev/sec-MWt

G_s = radiolytic hydrogen yield in solution, $\frac{\text{molecules}}{100\text{ ev}}$

$E_s(t)$ = energy absorbed in coolant outside core due to fission products dissolved in coolant, ev/sec-MWt.

The quantity $E_c(t)$ is defined by:

$$E_c(t) = (f_\gamma)_c\ H_\gamma(t)$$

where

$(f_\gamma)_c$ = fraction of fission product gamma energy absorbed by coolant in core region

$H_\gamma(t)$ = gamma energy production rate, $\frac{\text{ev}}{\text{sec-MWt}}$.

Similarly, $E_S(t)$ is defined by:

$$E_S(t) = (f_{\gamma+\beta})_S \, H_{\gamma+\beta}(t) + f_I \, H_I(t)$$

where

$(f_{\gamma+\beta})_S$ = fraction of total solid fission product energy absorbed in coolant outside core

$H_{\gamma+\beta}(t)$ = total solid fission product energy production rate, ev/sec-MWt

f_I = fraction of iodine isotope energy absorbed in coolant outside core

$H_I(t)$ = iodine isotope energy production rate, ev/sec-MWt.

The equations for oxygen generation by radiolysis are identical to those above describing hydrogen evolution except that the yield is one half that of hydrogen. These equations have been incorporated into the COGAP program. For calculational purposes, the reactor decay profiles ($H_\gamma(t)$, $H_{\gamma+\beta}(t)$, and $H_I(t)$) specified by the ANS-5.1 draft standard for two-year reactor operation have been fitted by several finite exponential series expressions and also incorporated into the program. The resulting equations are:

$$H_\gamma(t) = 10^{22}(5.1912e^{-9.8 \times 10^{-5}t} + 0.8743e^{-6.5 \times 10^{-6}t}$$

$$+ 0.6557e^{-5.7 \times 10^{-7}t} + .4098e^{-7.4 \times 10^{-8}t} + .0150e^{-8.0 \times 10^{-10}t})$$

$$H_{\gamma+\beta}(t) = 2.0 \, H_\gamma(t)$$

$$H_I(t) = 10^{22}(0.8197e^{-6.1 \times 10^{-5}t} + .3279e^{-1.1 \times 10^{-5}t}$$

$$+ .0574e^{-1.0 \times 10^{-6}t})$$

where

t = time after reactor shutdown, sec.

Between 400 and 4×10^7 sec, the equations overpredict the standard curve by 20%. The equations underpredict the standard curve soon after shutdown. However, this does not seriously affect the results due to the short time period involved. The equations are equivalent to the afterheat decay curve in BTP APCSB 9-2 over the times of interest for post-accident hydrogen generation. It should also be noted that the COGAP formulation overpredicts the radiolytic hydrogen generation by a small amount due to a "double-counting" of the gamma energy of those fission products assumed to be released from the fuel rods.

Hydrogen generation due to aluminum corrosion is normally considered only when additives are used in the cooling solution. When applicable, gas production is governed by the following expression:

$$S_c(t) = \frac{A\rho BC(t)}{(12)(3.15 \times 10^7)}$$

where

$S_c(t)$ = hydrogen production rate, lb-mole/sec

A = surface area of aluminum, ft^2

ρ = aluminum density, lb/ft^3

B = lb-moles of hydrogen per lb of aluminum

C(t) = aluminum corrosion rate, in/year.

The aluminum corrosion rate has been described by an exponential fit in COGAP to account for an increased rate due to high temperatures early in the accident followed by a constant rate for the remaining period of the analysis.

The chemical relationship by which hydrogen is formed has been assumed to be:

$$2\ Al + 3\ H_2O \rightarrow 3\ H_2 + Al_2O_3$$

$$1\ lb\ Al \rightarrow 0.111\ lb\ H_2$$

$$1\ lb\ Al \rightarrow 0.0555\ lb\text{-mole}\ H_2$$

therefore

$$B = 0.0555\ lb\text{-mole}\ H_2/lb\ Al$$

Zinc corrosion has been treated in a similar fashion.

COGAP INPUT REQUIREMENTS

COGAP has been developed to minimize the required input information. All data associated with the power decay profile has been incorporated into the program and need not be entered. Basic input requires eight input cards per case. Multiple cases can be stacked back to back, allowing an unlimited number of cases to be run at any given time.

The following is a detailed description of the data required per case:

1st card: title card.

Information contained within the first 72 columns will be printed as a general output heading. It should be used to describe the power plant under consideration.

2nd card: control card (right justified)(integers)

		columns			
5	10	15	20	25	30
I1	IH1	J1	K1	ITEMP	ICASE

I1 = total number of time steps considered (must not be greater than 50)(equal to IH1 + J1 + K1 + 2)

IH1 = number of time steps in initial time step grid

J1 = number of time steps in second time step grid

K1 = number of time steps in third time step grid

ITEMP = number of temperature points to be read

ICASE = 0 if this is last case

= 1 if another case following.

3rd card: time step information (floating)

	columns	
12	24	36
DELTA	DELTB	DELTC

DELTA = constant time step for first time grid, days

DELTB = constant time step for second time grid, days

DELTC = constant time step for third time grid, days.

4th card: containment data (floating)

		columns			
12	24	36	48	60	72
POW	V(1)	V(2)	ZIRWGT	O	H

POW = reactor power level, MWt

V(1) = containment free volume, ft^3

V(2) = 2nd containment free volume (wetwell), ft^3

ZIRWGT = zirconium cladding weight, pounds

O = oxygen dissolved in primary, pound-moles

H = hydrogen dissolved in primary, pound-moles.

5th card: containment data (continued)

		columns			
12	24	36	48	60	72
P	T	OF	QREC	TIME	PURG

P = initial containment pressure, psia

T = initial containment temperature, rankine

OF = initial oxygen volume fraction (.209 std. air)

QREC = recombiner flow rate, cfm

(Must be zero if purging is to be considered)

TIME = time recombiner is started, days

(program will start recombiner at time nearest but less than specified time)

PURG = purging rate, cfm (must be zero if recombiner is to be used).

6th card: gas constants (floating)

columns

12	24	36	48	60	71
f_{mw}	A	G_c	G_s	$(f_\gamma)_c$	$(f_{\gamma+\beta})_s$

f_{mw} = zirc-water reaction fraction

A = aluminum surface area, ft^2

G_c = G-H_2, core solution, mole/100 ev

G_s = G-H_2, sump solution, mole/100 ev

$(f_\gamma)_c$ = fraction of gammas absorbed in coolant in core region

$(f_{\gamma+\beta})_s$ = fraction of solid fission product energy absorbed in solution outside core

7th card: gas constants (floating)

columns

12	24	36	48	60	72
f_I	BLANK	D	HF	TI	FLOW

f_I = fraction of iodine fission product energy absorbed in solution outside core

D = time constant >9.0 x 10^8

HF = H_2 concentration fraction at which purging will begin

TI = time to initiate nitrogen addition, sec

FLOW = CAD nitrogen flow rate, scf/sec.

8th card: temperature profile

columns

12	24	36	48	60	72
T(1)	T(2)	T(3)			T(ITEMP)

T(1) = containment temperature, rankine (for first time increment)

T(2) = containment temperature, rankine (for second time increment)

T(ITEMP) = containment temperature, rankine (for ITEMP time increment)

BRANCH TECHNICAL POSITION CSB 6-2

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT*

A. BACKGROUND

General Design Criterion 41 requires that systems to be provided as necessary to control the concentrations of hydrogen, oxygen, and other substances which may be released into the reactor containment following postulated accidents, to assure that containment integrity is maintained. General Design Criterion 50 requires, in part, that containment be designed to accommodate with margin "metal-water and other chemical reactions that may result from degraded emergency core cooling functioning." This branch technical position (BTP) describes an acceptable method of implementing these criteria for light water reactor plants with cylindrical, zircaloy-clad, oxide fuel. Evaluations of other light water reactor fuels, with stainless steel cladding or with non-cylindrical cladding, will continue to be made on an individual case basis.

Following a loss-of-coolant accident (LOCA), hydrogen gas may accumulate within the containment as a result of:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.

2. Radiolytic decomposition of the post-accident emergency cooling solutions (oxygen will also evolve in this process).

3. Corrosion of metals by solutions used for emergency cooling or containment spray.

If a sufficient amount of hydrogen is generated, it may react with the oxygen present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following the accident. The reaction would take place at rates rapid enough to lead to high temperatures and significant overpressurization of the containment, which could result in a leakage rate above that specified in the limiting conditions for operation (technical specifications). Damage to systems and components essential to the continued control of post-LOCA conditions could also occur.

The extent of metal-water reaction and associated hydrogen production depends strongly on the course of events assumed for the accident and on the effectiveness of emergency cooling systems. Evaluations of the performance of emergency core cooling systems (ECCS) included as engineered safety features on current light water cooled reactor plants have been made by reactor designers using analytical models described in the Commission's Interim Policy Statement of 1971. These calculations are further discussed in the staff's Concluding Statement in the rulemaking hearings, Docket RM-50-1. The result of such evaluations is that for plants of current design, operated in conformance with the Interim Policy Statement, the calculated metal-water reaction

* See Section II.1 of Standard Review Plan 6.2.5.

amounts to only a fraction of one percent of the fuel cladding mass. As a result of the rule-making hearing (Docket RM-50-1), the Commission has recently adopted new regulations dealing with the effectiveness of ECCS (10 CFR §50.46).

The staff believes it appropriate to consider the experience obtained from the various ECCS-related analytical studies and test programs such as code developmental efforts, fuel densification, blowdown and core heat-up studies, and the PWR and BWR FLECHT tests, and to take account of the foregoing increased conservatism, for plants with ECCS evaluated under §50.46, in setting the amount of initial metal-water reaction to be assumed for the purpose of establishing design requirements for combustible gas control systems. The staff has always separated the design bases for ECCS and for containment systems, and has required containment systems such as the combustible gas control system to be designed to withstand a more degraded condition of the reactor than the ECCS design basis permits. The approach is consistent with provisions of General Design Criterion 50 where the need to provide margins to account for the effects of degraded ECCS function is noted. Although the level of degradation considered might lead to an assumed extent of metal-water reaction in excess of that calculated for acceptable ECCS performance, it does not lead to a situation involving a total failure of the ECCS. The staff feels that this "overlap" in protection requirements provides an appropriate and prudent safety margin against unpredicted events during the course of accidents.

Accordingly, the staff believes that the amount of hydrogen assumed to be generated by metal-water reaction in establishing combustible gas control system performance requirements should be based on the amount calculated in demonstrating compliance with §50.46, but that the amount of hydrogen required to be assumed should include a margin above that calculated. To obtain this margin, the assumed amount of hydrogen should be no less than five times that calculated in accordance with §50.46.

Since the amounts of hydrogen thus determined may be quite small for many plants, as a result of the other more stringent requirements for ECCS performance in the criteria of §50.46, it is consistent with the consideration of the potential for degraded ECCS performance discussed above to establish also a lower limit on the assumed amount of hydrogen generated by metal-water reactions in establishing combustible gas control system requirements. In establishing this lower limit, the staff has noted that the maximum metal-water reaction permitted by the ECCS performance criteria is one percent of the cladding mass.* In fact, the designs of several plants of the BWR6-Mark III type using one percent of the cladding mass as a combustible gas control system basis have recently been reviewed and accepted by the staff and the Advisory Committee on Reactor Safeguards. These plants were reviewed on an individual case basis, since they were the first of the design type. The general and continued use of this "one percent of the mass" value as a lower limit for assumed hydrogen production, however, would unnecessarily penalize reactors with thicker cladding, since for the same thermal conditions in the core in a postulated LOCA the thicker cladding would not, in fact, lead to increased hydrogen generation. This is because the hydrogen generation from metal-water reaction is a surface phenomenon. The staff considers that a more appropriate basis for setting the lower limit would be an amount of hydrogen assumed to be generated per unit cladding area. It is convenient to specify for

*10 CFR Part 50, §50.46(b)(3) "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

this purpose a hypothetical uniform depth of cladding surface reaction. The lower limit of metal-water reaction hydrogen to be assumed is then the hypothetical amount that would be generated if all metal to a specified depth in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) were to react.

In selecting a specified depth to be assumed as a lower limit for all reactor designs, the staff has calculated the depth that could correspond to the "one percent of the mass" value for the current core design with the thinnest cladding. This depth (0.01 times the thickness of the thinnest fuel cladding in use) is 0.00023 inches.

In summary, the amount of hydrogen assumed to be generated by metal-water reaction in determining the performance requirements for combustible gas control systems should be five times the maximum amount calculated in accordance with §50.46, but no less than the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inches.

It should be noted that the extent of initial metal-water reaction calculated for the first core of a plant, and used as a design basis for the hydrogen control system, becomes a limiting condition for all reload cores in that plant unless the hydrogen control system is subsequently modified and reevaluated.

The staff believes that hydrogen control systems in plants receiving operating licenses on the basis of ECCS evaluations under the Interim Policy Statement should continue to be designed for the five percent initial metal-water reaction specified in the original edition of Safety Guide 7. As operating plants are reevaluated as to ECCS performance under 10 CFR §50.46, a change to the new hydrogen control basis enumerated above may be made by appropriate amendments to technical specifications. For plants receiving construction permits on the basis of ECCS evaluations under the Interm Policy Statement, the staff believes that a commitment by the applicant to a specified maximum metal-water reaction, as determined by the provisions of this BTP, is an acceptable alternate basis for the design of a hydrogen control system.

No assumption as to rate of evolution was associated with the magnitude of the assumed metal-water reaction originally given in Safety Guide 7. The metal-water reaction rate is of significance when establishing system performance requirements for containment designs that employ time-dependent hydrogen control features. The staff recognizes that it would be unrealistic to assume an instantaneous release of hydrogen from an assumed metal-water reaction. The staff believes that for the design of a hydrogen control system, it should be assumed that the initial metal-water reaction would occur over a short period of time early in the LOCA transient, i.e., near the end of the blowdown and core refill phases of the LOCA transient. Any hydrogen thus evolved would mix with steam and air and be rapidly distributed throughout the containment compartments enclosing the reactor primary coolant system by the steam flowing from the postulated pipe break. These compartments include the "drywell" in typical boiling water reactor containments, the "lower volume" of ice condenser containments, and the full volume of "dry" containments. The blowdown and refill phase duration is generally several minutes, and the staff believes that the assumption of a two-minute evolution time at a constant reaction rate, with the resulting hydrogen uniformly distributed in the containment compartments enclosing the primary coolant system, is appropriately conservative for the design of hydrogen control systems. The effects of steam within the containment and containment subcompartments should be considered in the evaluation of the mixture composition.

The rate of production of gases from radiolysis of coolant solutions depends on (1) the amount and quality of radiation energy absorbed in the specific coolant solutions employed and (2) the net yield of gases generated from the solutions due to the absorbed radiation energy. Factors such as coolant flow rates and turbulence, chemical additives in the coolant, impurities, and coolant temperature can all exert an influence on the gas yields from radiolysis. The hydrogen production rate from corrosion of materials within the containment, such as aluminum, depends on the corrosion rate which in turn depends on such factors as the coolant chemistry, the coolant pH, the metal and coolant temperatures, and the surface area exposed to attack by the coolant. Accurate values of these parameters are difficult to establish with certainty for the conditions expected to prevail following a loss-of-coolant accident.

The staff has reviewed the available information concerning these parameters, including the results of calculations and experiments. Table 1 defines values and other assumptions which the staff believes to be reasonably conservative that may be used for purposes of evaluating the production of combustible gases following a loss-of-coolant accident.

If these assumptions are used to calculate the concentration of hydrogen (and oxygen) within the containment structures of reactor plants following a loss-of-coolant accident, the hydrogen concentration is calculated to reach the flammable limit within periods of less than a day after the accident for the smallest containments and up to more than a month for the largest ones. The hydrogen concentration could be maintained below its lower flammable limit by purging the containment atmosphere to the environs at a controlled rate after the LOCA; however, radioactive materials in the containment would also be released. If purging became necessary shortly after the accident, quantities of such material would be released. The staff believes that the capability for controlled purging should be provided, but that purging should not be the primary means for controlling combustible gases following a LOCA.

The Bureau of Mines has conducted experiments at their facilities with initial hydrogen volume concentrations in the range of four to twelve volume percent. On the basis of these experiments, and of review of reports by others, the staff concludes that a lower flammability limit of four volume percent hydrogen in air or steam-air atmospheres is well established and is adequately conservative. For initial concentrations of hydrogen greater than about six volume percent, it is possible in the presence of sufficient ignition sources that the total accumulated hydrogen could burn in the containment. For hydrogen concentrations in the range of four to six volume percent, partial burning of the excess hydrogen above four volume percent may occur. The staff believes that a limit of six volume percent would not result in effects that would be adverse to containment systems. Applicants or licensees should demonstrate through supporting analyses and experimental data that containment features and safety equipment required to operate a LOCA would not be made inoperative by burning of the excess hydrogen, if a design limit in the range of four to six volume percent hydrogen is proposed.

In small containments, the amount of metal-water reaction postulated in Table 1 may result in hydrogen concentrations above acceptable limits. The evolution rate of hydrogen from the metal-water reaction would be greater than that from either radiolysis or corrosion, and since it is difficult for a hydrogen control system to process large volumes of hydrogen very rapidly, an alternative approach is to operate some of the smaller containments with inert (oxygen deficient) atmospheres. This measure, the so-called "inerting" of a containment, provides sufficient time for combustible gas control systems to reduce the concentration of hydrogen following a loss-of-coolant accident before the oxygen generated by radiolysis results in flammable mixtures in the containment. Any requirement for inerting of a containment should be considered on an individual case basis, taking into account the features of the plant, the details of the inservice inspection program for components inside containment, and the need for protection against possible effects from combustible gases.

For all containments, it is advisable to provide means for mixing, sampling, and control of combustible gases resulting from the postulated metal-water reaction, radiolysis, and corrosion following a LOCA, which do not involve releases of radioactive materials to the environment. It is also advisable, as a back-up measure, to provide the capability of purging the containment. Filters should be provided as needed in the purge stream to limit the potential release of radioactive iodine and other radioactive materials so that the calculated radiological consequences of the LOCA, including the purge, do not exceed the guideline doses given in 10 CFR Part 100.

Since any system for combustible gas control is designed for the protection of the public in the event of an accident, it should meet the design and construction standards of engineered safety features. Care should be taken in its design to assure that the system itself does not introduce safety problems that may affect containment integrity; for example, if a flame recombiner is used, propagation of flame into the containment should be prevented.

For most reactor plants, operation of the hydrogen control system would not be required for time periods of the order of seven days or more following a postulated design basis LOCA. Thus, it is reasonable that hydrogen control systems need not necessarily be installed at each reactor. Provision for either onsite or offsite storage or a shared arrangement between licensees of plants in close proximity to each other may be developed. An example of an acceptable arrangement would be to provide at least one hydrogen control system per site with the provision that a redundant unit would be available from a nearby site.

B. BRANCH TECHNICAL POSITION

1. All water-cooled power reactor facilities should have the capability for measurement of the hydrogen concentration, for mixing the atmosphere in the containment, and for controlling combustible gas concentrations without reliance on purging of the containment atmosphere following a loss-of-coolant accident.

2. The continuous presence of combustible gas control equipment at the site may not be necessary provided it is available on an appropriate time scale; however, appropriate design and procedural provisions should be made for its use. In addition, centralized storage facilities that would serve multiple sites may be used provided that these facilities include provisions such as maintenance, protective features, testing, and transportation for redundant units to a particular site.

3. Combustible gas control systems and the provisions for mixing, measuring, and sampling should meet the design, quality assurance, redundancy, energy source, and instrumentation requirements for an engineered safety feature, and the system itself should not introduce safety problems that may affect containment integrity. The combustible gas control system should be designated seismic Category I (See Regulatory Guide 1.29), and the Group B quality standards of Regulatory Guide 1.26 should be applied.

4. All water-cooled power reactors should also have the installed capability for a controlled purge of the containment atmosphere. The purge system need not be redundant nor be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary. Filtration of the purge stream should be provided as necessary to reduce the sum of the long-term doses from the LOCA and the purge to values less than the guidelines of 10 CFR Part 100 at the low population zone outer boundary.

5. The parameter values listed in Table 1 should be used for the purpose of calculating hydrogen and oxygen gas concentrations in containments and evaluating designs provided to control and to purge combustible gases evolved in the course of loss-of-coolant accidents. These values may be changed on the basis of additional experimental evidence and analyses.

6. Materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practical.

7. For plants for which a notice of hearing on the application for a construction permit was published after November 5, 1970:

   a. Plants recieving operating licenses on the basis (in part) of ECCS evaluations under §50.46 should conform to items 1-6, above, prior to operation.

   b. Plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, should conform, prior to operation, to items 1-6, above, but with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis.

      Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

   c. Designs of plants receiving construction permits on the basis (in part) of ECCS evaluations under §50.46 should include combustible gas control measures in conformance with items 1-6, above.

d. Designs of plants receiving construction permits on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, should include combustible gas control measures that conform, at the option of applicants, to one of the following:

(1) Items 1-6, above, based on a commitment to a specified maximum metal-water reaction to be calculated according to §50.46.

(2) Items 1-6, above, but with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis.

8. For plants for which a notice of hearing on the application for a construction permit was published between December 22, 1968 and November 5, 1970:

a. A redundant combustible gas control system (such as a recombiner system) as described in items 1 and 2, above, or a repressurization system[1/] designed with redundant elements and designated seismic Category I should be provided unless purging doses are less than the limits given in subparagraph (b), below. Purging capability should also be provided as a backup measure to a combustible gas control system, but in this case no purging dose computations need be submitted and the purging system need not have redundant elements or be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary.

b. If the incremental long-term doses from purging in the event of a postulated LOCA are calculated to be less than 2.5 rem whole body and 30 rem thyroid at all points beyond the exclusion area boundary, no combustible gas control systems other than the purging system need be provided. The combination of the dose from the purge and the long-term dose from a postulated LOCA should be below the guidelines of 10 CFR Part 100 at the low population zone outer boundary. Any filtration system for which credit is taken in calculating the purging dose should be redundant, should be designated seismic Category I, and the Group B quality standards of Regulatory Guide 1.26 should be applied. Such filtration systems should be designed, constructed, and tested to meet the recommendations of Regulatory Guide 1.52 to the extent practical. The purging system should be designed to that it is not made inoperative by the failure of any single active component (such as a valve, blower, or electrical power source).

[1/] Provisions such as a containment atmospheric dilution system that introduces additional gas into the drywell of some BWR plants may be provided to delay the time to purge on plants in this category; however, the containment should not be repressurized beyond 50% of the containment design pressure.

c. For plants receiving operating licenses on the basis (in part) of ECCS evaluations under §50.46, the parameter values listed in Table 1 should be used to calculate combustible gas concentrations in containments and to evaluate designs provided to control and to purge these gases.

For operating plants, or plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, the parameter values of Table 1 should be similarly used, with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis. Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

d. Combustible gas control systems conforming to this section (B.8) should be provided prior to operation or as soon thereafter as practical.

9. For plants for which a notice of hearing on the application for a construction permit was published before December 22, 1968:

a. Information regarding the calculated dose from purging should be furnished to the staff. If the sum of the long-term doses from a postulated LOCA and the purging dose is below the guidelines of 10 CFR Part 100 at the low population zone outer boundary, no combustible gas control systems other than the purging system need be provided.

b. Any filtration system for which credit is taken in calculating the purging dose should be redundant and designated seismic Category I, and the Group B quality standards of Regulatory Guide 1.26 should be applied. Such filtration systems should be designed, constructed, and tested to meet the recommendations of Regulatory Guide 1.52 to the extent practical.

c. The purging system should be designed so that it is not made inoperative by the failure of any single active component (such as a valve, blower, or electrical power source).

d. If the long-term dose limit of subparagraph (a) cannot be met by a purging system with filtration, either a redundant combustible gas control system (such as a recombiner system) as described in items 1 and 2, above, or a repressurization system[1] with redundant elements and designated seismic Category I should be provided. Purging capability should also be provided as a backup measure for the combustible gas control system, but the purging system need not have redundant filters, be designated seismic Category I, except insofar as portions of the system constitute part of the primary containment boundary, or meet the single failure or long-term dose limit criteria, above.

[1] Ibid., page 21.

e. For plants receiving operating licenses on the basis (in part) of ECCS evaluations under §50.46, the parameter values listed in Table 1 should be used to calculate combustible gas concentrations in containments and to evaluate designs provided to control and to purge these gases.

For operating plants, or plants receiving operating licenses on the basis (in part) of ECCS evaluations under the Interim Policy Statement of June 29, 1971, the parameter values of Table 1 should be similarly used, with item 4 of Table 1 changed to specify a five percent metal-water reaction and an evolution time determined on an individual case basis. Reevaluations of combustible gas control measures for plants in this category to take account of the change in amount of assumed metal-water reaction may be made at the option of applicants and licensees after submission of §50.46 ECCS analyses and final approval by the staff.

f. Schedules for installation of purging systems or other combustible gas control systems should be considered on an individual case basis.

C. REFERENCES

The references for this branch technical position are the same as those for Standard Review Plan 6.2.5, given in Section V of the plan.

TABLE 1

1.	Fraction of fission product radiation energy absorbed by the coolant[1/]	a. Beta (1) Betas from fission products in the fuel rods: 0 (2) Betas from fission products intimately mixed with coolant: 1.0 b. Gamma (1) Gammas from fission products in the fuel rods, coolant in core region: 0.1[2/] (2) Gammas from fission products intimately mixed with coolant, all coolant: 1.0
2.	$G(H_2)$[1/]	0.5 molecules/100 ev
3.	$G(O_2)$[1/]	0.25 molecules/100 ev
4.	Extent and evolution time of initial core metal-water reaction hydrogen production from the surrounding fuel.	Hydrogen production is 5 times the amount from the maximum calculated reaction under 10 CFR §50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.00023 inches, whichever is greater, in 2 minutes.
5.	Aluminum corrosion rate for aluminum exposed to alkaline solutions.	200 mils/yr (This value should be adjusted upward for higher temperatures early in the accident sequence)
6.	Fission product distribution model.	a. 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water. b. All noble gases are released to the containment. c. All other fission products remain in fuel rods.
7.	a. Hydrogen concentration limit. b. Oxygen concentration limit.	4 volume percent[3/] 5 volume percent (This limit should not be exceeded if more than 6 v/o hydrogen is present.)

[1/]For water, borated water, and borated alkaline solutions; for other solutions, data should be presented.

[2/]This fraction is thought to be conservative; further analysis may show that it should be revised.

[3/]The 4 v/o hydrogen concentration limit should not be exceeded if burning is to be avoided and more than 5 v/o oxygen is present in containment.

This amount may be increased to 6 v/o, with the assumption that the 2 v/o excess hydrogen would burn in the containment (if more than 5 v/o oxygen is present). The effects of the resultant energy and burning should not create conditions exceeding the design conditions of either the containment or safety equipment necessary to mitigate consequences of a LOCA. Applicants and licensees should demonstrate such capability by suitable analyses and qualification test results.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.2.6 CONTAINMENT LEAKAGE TESTING

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

Information describing the reactor containment leakage testing program is reviewed by the CSB. At the construction permit (CP) stage the preliminary safety analysis report (PSAR) will not usually contain a description of the program in detail but will contain commitments by the applicant to develop a program which will meet the intent of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The CSB review of the reactor containment leakage testing program at the operating license (OL) stage covers the following specific areas:

1. Containment integrated leakage rate tests (Type A tests as defined by Appendix J), including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria, and scheduling of tests.

2. Containment penetration leakage rate tests (Type B tests as defined by Appendix J), including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests.

3. Containment isolation valve leakage rate tests (Type C tests as defined by Appendix J), including identification of isolation valves, general test methods, test pressures, acceptance criteria, and scheduling of tests.

4. Technical specifications pertaining to containment leakage rate testing.

The Accident Analysis Branch analyzes the radiological consequences of loss-of-coolant accidents using the containment design leakage rate. The containment leakage testing program must verify that the containment leakage rate is less than the design limit.

In addition to the tests described above, CSB reviews the special leakage testing programs needed for subatmospheric-type containments, the secondary containments for plants using the dual containment concept and the Mark III drywell-containment system. For subatmospheric containments, the leakage into the reactor containment must be monitored and limited such that the reactor containment pressure can be maintained below atmospheric for the duration of the post-accident period. A testing program should be described in the safety analysis report (SAR) and its adequacy is reviewed by CSB. Dual containments are proposed for some plants because of site limitations. The intent of the dual containment is to collect and process reactor containment leakage. Testing programs to ensure that leakage will be contained as proposed by applicants using this kind of containment are reviewed by CSB. CSB will also review the testing program for the Mark III type containment with regard to the integrity of the drywell and drywell bypass leakage paths.

II. ACCEPTANCE CRITERIA

The reactor containment leakage rate testing program, as described in the SAR, will be acceptable if it meets the requirements stated in Appendix J to 10 CFR Part 50. Appendix J provides the test requirements and acceptance criteria for preoperational and periodic leak testing of the reactor containment, and of systems and components which penetrate the containment. Appendix J also references ANSI N45.4-1972, which identifies acceptable methods for determining the leakage rate of containment structures. Exceptions to Appendix J requirements will be reviewed on a case-by-case basis.

The minimum acceptable design containment leakage rate shall not be less than 0.1% per day.

The inleakage rate to the reactor containment of subatmospheric containments will be acceptable if it is less than the inleakage rate used in the analysis of the containment response to loss-of-coolant accidents. Systems should be provided to measure the inleakage rate at periodic intervals.

The leakage limits of the secondary containments of dual-type containments are acceptable if they are based on the limits used in the analysis of the secondary containment depressurization time. These tests should be conducted at each refueling or at intervals not exceeding 18 months. Type B and C tests will be done on those penetrations and isolation valves that communicate directly with the atmosphere, where there is a possibility of uncollected leakage. The test limits should be consistent with the limits used for direct leakage in the analysis of the radiological consequences by AAB.

The leakage limit for Mark III drywells should be such that the measured leakage does not exceed 10% of the drywell bypass capability for small breaks. In terms of the leakage path area, this corresponds to a value of the parameter $A/\sqrt{k}$ of the order of one square foot.

III. REVIEW PROCEDURES

In the review of the PSAR, CSB confirms that the design leakage is stated and that the reactor containment leakage testing program, to be detailed at the OL stage, will be consistent with the requirements of Appendix J. In the review of the final safety analysis report (FSAR) at the OL stage, CSB reviews the reactor containment leakage testing program and the applicant's proposed technical specifications for completeness and for conformance to Appendix J.

The review of the reactor containment leakage rate test program at the OL stage includes the following:

1. Containment Integrated Leakage Rate Test (Type A Test)

   a. Pretest requirements are reviewed to ensure that a general inspection for containment structural deterioration is included and procedures for corrective action are available. The reviewer confirms that an inspection of the accessible interior and exterior surfaces of the containment structures and components will be performed prior to any Type A tests. The purpose of this inspection is to identify any structural deterioration which may affect the containment structural integrity or leak tightness. The reviewer confirms that procedures for corrective action, if necessary, are specified in the test program. The reviewer also confirms that the test program includes a provision that, in the event that structural deterioration is discovered, a Type A test will not be performed until corrective action is taken in accordance with procedures specified in the test program. In addition, these corrective actions will be reported to the staff in the test report.

   The program should require that the containment isolation valves be closed by normal operating procedures, with no accompanying adjustments.

   The reviewer should confirm that the test program includes stabilization of containment conditions (temperature, pressure, humidity) for a period of at least four hours as a pretest requirement.

   The pretest requirements should identify those portions of fluid systems which will be opened or vented to atmosphere and drained of fluids, to assure that isolation valves are exposed to the containment test air pressure.

   Those systems not vented or drained should be identified and the reason for not venting or draining should be stated. Piping and instrumentation diagrams and process flow drawings are used by the reviewer to confirm that in the vented and drained condition, the isolation valves of those portions of fluid systems that are part of the reactor coolant pressure boundary and are opened directly to the containment atmosphere during a LOCA, are exposed to the test air pressure. Those systems required to maintain the plant in a safe condition or normally filled with water and needed post-LOCA (i.e., heat removal systems) need not be vented. By reference to the drawings, the reviewer assures himself that leakage to the environment cannot occur for those systems not vented and drained.

   b. Test methods described in the program are reviewed to assure that they are consistent with the methods stated in ANSI N45.4-1972. The accuracy of the Type A leak test must be confirmed by a supplemental test. The supplemental test is prescribed by Appendix J.

The proposed supplemental test is acceptable if the difference in leakage rate between the Type A test and the supplemental test is specified to be within 25 percent of the maximum allowable leakage rate of the Type A test.

c. Acceptance criteria for preoperational and periodic leakage rate tests should be included in the test program and in the technical specifications. The reviewer confirms that the acceptable measured containment leakage rate will not exceed 75% of the maximum allowable leakage rate during either preoperational or periodic leakage rate tests.

d. Provisions for additional testing in the event of failure to meet acceptance criteria should be stated in the program. The reviewer assures that the test program specifies that if two consecutive Type A tests fail the acceptance criteria, a Type A test shall be performed at each refueling shutdown or every 18 months until two Type A tests meet the acceptance criteria. Also, it should be stated that if any periodic Type A test fails the acceptance criteria, the test schedule for subsequent Type A tests will be submitted to the staff for review.

2. <u>Containment Penetration Leakage Rate Test (Type B Test)</u>

a. All containment penetrations should be listed in the test program. By reference to piping and instrumentation diagrams, the reviewer confirms that all penetrations have been listed. The program should identify any penetration not requiring leakage testing and the reason for not requiring a test should be stated. The reviewer confirms that those penetrations not requiring testing cannot result in leakage to the atmosphere during normal operation or a LOCA. An example of such penetrations is a seal-welded equipment hatch.

b. Test methods for determining penetration leakage rates are accepted by the reviewer if they include any of the following methods: examination by the halide leak detection method of a pressurized test chamber constructed as part of the penetration; measurement of the rate of pressure loss of the pressurized test chamber of the penetration; and leakage surveillance by means of a permanently installed system for continuous or intermittent pressurization of individual or groups of penetrations, and measurement of pressure loss.

c. Test pressures for containment penetrations should be stated in the test program and in the technical specifications. The test pressure is acceptable if it is the maximum calculated containment accident pressure.

d. Acceptance criteria for penetration leakage rate testing should be included in the test program and the technical specifications. The reviewer confirms that the combined leakage rate of all penetrations and valves subject to Type B and

Type C tests is specified in the SAR to be less than 60 percent of the maximum allowable containment leakage rate. Leakage measurements obtained through leakage surveillance systems that maintain a pressure not less than the calculated containment accident pressure at penetrations during normal reactor operation are acceptable in lieu of Type B tests.

3. Containment Isolation Valve Leakage Rate Test (Type C Test)

   a. All containment isolation valves requiring a Type C test should be listed in the test program. By reference to the piping and instrumentation diagrams, the reviewer confirms that all isolation valves to be tested have been listed. The basis for determination that an isolation valve requires Type C testing is: a direct connection is provided between the inside and outside of the reactor containment and the valve forms a part of the containment boundary; the valve is required to close automatically upon receipt of a containment isolation signal; the valve is required to operate intermittently under post-accident conditions; or the valve is in main steam or feedwater piping or other systems which penetrate containment of boiling water reactors.

   b. Test methods for isolation valve Type C tests should be included in the test program. The method of testing is acceptable if the following is stated in the test program: Type C test pressure shall be applied in the same direction as that existing when the valve is required to perform its safety function; each valve to be tested will be closed by normal operating procedures with no prelimiary adjustment or exercising; and test methods similar to those methods used for leak testing containment penetrations will be used.

   c. Test pressures for isolation valve Type C tests should be included in the test program and technical specifications. The reviewer should confirm that the test pressure specified is the maximum calculated containment accident pressure and that the test pressure for valves sealed with fluid from a seal system is 110% of the maximum calculated accident pressure.

   d. Acceptance criteria for isolation valve leak testing should be included in the test program and technical specifications. The reviewer confirms that the combined leakage rate of all penetrations and valves subject to Type B and C tests is specified in the SAR to be less than 60 percent of the maximum allowable containment leakage rate.

4. The scheduling and reporting of periodic tests should be included in the test program and technical specifications. The reviewer accepts test schedules if they are in accordance with the following:

   a. Type A periodic tests should be performed at three equal intervals during each ten-year service period. The third test should be scheduled for the ten-year shutdown inspection.

b. Type B periodic tests should be performed during each shutdown for refueling but no longer than two years (except air locks). Air locks should be tested at six month intervals or after each opening.

   Penetrations using continuous monitoring should be tested with every other reactor shutdown for refueling but no longer than three years (except air locks).

c. Type C periodic tests should be performed during each reactor shutdown for refueling but no longer than two years.

   Test reports should be discussed in the test program and the reviewer should confirm that it is stated in the program that preoperational and periodic tests shall be the subject of a summary report submitted to the Commission approximately three months after each test. It should be stated in the program that the following will be included in the preoperational test report: schematic of leak measuring system; instrumentation used; supplemental test method; test program; and analysis and interpretation of leakage rate test data for the Type A test.

   It should be stated that test results that fail to meet acceptance criteria will be reported in a separate summary report, including analysis and interpretation of test data.

5. Special testing requirements should be included in the test program. Special testing procedures for subatmospheric and dual-type containments should be identified.

   CSB assures that the applicant has provided a leakage testing program and has specified the maximum leakage which may occur from inleakage for subatmospheric type containments and bypass (or dilution) leakage for dual-type containments. Potential leakage paths which bypass the annulus or the auxiliary building areas or may leak directly to atmosphere must be identified. The total amount of containment bypass leakage to the environment must be specified and included in the technical specifications. The reviewer determines that the test provisions are adequate to confirm the bypass leakage specified.

6. Preoperational and post-operation test reports are not reviewed by CSB on a routine basis. Audits of tests may be conducted at the discretion of CSB or special reviews may be conducted at the request of other staff organizations.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR Part 50. Such compliance provides assurance that containment leak-tight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

"Maintaining containment leakage rates within such limits provides assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of General Design Criteria 52, 53, and 54 of Appendix A to 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

2. 10 CFR Part 50, Appendix A, General Design Criterion 52, "Capability for Containment Leakage Rate Testing."

3. 10 CFR Part 50, Appendix A, General Design Criterion 53, "Provisions for Containment Testing and Inspection."

4. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Systems Penetrating Containment."

5. ANSI N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors," American National Standards Institute (1972).

6. A. K. Postma and B. M. Johnson, "Containment System Experiment Final Program Summary," BNWL-1592, Battelle Pacific Northwest Laboratories, July 1971.

11/24/75

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.3 EMERGENCY CORE COOLING SYSTEM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The RSB reviews the information presented in the applicant's safety analysis report (SAR) regarding the emergency core cooling system (ECCS). The major elements of the review are:

1. Design Bases

   The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the general design criteria and the amendments to 10 CFR 50 regarding ECCS acceptance criteria issued by the Commission on December 28, 1973 (Ref. 1).

2. Design

   The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

3. Test Program

   The preoperational and initial startup test programs for the ECCS are reviewed to determine if they are sufficient to confirm the performance capability of the ECCS. The need for special design features to permit the performance of adequate test programs is also evaluated as a part of this review.

4. Technical Specifications

   The proposed technical specifications are reviewed to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The ECCS is also reviewed to assure that it has the proper seismic and quality group classifications. This aspect of the review is performed as a part of the effort described in

Standard Review Plans (SRP) 3.2.1 and 3.2.2. The ECCS system is to be enclosed in structures having the proper seismic classification. This review is done as described in SRP 3.2.2. The structures are reviewed by SEB.

APCSB reviews, as described in SRP 9.2.1, 9.2.2, 9.2.5 and 9.2.6, those auxiliary systems essential for ECCS operation (service water system, component cooling system, ultimate heat sink, and condensate storage facility) and assesses the capability of these systems to perform all functions required by the ECCS. The APCSB will supply, on request, evaluations of portions of the power conversion systems (e.g., steam supply lines, steam generators, feedwater systems) which interface with the reactor coolant system in such as way as to influence the course of a loss-of-coolant accident (LOCA) for a particular plant.

The EICSB, as described in SRP 7.3 and 8.3.1, reviews the adequacy of ECCS-associated controls and instrumentation with regard to features of automatic actuation, remote sensing and indication, remote control, and emergency onsite power.

The MEB, as described in SRP 3.9.3, reviews the loading combinations (operational, LOCA, and seismic) and the associated stress limits. On a generic basis, the MTEB reviews the thermal shock effect of water injected into the primary coolant system from the ECCS. The CSB, as described in SRP 6.2.1.3, reviews the analyses of mass and energy released to the containment following a LOCA to determine acceptability of the containment backpressure used in the ECCS capability studies.

The MEB and APCSB review the effects of pipe breaks both inside and outside containment on ECCS. This review includes the effects of pipe whip, jet impingement forces, and any environmental conditions created.

The ability of the ECCS to mitigate the consequences of a spectrum of loss-of-coolant accidents is reviewed by RSB under SRP 15.6.5.

II. ACCEPTANCE CRITERIA

The general objective of the review of the ECCS is to determine that the system meets the applicable general design criteria (GDC), the ECCS acceptance criteria of 10 CFR §50.46, and the intent of applicable regulatory guides.

In regard to the ECCS acceptance criteria (Ref. 1), the five major performance criteria deal with:

1. Peak cladding temperature.
2. Maximum calculated cladding oxidation.
3. Maximum hydrogen generation.
4. Coolable core geometry.
5. Long-term cooling.

These areas are reviewed as a part of the effort associated with the LOCA analysis (SRP 15.6.5). However, the impact of various postulated single failures on the operability of the ECCS is evaluated under this plan.

The ECCS must meet the requirements of GDC 35 (Ref. 8). The system must have alternate sources of electric power, as required by GDC 17 (Ref. 5), and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident. Further, the ECCS should be designed to perform its function considering simultaneous LOCA and seismic loadings.

The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of GDC 36 (Ref. 9). The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by GDC 37 (Ref. 10).

The portions of the protection system associated with ECCS must meet the requirements of GDC 20 (Ref. 6) and GDC 27 (Ref. 7) and should conform to the recommendations of Regulatory Guide 1.47 (Ref. 17). The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.

The design of the ECCS must be in accord with GDC 2 (Ref. 2); GDC 4 (Ref. 3); and GDC 54 (Ref. 11); and should conform to the recommendations of Regulatory Guide 1.1 (Ref. 13), Regulatory Guide 1.29 (Ref. 15); Regulatory Guide 1.46 (Ref. 16); and staff positions on protection against piping failures outside containment (Ref. 21).

All ECCS and instrument lines that penetrate the primary reactor containment must be provided with suitable isolation valves in accordance with the requirements of GDC 56 (Ref. 12) and should meet the recommendations of Regulatory Guide 1.11 (Ref. 14).

Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to GDC 5 (Ref. 4).

The preoperational and initial startup test programs should meet the intent of Regulatory Guide 1.68 (Ref. 19) and Regulatory Guide 1.79 (Ref. 20).

III. <u>REVIEW PROCEDURES</u>

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan.

For operating license (OL) reviews, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes: (1) the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing; and (2) the preoperational and initial startup test programs, to assure that they meet the intent of Regulatory Guides 1.68 and 1.79 (Refs.19 and 20).

Much of the review described below is generic in nature and is not performed for each plant. That is, the RSB reviewer compares the ECCS design and parameters to those of previously reviewed plants and then devotes the major portion of the review effort to those areas where the application is not identical to previously reviewed plants. The following steps are taken by the RSB reviewer to determine that the acceptance criteria of Section II have been met. These steps should be adapted to CP or OL reviews as appropriate.

1. The relationship of the system under review to other previously approved plants is established. Systems or design features claimed to be identical or equivalent to those of previously approved plants are confirmed to be identical or equivalent.

2. Piping diagrams are reviewed to evaluate the functional reliability of the system in the event of single failures. That is, by referring to piping and instrumentation diagrams, the existence of the redundancy required by the criteria is confirmed.

3. The significant design parameters (e.g., pump net positive suction head, pump head vs. flow, accumulator volume and pressure, water storage volume, system flow rate and pressure, etc.) are examined for each component to confirm that these parameters satisfy operating requirements and the recommendations of Regulatory Guide 1.1 (Ref. 13).

4. The piping and instrumentation diagrams are checked to see that essential ECCS components are designated seismic Category I and Safety Class II (the cooling water side of heat exchangers can be Safety Class III).

5. The ECCS design is reviewed to confirm that the system can function in post-accident environments, considering possible mechanical effects, missiles, and the pressure, temperature, moisture, radioactivity, and chemical conditions resulting from LOCA. Protection against valve motor flooding should be confirmed by the RSB reviewer. In regard to possible mechanical effects and missiles, the RSB reviewer confirms that appropriate reviews have been made as described in other review plans, and that the ECCS has been found, in these reviews, to meet the requirements of GDC 2 and 4, the recommendations of Regulatory Guides 1.29 and 1.46, and staff positions on pipe breaks outside containment (Refs. 2, 3, 15, 16 and 21). Regarding the effects of pressure, temperature, etc., the RSB reviewer should confirm that, prior to installation, representative active components used in the ECCS will be proof-tested under environmental conditions and for time periods representative of the most severe operating conditions to which they may be subjected.

6. The criteria, supporting analyses, plant design provisions, and operator actions that will be taken are reviewed to ensure that there will not be unacceptably high concentrations of boric acid in the core region (resulting in precipitation of a solid phase) during the long-term cooling phase following a postulated LOCA.

7. The ECCS design is reviewed to confirm that there are provisions for maintenance of the long-term coolant recirculation and decay heat removal systems, e.g., pump or valve overhaul, in the post-LOCA environment (including consideration of radioactivity).

8. The availability of an adequate source of water for the ECCS is confirmed, and the source volume, location, and susceptibility to failure (e.g., freezing) are evaluated. (RSB will request APCSB review as required.) In PWR's, the piping from the water source to the ECCS safety injection pumps are evaluated for conformance with RSB 6-1 (Ref. 22).

9. The ECCS flow paths are reviewed to determine the extent to which flow from the ECCS pumps is diverted as a backup feature to other safeguards equipment (e.g., RHR, containment spray). The reviewer should confirm that the remaining portion of the flow provides abundant core cooling, despite the most severe single failure that affects ECCS flow.

10. For a boiling water reactor (BWR), the reactor coolant automatic depressurization system is reviewed to confirm the capability to satisfy LOCA pressure relief functions, including consideration of a single failure.

11. The design of ECCS injection lines is reviewed to confirm that the isolation provisions at the interface with the reactor coolant system are adequate. The number and type of valves used to form the interface between low pressure portions of the ECCS and the reactor coolant system must provide adequate assurance that the ECCS will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following provisions:

    a. One or more check valves in series with a normally closed motor-operated valve. The motor-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

    b. Three check valves in series.

    c. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually.

12. ECCS piping and instrument lines that penetrate containment are reviewed to confirm that there are appropriate containment isolation measures, in accordance with Regulatory Guide 1.11 (Ref. 14).

13. Motor-operated isolation valves in ECCS lines connecting the accumulators to the reactor coolant system in a pressurized water reactor (PWR) are reviewed to ensure that adequate provisions are made against inadvertent isolation.

14. The capacity and settings of relief valves provided for the ECCS to satisfy system overpressure protection requirements are reviewed. In particular, for PWR's, the reviewer confirms that the accumulator relief valves have adequate capacity so that leakage from the reactor coolant system will not jeopardize the integrity of the accumulators.

15. The reviewer confirms that the design has provisions to assure that ECCS injection lines are maintained in a filled condition, to preclude the possibility of a water hammer when injection flow is initiated.

16. The reviewer confirms that no component or feature of the ECCS in one reactor facility on a multiple plant site is shared with the ECCS in another facility, or that shared features clearly meet the requirements of GDC 5 (Ref. 4).

17. The reviewer confirms that within an individual reactor facility, any components shared between the ECCS and other systems (e.g., coolant makeup systems, residual heat removal systems, containment cooling systems) satisfy engineered safeguard feature design requirements and that the ECCS function of the shared component is not diminished by the sharing.

18. The reviewer confirms that ECCS components located exterior to the reactor containment are housed in a structure which, in the event of leakage from the ECCS, permits venting of releases through iodine filters designed in accordance with Regulatory Guide 1.52 (Ref. 18).

19. The complete sequence of ECCS operation from accident occurrence through long-term core cooling is examined to see that a minimum of manual action is required, and where manual action is used, a sufficient time (greater than 20 minutes) is available for the operator to respond.

20. The reviewer confirms that long-term cooling capacity is adequate in the event of failure of any single active or passive component of the ECCS. If an intermediate heat transport system, such as the component cooling water system, is used to provide long-term cooling capability, the system must be designed and constructed to an appropriate group classification, must be seismic Category I, and must be capable of sustaining a single active or passive failure without loss of function. Intermediate systems that conform to the staff positions on piping failures outside containment (Ref. 21) are considered to meet these requirements. (RSB will request APCSB review as required.)

21. The RSB reviewer consults with the EICSB reviewer to:

a. Confirm that the power requirements of the ECCS, including the timing of electrical loads, are compatible with the design of onsite emergency power systems, both a-c and d-c.

b. Confirm that there are sufficient instrumentation and controls available to the reactor operator to provide adequate information in the control room to assist in assessing post-LOCA conditions, including the more significant parameters such as coolant flow, coolant temperature, and containment pressure. If ECCS flow is diverted as a backup to other safeguards systems, the reviewer confirms that instrumentation and controls are available to provide sufficient information in the control room to determine that adequate core cooling is being provided.

c. Confirm that automatic actuation and remote-manual valve controls are capable of performing the functions required, that suitable interlocks are provided, which do not impair separation of power trains or inhibit the required valve motions, and that instrumentation and controls have sufficient redundancy to satisfy the single failure criterion.

22. Analyses are provided by the applicant in Chapter 15 of the SAR to assess the capability of the ECCS to meet functional requirements. These analyses are reviewed by the RSB, as described in SRP 15.6.5, to determine conformance to the acceptance criteria for ECCS. However, the following portions of the review of ECCS response in loss-of-coolant accidents are performed by the RSB reviewer under this plan:

a. The lower limit of break size for which ECCS operation is required is established; i.e., the maximum break size for which normal reactor coolant makeup systems can maintain reactor pressure and coolant level is determined. The capability of the ECCS to actuate and perform at this lower limit of break size is confirmed.

b. The reviewer confirms that the analyses take into account a variety of potential locations for postulated pipe breaks, including ECCS injection lines.

c. The reviewer confirms that the analyses take into account a variety of single active failures. The reviewer should keep in mind that different single failures may be limiting, depending on the particular break location and break size postulated.

d. The ECCS component response times (e.g., for valves, pumps, power supply) are reviewed to confirm that they are within the delay times used in the accident analyses.

e. The ECCS design adequacy for all modes of reactor operation (e.g., full power, low power, hot standby, cold shutdown, partial loop isolation) is confirmed.

23. The proposed plant technical specifications are reviewed to:

a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when ECCS equipment is inoperable due to repairs and maintenance. The means of indicating that safety systems have been bypassed or are inoperable should be in accordance with Regulatory Guide 1.47 (Ref. 17).

b. Confirm that the limiting conditions of operation ensure that the specified operating parameters (minimum poison concentrations, minimum coolant reserve in storage, etc.) are within the bounds of the analyzed conditions.

c. Verify that the frequency and scope of periodic surveillance testing is adequate.

24. The reviewer confirms that the design provides the capability for periodically demonstrating that the system will operate properly when an accident signal is received. That is, it should be demonstrated by an applicant that pumps and valves operate on normal and emergency power and that water pressure and flow are as designed when the plant is operating (periodic system surveillance). When the plant is shut down for refueling, the system should be tested for delivery of coolant to the vessel.

25. The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guides 1.68 and 1.79 (Refs. 19 and 20). At the OL stage, this aspect of the review is to assure that sufficient information is provided by the applicant to identify the test objectives, methods of testing, and test acceptance criteria (see par. C.2.b of Regulatory Guide 1.68).

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the ECCS will perform its safety function. As an alternative to this detailed evaluation, the reviewer may compare the ECCS design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same, the reviewer may conclude that the proposed test programs are adequate for the ECCS. If the proposed ECCS differs significantly from that of previously reviewed designs, the impact of the proposed changes on the required preoperational and initial startup testing programs are reviewed at the CP stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

26. Information is provided to other branches in those areas where the RSB has a secondary review responsibility that is not explicitly covered by steps 1-25, above. These additional areas of RSB secondary review responsibility include:

a. Confirmation that the LOCA forcing functions (blowdown loads) used to conduct the system dynamic analysis are representative of the most adverse LOCA loadings. This information is used by MEB as described in SRP 3.9.3.

b. Determining the acceptability of the mass and energy release rates to the containment following a LOCA for use in the minimum containment backpressure analyses performed by CSB, as described in SRP 6.2.1.3. CSB will then inform RSB of the acceptability of the containment backpressure used in the ECCS performance capability studies.

c. Identification (to APCSB) of essential auxiliary systems and components associated with the ECCS that are required for accident conditions, and accident cooling load functional requirements and minimum time intervals.

d. Identification (to APCSB) of process sampling system functional performance requirements for the reactor coolant system during and subsequent to postulated accident conditions.

e. Identification (to APCSB) of essential components associated with the main steam and auxiliary feedwater systems that are required to operate during and following accident conditions.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report. (For completeness, this evaluation finding includes the RSB review effort described in SRP 15.6.5.)

"The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transfer heat from the core following a loss-of-coolant accident. The scope of review of the ECCS for the ___________ plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

"The drawings, component descriptions, design criteria, and supporting analyses have been reviewed and have been found to conform to Commission regulations as set forth in the general design criteria, and to applicable regulatory guides and staff technical positions. The system was found capable of performing its function with only onsite electric power or with only offsite electric power, assuming the most restrictive single active failure.

"The applicant provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR §50.46. The five major performance criteria deal with:

(1) Peak cladding temperature.
(2) Maximum calculated cladding oxidation.
(3) Maximum hydrogen generation.
(4) Coolable core geometry.
(5) Long-term cooling.

"Based on our review of the applicant's analysis, we conclude that the proposed ECCS satisfies the acceptance criteria.

"The staff concludes that the design of the emergency core cooling system conforms to the Commission's regulations and to applicable regulatory guides and staff technical positions, and is acceptable."

V. REFERENCES

1. 10 CFR §50.46,"Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," issued by the Commission December 28, 1973; Federal Register, Vol. 39, No. 3, January 4, 1974.

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

5. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."

6. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

7. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control System Capability."

8. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

9. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System."

10. 10 CFR Part 50, Appendix A, General Design Criterion 37, "Testing of Emergency Core Cooling System."

11. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

12. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

13. Regulatory Guide 1.1, "Net Position Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

14. Regulatory Guide 1.11, "Instrument Lines Penetrating Containment."

15. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

16. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment."

17. Regulatory Guide 1.47, "Bypass and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

18. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

19. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

20. Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

21. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid Systems Piping Outside Containment," attached to SRP 3.6.2.

22. Branch Technical Position RSB 6-1, "Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps," attached to SRP 6.3.

BRANCH TECHNICAL POSITION RSB 6-1

PIPING FROM THE RWST (OR BWST) AND CONTAINMENT SUMP(S)
TO THE SAFETY INJECTION PUMPS

A. Background

Current PWR's utilize the refueling water storage tank (RWST) or the borated water storage tank (BWST) as the sole source of water for the safety injection pumps during the first twenty to forty minutes of any accident that trips a safety injection signal. Since acceptable results of safety analyses of the accidents are based on the operation of a minimum number of these pumps, interruption of this water supply for even a short period of time could result in unacceptably high fuel and cladding temperatures if the safety injection pumps fail because of cavitation or over heating.

General Design Criteria 35 requires that the emergency core cooling system have suitable redundancy in components and features and suitable interconnections to assure the system safety function can be accomplished assuming a single failure. The principal problem appears to be a definition of single failure. A recent draft of ANSI N658, "Single Failure Criteria for PWR Fluid Systems" defines an active failure as:

(a) "An active failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function upon demand."

(b) "Spurious action of a powered component originating within its actuation system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious action."

This branch position on the availability of the RWST is based on the above criteria and the recognition that water supplied from the RWST to the ECCS system is absolutely essential in the event of a LOCA.

B. Branch Position

1. The single active failure criterion defined in (a) and (b) above will be applied in evaluating the design of the piping systems that connect the safety injection pumps to the RWST (BWST) and the containment sumps.

2. The piping systems, including valves, shall be designed to satisfy the requirements listed below without the need to disconnect the power to any valve.

3. The valves and piping between the RWST (or BWST) and the safety injection pumps must be arranged so that no single failure will prevent the minimum flow to the core required to satisfy 10 CFR 50.46.

4. The valves and piping between the RWST (or BWST) and safety injection pumps must be arranged so that no single active failure will result in damage to pumps such that the minimum flow requirements for long-term core and containment cooling after a LOCA are not satisfied.

5. The valves and piping that connect the RWST (or BWST) and the containment sump(s) to the safety injection pumps must be arranged so as not to preclude automatic switchover from the injection mode of ECCS operation to recirculation cooling from the sump. These piping systems must be arranged so that the differential pressure between the sump and the RWST (or BWST), even if there is a single active failure, will not result in a loss of core cooling or a path that permits release of radioactive material from the containment to the environment.

C. Implementation

1. CP's Under Review and Future CP Reviews

The proposed position will be applied to all CP reviews for which an SER was not published prior to April 16, 1975. It is expected that all of the elements of the proposed position will be applied for such reviews. Taking this position on CP's would eliminate the need for various schemes such as locking out power to valves located in the line between the various ECCS pumps and refueling water storage tank.

2. OL's Under Review

For operating licenses that are presently under review and OL's to be reviewed in the future that are not covered by item 1, the proposed position will not be completely applied. Specifically, locking out power to valves will be permitted. For most plants it is expected that this will be sufficient to meet the single failure criteria. However, in other plants changes to the piping and valving arrangements may be required to satisfy the single failure criteria.

3. Plants Under Construction

These plants will be handled as discussed in item C2. It is expected, however, that we will discuss the proposed position with each of the applicable PWR vendors. It will be obvious to the vendors which plants now under construction may have a problem. Then a generic review may be conducted for those plants that have a severe problem.

4. Operating Plants

All of the operating plants are being evaluated as an ongoing part of the current ECC review. The review should be conducted as discussed in item C2 to assure that these plants meet the essential parts of the proposed position.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.4 HABITABILITY SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Effluent Treatment Systems Branch (ETSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The control room ventilation system and control building layout and structures as described in the applicant's safety analysis report (SAR), are reviewed with the objective of assuring that plant operators are adequately protected against the effects of accidental releases of toxic or radioactive gases. A further objective is to assure that the control room can be maintained as the center from which emergency teams can safely operate in the case of a design basis radiological release. To assure that these objectives are accomplished the following items are reviewed:

1. The zone serviced by the control room emergency ventilation system is examined to ascertain that all critical areas requiring access in the event of an accident are included within the zone (control room, kitchen, sanitary facilities, etc.) and to assure that those areas not requiring access are generally excluded from the zone.

2. The capacity of the control room in terms of the number of people it can accommodate for an extended period of time is reviewed to confirm the adequacy of emergency food and medical supplies and self-contained breathing apparatus and to determine the length of time the control room can be isolated before CO_2 levels become excessive.

3. The control room ventilation system layout and functional design is reviewed and flow rates and filter efficiencies are determined for input into the AAB analyses of the buildup of radioactive or toxic gases inside the control room, assuming a design basis release. Basic deficiencies that might impair the effectiveness of the system are examined. In addition, the system operation and procedures are reviewed. The APCSB has primary responsibility in the system review area under Standard Review Plan (SRP) 9.4.1. The APCSB is consulted when reviewing hardware and operating procedures.

4. The flow rates and iodine removal efficiencies used in the analysis are obtained from the ETSB (see SRP 6.5.1).

5. The physical location of the control room with respect to potential release points of hazardous airborne materials (SAR chapter 2 and other pertinent chapters) is reviewed to determine the location and source strength of radioactive, toxic, or noxious materials. The layout of the control building is reviewed to assure that airborne materials will not enter the control room from corridors or ventilation ducts, etc. Estimates of dispersion of airborne contamination are made.

6. Radiation shielding provided by structural concrete is analyzed to determine the effectiveness of shielding and structure surrounding the control room. The control building layouts are checked to see if radiation streaming through doors (or other apertures) or from equipment might be a problem.

7. Independent analyses are performed to determine whether dose values or toxic gas concentrations remain below recommended levels. The SAB checks and concurs with the meteorological analysis used to obtain the X/Q values for the control room location.

II. ACCEPTANCE CRITERIA

1. Control Room Emergency Zone
   See Section III.1 of this plan.

2. Control Room Personnel Capacity
   Food, water, and medical supplies should be sufficient to maintain the emergency team (at least 5 men) for 5 days. Also see Section III.2 of this plan.

3. Ventilation System Criteria (See III.3 of this plan)
   Self-contained breathing apparatus for the emergency team (at least 5 men) should be on hand. A six-hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). Refer to References 3 thru 6, and see Section III.3 of this plan.

4. Emergency Standby Filters
   See Standard Review Plan 6.5.1 for acceptance criteria for control room ESF systems.

5. Relative Location of Source and Control Room
   In general, the control room inlets must be so placed in relation to the location of potential release points as to minimize control room contamination in the event of a release. Specific criteria as to radiation and toxic gas sources are as follows:

   a. Radiation Sources
      As a general rule the control room ventilation inlet should be separated from the major potential release points by at least 100 ft laterally and by 50 ft vertically. However, the actual minimum distances must be based on the dose analyses. Refer to Section III of this plan and Reference 7 for further information.

b. Toxic gases

The minimum separation distance is dependent upon the gas in question, the container size, and the available control room protection provisions. Refer to Regulatory Guide 1.78 (Ref. 3) for general guidance and to Regulatory Guide 1.95 (Ref. 4) for specific acceptable design provisions related to chlorine.

6. Radiation Shielding

See discussion of General Design Criterion 19 below.

7. Radioactive and Toxic Gas Hazards

a. Radiation Hazards

The dose guidelines (see General Design Criterion 19, Appendix A of 10 CFR Part 50) used in approving emergency zone radiation protection provisions are as follows:

(1) Whole body gamma: 5 rem

(2) Thyroid: 30 rem

(3) Beta skin dose: 30 rem*

The whole body gamma dose consists of contributions from airborne radioactivity inside and outside the control room, as well as direct shine from fission products inside the reactor containment building.

b. Toxic Gases

For acceptance purposes, three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure, and acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of medication other than the use of self-contained breathing apparatus.

*Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. The skin and thyroid dose levels are to be used only for judging the acceptability of the design provisions for protecting control room operators under postulated design basis accident conditions. They are not to be interpreted as acceptable emergency doses. The dose levels quoted here are derived for use in the controlled plant environment and should not be confused with the conservative dose computation assumptions used in evaluating exposures to the general public for the purposes of comparison with the guideline values of 10 CFR Part 100.

The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows:

(1) Long-term limit (1 hour or greater): use a limit assigned for occupational exposure (40-hour week).

(2) Short-term limit (2 min. to 1 hour): use a limit that will assure that the operator will not suffer incapacitating effects after a one-hour exposure.

(3) Protective action limit (2 min. or less): use a limit that will assure that the operator will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes.

The protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. (They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient.) As an example of appropriate limits, the following are the three levels for chlorine gas:

Long-term:	1 ppm by volume
Short-term:	4
Protective action:	15

(See Reference 3 for protective action levels for other toxic gases)

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed for other plants and whether items of special safety significance are involved.

1. Control Room Emergency Zone

The reviewer checks to see that the zone includes the following:

a. Instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file.

b. The computer room, if it is used as an integral part of the emergency response plan.

c. The shift supervisor's office.

d. The operators' wash room and the kitchen.

The emergency zone should be limited to those spaces requiring operator occupancy. Spaces such as battery rooms, cable-spreading rooms, or any other spaces not requiring continuous or frequent occupancy after a design basis accident (DBA) generally should be excluded from the emergency zone. Inclusion of these spaces may increase the probability of smoke or hazardous gases entering the emergency zone. They may also increase the possibility of infiltration into the emergency zone, thus decreasing the effectiveness of the ventilation system in excluding contamination. It is advantageous to have the emergency zone located on one floor, with the areas included in the zone being contiguous.

2. Control Room Personnel Capacity

The reviewer checks to see that emergency food and water are provided. Normally a five-day supply for five men would be sufficient for land-based plants. A medical kit is also helpful. Specific requirements for these items have not been formulated. The air inside a 100,000 cubic-foot control room would support five persons for at least six days. Thus, CO_2 buildup in an isolated emergency zone is not normally considered a limiting problem.

3. Ventilation System Layout and Functional Design

This area is a major portion of the review. The procedures are as follows:

a. The type of system proposed is determined. The following types of protection provisions are currently being employed for boiling water reactor (BWR) or pressurized water reactor (PWR) plants:

(1) Zone isolation, with the incoming air filtered and a positive pressure maintained by the ventilation system fans. This arrangement is often provided for BWR's having high stacks. Air flow rates are between 400 and 4000 cfm.

(2) Zone isolation, with filtered recirculated air. This arrangement is often provided for BWR's and PWR's with roof vents. Recirculation rates range from 2,000 to 30,000 cfm.

(3) Zone isolation, with filtered recirculated air, and with a positive pressure maintained in the zone. This arrangement is essentially the same as that in (2), with the addition of the positive pressure provision.

(4) Dual air inlets for the emergency zone. In this arrangement, two widely spaced inlets are located outboard (on opposite sides) of potential toxic and radioactive gas sources. The arrangement guarantees at least one inlet being free of contamination (except under extreme no-wind conditions). It can be used in all types of plants. Makeup air supplied from the contamination-free inlet provides a positive pressure in the emergency zone and thus minimizes infiltration.

(5) Bottled air supply for a limited time. In this arrangement, a flow rate of 400 to 600 cfm is provided from compressed air containers for about one hour, to prevent inleakage. It is used in systems having containments whose internal atmospheric pressure becomes negative within an hour after a DBA (subatmospheric containments).

b. The input parameters to the radiological dose model are determined (see Item 5). The parameters are emergency zone volume, filter efficiency, filtered makeup air flow rate, unfiltered inleakage (infiltration), and filtered recirculated air flow rate.

c. The ventilation system components and the system layout diagrams are examined. The responsible reviewer in the APCSB should be consulted if there are questions pertaining to the system design. He will determine if the system meets the single failure criterion as well as other safety requirements (see SRP 9.4.1). Damper failure and fan failure are especially important. The review should confirm that the failure of isolation dampers on the upstream side of fans will not result in too much unfiltered air entering the control room. The AAB dose analysis results are used to determine how much unfiltered air can be tolerated.

d. The following information may be used in evaluating the specific system types (see Reference 7 for further discussion):

(1) Zone isolation, with filtered incoming air and positive pressure. These systems may not be sufficiently effective in protecting against iodine. The staff allows an iodine protection factor (IPF), which is defined as the time-integrated concentration of iodine outside over the time-integrated concentration within the emergency zone of 20 to 100 for filters built, maintained, and operated according to Regulatory Guide 1.52 (an IPF of 100 requires deep bed filters). Such systems are likely to provide a sufficient reduction in iodine concentration only if the source is at some distance from the inlets. Thus, in most cases only plants with high stacks ($\sim$ 100 m) would meet Criterion 19 with this system. Normally the staff suggests that these systems be modified to allow isolation and operation with recirculated air since only minor ducting changes are necessary.

(2) Zone isolation, with filtered recirculated air. These systems have a greater potential for controlling iodine than those having once-through filters. IPF's ranging from 20 to over 150 can be achieved. These are the usual designs for plants having vents located at containment roof level. A system having a recirculation rate of 5000 cfm and a filter efficiency of 95% would be rated as follows:

Infiltration (cfm)	IPF*
200	25
100	49
50	96
25	191

*Within the range of interest, the iodine protection factor is directly proportional to recirculation flow rate times efficiency.

Infiltration should be determined conservatively. The calculated or measured gross leakage is used to determine the infiltration rate that will be applied in the evaluation of the radiological consequences of postulated accidents. This rate is determined as follows:

(i) The leakage from the control room when pressurized to 1/8-inch water gauge is calculated on the basis of the gross leakage data. One-half of this value is used to represent the base infiltration rate. Component leak rates may be used to calculate gross leakage (see, for example, References 8 and 9).

(ii) The base infiltration rate is augmented by adding to it the estimated contribution of opening and closing of doors associated with such activities as the required emergency procedures external to the control room. Normally 10 cfm is used for this additional contribution.

(iii) An additional factor that is used to modify the base infiltration rate is the enhancement of the infiltration occurring at the dampers or valves upstream of recirculation fans. When closed, these dampers typically are exposed to a several-inch water gauge pressure differential. This is accounted for by an additional infiltration contribution over the base infiltration at 1/8-inch water gauge.

The use of an infiltration rate that is based on calculation is acceptable except in the case where the applicant has assumed exceptionally low rates of infiltration. In these cases, more substantial verification or proof may be required. For instance, if an applicant submits an analysis that shows a gross leakage rate of less than 0.06 volume changes per hour the reviewer would require that the gross leakage be verified by periodic tests as described in Regulatory Position C.5 of Regulatory Guide 1.95.

(3) Zone isolation, with filtered recirculated air, and with a positive pressure. This system is essentially the same as the preceding one. However, an additional operational mode is possible. Makeup air for pressurization is admitted. It is filtered before entering the emergency zone. Pressurization reduces the unfiltered inleakage that is assumed to occur when the emergency zone is not pressurized. Assuming a filter fan capacity of 5000 cfm and a filter efficiency of 95%, the following protection factors result (flows in cfm):

Makeup Air	Recirculated Air	IPF (Assuming No Infiltration)	IPF (Assuming Infiltration*)
400	4600	238	159
750	4250	128	101
1000	4000	96	80

*Normally 10 cfm infiltration is assumed for conservatism. This flow could be reduced or eliminated if the applicant provides assurance that backflow (primarily as a result of ingress and egress) will not occur. This may mean installing two-door vestibules or equivalent.

The makeup flow rate should have adequate margin to assure that the control room will be maintained at a pressure of at least 1/8-inch water gauge. The applicant should indicate that an acceptance test will be performed to verify adequate pressurization. If the makeup rate is less than 0.5 volume changes per hour, supporting calculations are required to verify adequate air flow. If the makeup rate is less than 0.25 volume changes per hour, periodic verification testing is required in addition to the calculations and the acceptance test.

A question that often arises is whether "pressurization" or "isolation and recirculation" of the control room is to be preferred. Which design gives the lowest doses depends on the assumptions as to unfiltered inleakage. Isolation is generally preferred in that it will limit the entrance of noble gases (not filterable) and, in addition, it is a better approach when the accident involves a short term "puff release." If infiltration is 25 cfm or less, "isolation" would be best in any event.

A second question related to the first involves the method of operation. The following possibilities have been considered:

(i) Automatic isolation with subsequent manual control of pressurization.

(ii) Automatic isolation with immediate automatic pressurization.

The first is advantageous in the case of external puff releases. Simple isolation would minimize the buildup of the unfilterable noble gases. It would also protect the filters from excessive concentrations in the case of a chlorine release. However, the second method does guarantee that infiltration (unfiltered) is reduced to near zero immediately upon accident detection. This would be beneficial in the case where the contamination transport path to the emergency zone is mainly inside the building. Method (i) should be used in the case of a toxic gas release and either method (i) or (ii) should be used in the case of a radiological release, provided Criterion 19 guidelines can be satisfied. (A substantial time delay should be assumed where manual isolation is assumed, e.g., 20 minutes for the purposes of dose calculations.)

(4) Dual air inlets for the emergency zone. Several plants have utilized this concept. The viability of the dual inlet concept depends upon whether or not the placement of the inlets assures that one inlet will always be free from contamination. The assurance of a contamination-free inlet depends in part upon building wake effects, terrain, and the possibility of wind stagnation or reversal. For example, in a situation where the inlets are located at the extreme edges of the plant structures (e.g., one on the north side and one on the south side), it is possible under certain low probability conditions for both inlets to be contaminated from the same point source. Reference 7 presents the interim position for dealing with the evaluation of X/Q's for dual inlet systems. These X/Q's are used only if the system incorporates automatic selection of the best inlet. If manual selection is used, the X/Q's are increased assuming that the worst inlet is operating as follows:

Time after accident	Hours of improper operation
0 - 8 hr	2.4 hr out of 8 hr
8 - 24 hr	3.2 hr out of 16 hr
1 - 4 days	2.4 hr each day
4 - 30 days	1.2 hr each day

Because damage to the ducting might seriously affect the system capability to protect the operators, the ducting should be seismic Category I and should be protected against tornado missiles. In addition, the number and placement of dampers must be such as to assure both flow <u>and</u> isolation in each inlet assuming one single active component failure. The location of the intakes with respect to the plant security fence should also be reviewed.

(5) Bottled air supply for a limited time. In some plant designs the containment pressure is reduced below atmospheric within one hour after a DBA. This generally assures that after one hour significant radioactive material will not be released from the containment. Such a design makes it feasible to maintain the control room above atmospheric pressure by use of bottled air.

Periodic pressurization tests are required to determine that the rated flow (normally about 300 to 600 cfm) is sufficient to pressurize the control room to at least 1/8-inch water gauge. The system is also required to be composed of several separate circuits (one of which is assumed to be inoperative to account for a possible single failure). At least one (non-redundant) once-through filter system for pressurization as a stand-by for accidents of long duration is also desirable.

Compressed air bottles should be protected from tornado missiles or internally generated missiles and should be placed so as not to cause damage to vital equipment or interference with operation if they fail.

4. Emergency Standby Filters
Refer to SRP 6.5.1.

5. Relative Location of Source and Control Room
This review area involves identification of all potential sources of toxic, radioactive, or otherwise potentially hazardous gases and analysis of their transport to the control room. There are three basic categories: DBA radioactive sources, toxic gases such as chlorine, and gases with the potential for being released inside confined areas adjacent to the control room.

a. DBA Radioactive Sources
The LOCA source terms determined in Appendix A to SRP 15.6.5 review are referred to and routinely used to evaluate radiation levels external to the control room. The dispersal from the containment or the standby gas-treatment vent is determined with a building wake diffusion model. This model is discussed in Reference 7. Other DBA's are reviewed to determine whether they might constitute a more severe hazard than the LOCA. If this is suspected, then an additional analysis is performed for the suspect DBA's. The SAB reviews the AAB meteorological analysis and compares it with site meteorological data as it becomes available.

b. Toxic Gases
The applicant is asked to identify those toxic substances stored (or transported) on or in the vicinity of the site which may pose a threat to the reactor operators by producing toxic gases upon accidental release. The method used to determine whether the quantity or location of the toxic material is such as to require closer study is described in Regulatory Guide 1.78 (Ref. 3). This guide also discusses the methods for analyzing the degree of risk and states, in general terms, the various protective measures that could be instituted if the hazard is found to be too great. In the case of chlorine, specific acceptable protective provisions have been determined; these are described in detail in Reference 4.

In summary, the following provisions or their equivalent are required (pertaining to the emergency zone ventilation system):

(1) Quick acting toxic gas detectors.

(2) Automatic emergency zone isolation.

(3) Emergency zone leak tightness.

(4) Limited fresh air makeup rates.

(5) Breathing apparatus and associated bottled air supply.

(Note that the best solution for a particular case will depend on the toxic gas in question and on the specific ventilation system design.)

c. Confined Area Releases

The reviewer studies the control building layout in relation to potential sources inside the control building or adjacent connected buildings. The following concerns are checked:

(1) Storage locations of CO_2 or other firefighting materials should be such as to eliminate the possibility of significant quantities of the gases entering the emergency zone. (The APCSB has the primary responsibility in this area.)

(2) The ventilation zones adjacent to the emergency zone should be configured and balanced to preclude air flow toward the emergency zone.

(3) All pressurized equipment and piping (e.g., main steam lines and turbines) that could cause significant pressure gradients when failed inside buildings should be isolated from the emergency zone by multiple barriers such as multiple door vestibules or their equivalent.

6. Radiation Shielding

Control room operators as well as other plant personnel are protected from radiation sources associated with a normally operating plant by various combinations of shielding and distance. The adequacy of this type of protection for normal operating conditions is reviewed and evaluated by the RAB. To a large extent the same radiation shielding (and missile barriers) also provides protection from design basis accident radiation sources. This is especially true with respect to the control room walls which usually consist of at least 18 inches of concrete. In most cases, the radiation coming from external design basis accident radiation sources is attenuated to negligible levels. However, the following items should be considered qualitatively in assessing the adequacy of control room radiation shielding:

a. Control room structure boundary. Wall, ceiling, and floor materials and thickness should be reviewed. Eighteen inches to two feet of concrete or its equivalent will be adequate in most cases.

b. Radiation streaming. The control room structure boundary should be reviewed with respect to penetrations (e.g., doors, ducts, stairways, etc.). The potential for radiation streaming from accident sources should be identified, and if deemed necessary, quantitatively evaluated. Support from the RAB may be required for some radiation streaming dose calculations.

c. Radiation shielding from internal sources. If sources internal to the control room complex are identified, radiation shielding against them should be reviewed. Typical sources in this category include contaminated filter trains, or airborne radioactivity in enclosures adjacent to the control room.

Evaluations of radiation shielding effectiveness with respect to the above items should be performed using simplified analytical models for point, line, or volume sources such as those presented in References 10 and 11. If more extended analysis is required, analytical support from the RAB should be requested. The applicant's coverage of the above items should also be reviewed in terms of completeness, method of analysis, and assumptions.

7. Independent Analyses

a. Control Room Doses

Notwithstanding the fact that the applicant is required to calculate dose to control room operators, independent analyses are made by the AAB. Using the approach indicated in Reference 7, the source terms and doses due to a DBA are calculated. The source terms determined by the AAB's independent analysis of LPZ doses for a LOCA are used. The methods and assumptions for this calculation are presented in Appendix A to SRP 15.6.5. The control room doses are determined by estimating the X/Q from the source points to the emergency zone (see above), by determining the credit for the emergency zone's protection features, and by calculating the dose. Figure 6.4-1 shows a form which may be used to summarize the information that is needed for the control room dose calculation. The effective X/Q's are used for calculating the doses. The dose is then compared with the guidelines of General Design Criterion 19. If the guidelines are exceeded, the applicant is asked to improve the system. In the event that other DBA's are expected to result in doses comparable to or higher than the LOCA, additional analyses are performed. The limiting accidents are compared with Criterion 19.

b. Other Analyses

Special case analyses are performed when questions are raised about certain potential sources of toxic or radioactive gases. The methods used in these analyses conform to current DBA methods concerning dispersion and dose calculations. Regulatory Guide 1.78 should be consulted by the site analyst to see if nearby facilities could present a potential hazard that requires detailed analysis.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

a. If the plant meets Criterion 19, the following statement or its equivalent is made:

> "The applicant proposes to meet General Design Criterion 19 of Appendix A to 10 CFR Part 50 by use of concrete shielding and by installing redundant _____ cfm recirculating charcoal filters in the control room ventilation system. These filters will be automatically activated upon an accident signal, high radiation signal, or high chlorine signal. Independent calculations of the potential radiation doses to control room personnel following a LOCA show the resultant doses to be within the guidelines of Criterion 19."

b. If the design is not adequate, the fact is stated. Alternatives such as an increase in the charcoal filter flow rate may be indicated as is given in the example below:

> "The staff has calculated the potential radiation doses to control room personnel following a LOCA. The resultant whole body doses are within the guidelines of Criterion 19. The thyroid dose resulting from exposure to radioactive iodine exceeds the dose guidelines. The applicant will be required to commit to increasing the filtration system size from 2000 cfm to 4000 cfm. This increased filtration will be sufficient to keep the estimated thyroid doses within the guidelines."

c. If special protection provisions for toxic gases are not required, the following statement or its equivalent is made:

> "The habitability of the control room was evaluated using the procedures described in Regulatory Guide 1.78. As indicated in Section 2.2, no offsite storage or transport of chemicals is close enough to the plant to be considered a hazard. There are no onsite chemicals that can be considered hazardous under Regulatory Guide 1.78. A sodium hypochlorite biocide system will be used, thus eliminating an onsite chlorine hazard. Therefore special provisions for protection against toxic gases will not be required. Self-contained breathing apparatus is provided for the emergency crew to provide assurance of control room habitability in the event of occurrences such as smoke hazards."

d. If special protection provisions are required, compliance or non-compliance with the guidelines of Regulatory Guides 1.78 and 1.95 should be stated.

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

2. Regulatory Guide 1.53, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

3. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

4. Draft Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

5. Draft Regulatory Guide 8.X, "Acceptable Programs for Respiratory Protection."

6. "Manual of Respiratory Protection Against Airborne Radioactive Material," WASH-1287, U.S. Atomic Energy Commission (1974).

7. K.G. Murphy and K.M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

8. "Leakage Characteristics of Openings for Reactor Housing Components," NAA-SR-MEMO-5137, Atomics International, Div. of North American Aviation, Inc., June 20, 1960.

9. R.L. Koontz, et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nucl. Soc., November 1961.

10. R.G. Jaeger, et al., eds., "Engineering Compendium on Radiation Shielding," Vol. 1, "Shielding Fundamentals and Methods," Springer-Verlag (1968).

11. N.M. Schaeffer, "Reactor Shielding for Nuclear Engineers," TID-75951, U.S. Atomic Energy Commission.

FIGURE 6.4-1 Summary Sheet for Control Room Dose Analysis

MEMORANDUM TO: (Site Analyst)
(Meteorologist)

cc: E. Markee
B. Grimes (Habitability File)

CONCERNING CONTROL ROOM DOSE ANALYSIS FOR

The following summarizes the X/Q's used in determining the control room operator dose for the subject plant:

VENTILATION SYSTEM DESCRIPTION

SKETCH OF SYSTEM (and inlets/sources if applicable)

SUMMATION OF X/Q ANALYSIS

Source/Receptor Type and Distance

S/D Ratio K Factor

Number of 22 1/2° Wind Direction Sectors that Result in Exposure

Central Wind Sector (sector wind is blowing from)

5% Wind Speed (m/sec) 40% Wind Speed (m/sec)

Projected Area of Wake(m^2) 5% X/Q (sec/m^3)

Time	Wind Speed Factor	Wind Direction Factor	Occupancy Factor	Effective X/Q's
0-8 hr	1	1	1	
8-24 hr			1	
1-4 day			0.6	
4-30 day			0.4	

ACTION REQUESTED

Site Analyst

- For your information only
- Please use the effective X/Q's in TACT run and provide control room doses. In addition, please summarize safety system assumptions and indicate their status (interim or final).

Meteorologist

- These are interim X/Q's. Please review to determine their reasonableness.
- These are final X/Q's. Please determine if they are accurate based on your analysis of site data.

Please Contact________________________

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.5.1 ESF FILTER SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Accident Analysis Branch (AAB)

I. AREAS OF REVIEW

At the construction permit (CP) stage of review, ETSB reviews the information in the applicant's safety analysis report (SAR) in the areas listed below. At the operating license (OL) stage, the ETSB review consists of confirming the design accepted at the CP stage and evaluating the adequacy of the applicant's technical specifications in these areas. The specific review areas are as follows:

1. The engineered safety feature (ESF) air filtration units designed for fission product removal in post-accident environments. These generally include primary systems, e.g., recirculation (in-containment), and secondary systems, including standby gas treatment systems and the emergency air cleaning systems for the fuel handling building, control room, and shield building and areas containing engineered safety feature components.

2. The system design, design objectives and design criteria. The ETSB reviews the methods of operation and the factors that influence the filtration capabilities of the system, e.g., system interfaces and potential bypass routes. The components included in each atmospheric cleanup system and the seismic design category of each system are reviewed. Redundancy of the atmosphere cleanup systems, the physical separation of the redundant trains, and the volumetric air flow rate of each train are reviewed.

3. The environmental design criteria, the design pressure and pressure differential, integrated radiation dose rate, relative humidity, maximum and minimum temperature, radiation source term, and the shielding of essential services such as power and electrical control cables associated with the atmosphere cleanup systems.

4. The component design criteria and qualification testing, qualification requirements of demisters, prefilters, and high-efficiency particulate air (HEPA) filters, design requirements of the filter and adsorber mounting frames, system filter housings, and water drains, the adsorbent used for removal of gaseous iodines (in the preliminary safety analysis report, PSAR), the physical properties of the adsorbent and the

design of the adsorber section of the filter trains (in the final safety analysis report, FSAR). Provisions to inhibit offdesign temperatures in the adsorber section and the design criteria of the system fans or blowers, ductwork, and housings are also reviewed.

5. Design provisions incorporated in the equipment and features to facilitate operation and maintenance. The design of doors to the filter housings, the spacing of components, alignment and support of filter elements, the spacing of filter elements in the same bank, design of test probes, and provisions for adequate lighting in the filter housing are also reviewed.

6. The design criteria for inplace testing of the air flow distribution to the HEPA filters, dioctyl phthalate (DOP) testing of the HEPA filter sections, and gaseous halogenated hydrocarbon refrigerant testing of the activated carbon adsorber section.

7. The laboratory testing criteria for the activated carbon adsorbent, qualification and batch tests, provisions for obtaining representative adsorbent samples for laboratory testing in order to estimate the amount of penetration of the system adsorbent throughout its service life (PSAR), and the provisions and conditions for each field and laboratory test (FSAR).

The review of the ESF filter systems involves secondary review evaluations performed by other branches. The conclusions from their evaluations on request are used by ETSB to complete the overall evaluation of the facility. The evaluations provided by the branches are as follows:

EICSB reviews the associated instrumentation including the power supply and electrical distribution systems under Standard Review Plans (SRP) 3.11, 7.3, 7.5, and 8.2; AAB calculates the doses that result as a consequence of postulated accidents under the SRP for Chapters 6, 9, and 15 of the SAR.

II. ACCEPTANCE CRITERIA

The installed ESF filter systems are needed to mitigate the consequences of postulated accidents by removing from the atmosphere radioactive material that may be released in the event of an accident. ETSB will accept ESF filter systems if the following criteria are met:

1. Air filtration units should be designed so that they can operate after a design basis accident (DBA) and retain radioactive material after the DBA.

2. For the system design, ETSB will use the following guidelines for determining acceptability:

    a. Each atmosphere cleanup train should be able to prefilter the air, remove moisture ahead of charcoal adsorbers, and remove particulate matter by HEPA filters before and after the charcoal adsorbers.

b. Redundancy of filter trains should be provided, with the trains physically separated so that damage to one system will not cause damage to the other system.

c. All components should be designated as seismic Category I, if failure of the component would lead to the release of fission products.

d. Individual trains should be limited to a volumetric air flow rate of 30,000 cfm.

e. Each train should be instrumented to signal, alarm, and record pressure drop and flow rate at the control room.

3. For environmental design, ETSB will use the following guidelines to determine acceptability:

   a. Expected conditions for the filter trains, including maximum and pressure differential, radiation dose rate, relative humidity, and maximum and minimum temperature, should be based on the conditions in a postulated DBA.

   b. The radiation source terms should be consistent with the guidelines in Regulatory Guides 1.3, 1.4, and 1.25.

   c. Shielding should be provided for essential services such as power and electrical control cables associated with the atmosphere cleanup system.

4. For component design and qualification testing, ETSB will use the following guidelines to determine acceptability:

   a. The demisters should be designed in accordance with the recommendations of MSAR 71-45 (Ref. 7) and meet the Underwriters' Laboratory (UL) Class 1 requirements (Ref. 8).

   b. Moisture removal equipment should be capable of reducing the relative humidity of the incoming atmosphere from 100% to 70%.

   c. If prefilters are provided, they should meet UL Class 1 requirements and be listed in the current UL Building Materials List (Ref. 9).

   d. HEPA filters should be designed in accordance with the recommendations of MIL-F-51068 D (Ref. 10) and MIL-F-51079 B (Ref. 11).

   e. Filter and adsorber mounting frames should be designed, arranged, and constructed in accordance with the recommendations of Sections 4.3 and 4.4 of ORNL-NSIC-65 (Ref. 12).

   f. Filter housings, including floors and doors, should be designed and constructed in accordance with the recommendations of Sections 4.5.2, 4.5.5, 4.5.7, and 4.5.9 of ORNL-NSIC-65.

g. Water drains should be designed in accordance with the recommendations of Section 4.5.6 of ORNL-NISC-65.

h. The adsorbent to be used for adsorbing gaseous iodine (elemental iodine and organic iodides) should be an adsorbent that has been demonstrated to remove the gaseous iodines from air at the required efficiencies listed in Table 2 of Regulatory Guide 1.52 Rev. 1 (PSAR). If impregnated activated charcoal is the adsorbent, the physical properties of the adsorbent should be in accordance with the guidelines of Table 2 of Regulatory Guide 1.52 Rev. 1 (FSAR). If an adsorbent other than impregnated activated charcoal is proposed, ETSB will review supporting data and accept adsorbents expected to perform equal to, or better than, impregnated activated charcoal.

i. The adsorber should be designed for a maximum loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated charcoal.

j. Provisions should be included to inhibit off-design temperatures in the adsorber section. To dissipate heat generated from iodine decay and charcoal oxidation effects, ETSB will consider cooling mechanisms such as low flow air bleed systems and cooling coils. To extinguish ignited charcoal, ETSB will consider water sprays, carbon dioxide injection systems, and liquid nitrogen cooling systems.

k. The system fan, its mounting, and ductwork connections should be designed and constructed in accordance with the recommendations of Section 2.7 of ORNL-NSIC-65.

l. Ductwork should be designed in accordance with the recommendations of Section 2.8 of ORNL-NSIC-65.

5. ETSB will accept ESF filter systems that are designed for accessibility of components and ease of maintenance in accordance with the recommendations of Section 2.5 of ORNL-NSIC-65 as follows:

   a. Components to be replaced should be provided with a minimum of three linear feet from mounting frame to mounting frame between banks of components; components to be replaced should be provided with a minimum of three linear feet plus the maximum length of the component.

   b. Provisions should be made for permanent test probes with external connections.

6. For in-place testing, ETSB will use the following guidelines for determining acceptability:

   a. Provisions should be made for visual inspection of the system and all associated components in accordance with the recommendations of Section 5 of ANSI Standard N510 (Ref. 6).

b. Provisions should be made for testing the air flow distribution upstream of HEPA filters and charcoal adsorbers, and demonstrating uniformity ± 20% of averaged flow per unit.

c. Provisons should be made for DOP testing of the HEPA filter sections in accordance with the recommendation of ANSI N510 (Ref. 6).

d. Provisions should be made for leak-testing the activated carbon adsorber section with a gaseous halogenated hydrocarbon refrigerant in accordance with the recommendations of ANSI N510 (Ref. 6).

e. Provisons should be made for in-place testing initially, and routinely thereafter. Frequency and testing requirements will be established in the technical specifications.

7. For laboratory testing of activated carbon adsorbent, ETSB will use the following guidelines for determining acceptability:

   a. Qualification and batch tests on new unused adsorbent should be performed in accordance with the guidelines of Table 2 in Regulatory Guide 1.52.

   b. Provisions should be made for obtaining representative adsorbent samples in order to estimate the amount of penetration of the system adsorbent throughout its service life (PSAR).

   c. Provisions should be made for laboratory testing initially, and routinely thereafter. Frequency and testing requirements will be established in the technical specifications.

ETSB will accept the following deviations from the above acceptance criteria for the post loss-of-coolant accident (LOCA) hydrogen purge filtration system:

1. If the calculated dose (sum of the long-term doses from the LOCA and the purge dose at the low population zone outer boundary) is less than the guidelines of 10 CFR Part 100, no filtration system is required.

2. If a radioiodine decontamination factor of 10 or less is needed for the calculated dose to be below Part 100, a filtration system that meets the acceptance criteria listed in Item 5 of Acceptance Criteria in SRP 11.3 should be provided.

3. If a radioiodine decontamination factor of greater than 10 is needed for the calculated dose to be below Part 100, the filtration system should meet all of the above acceptance criteria, except for Items 2b and 2c.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. In the ETSB review the plant design is reviewed to determine where ESF units are needed.

2. The ETSB review is carried out by making a detailed comparison of filtration unit designs with the acceptance criteria of Section II, above. The capability of a system to remove fission products in the atmosphere after a DBA is reviewed, based on a design loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated charcoal adsorbent. Designs consistent with the guidelines of Regulatory Guide 1.52 will be assigned the system efficiencies for removal of elemental iodine and organic iodides given in Table 2 of Regulatory Guide 1.52 and a system efficiency of 99% for removal of particulates resulting from a DBA. The assigned efficiencies are for Accident Analysis Branch use in accident analyses to calculate offsite doses to the whole body and thyroid.

IV. EVALUATION FINDINGS

ETSB verifies that sufficient information has been provided and that the review is adequate to support conclusions of the following type, to be included in the staff's safety evaluation report:

> "The ESF atmosphere cleanup systems include the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated design basis accident. The scope of our review included an evaluation of these systems with respect to the guidelines of Regulatory Guide 1.52. We have reviewed the applicant's system descriptions and design criteria for the ESF air filtration units. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the ESF air filtration units to applicable regulations and guides and to staff technical positions and industry standards. Based on our evaluation, we find the proposed ESF air filtration units are acceptable, and the filter efficiencies given in Table 2 of Regulatory Guide 1.52 are appropriate for use in accident analyses."

V. REFERENCES

1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors."

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

3. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

5. ANSI N510, "Testing of Nuclear Air Cleaning Systems," American National Standards Institute (1975).

6. G. H. Griwatz, J. V. Friel, and J. L. Bicehouse, "Entrained Moisture Separators for Fine (1-10u) Water-Air-Steam Service: Their Performance, Development and Status," MSAR 71-45, Mine Safety Appliances Research Corporation (1971).

7. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.

8. "Building Materials List," Underwriters' Laboratories, Inc.

9. MIL-F-51068 D, "Filter, Particulate, High Efficiency, Fire Resistant," Government Printing Office (1974).

10. MIL-F-51079 B, "Filter Medium, Fire Resistant, High Efficiency," Government Printing Office (1974).

11. "Design, Construction, and Testing of High Efficiency Air Filtration Systems for Nuclear Application," ORNL-NSIC-65, Oak Ridge National Laboratory (1970).

12. 10 CFR Part 100, "Reactor Site Criteria."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.5.2 CONTAINMENT SPRAY AS A FISSION PRODUCT CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The AAB reviews the containment spray and spray additive system to determine the fission product removal effectiveness of the system whenever the applicant claims a containment air cleanup function for the system. The heat removal and hydrogen mixing functions (where applicable) are reviewed by the CSB. The sump design is also reviewed by the CSB. Long-term pH requirements are reviewed by the AAB under Standard Review Plan (SRP) 6.1.3.

The following areas of the applicant's safety analysis report (SAR) relating to the fission product removal and control function of the containment spray system are reviewed by the AAB:

1. Fission Product Removal Requirement for Containment Spray
   Sections of the SAR related to accident analyses, dose calculations, and fission product removal and control are briefly reviewed to establish whether fission product scrubbing of the containment atmosphere for the mitigation of offsite doses following a postulated accident is claimed by the applicant. This review usually covers Sections 6.2.3.1, 6.5.2.1, and 15.X of the SAR (Ref. 1.).

2. Design Bases
   The design bases of such containment spray systems are reviewed to determine whether they reflect the requirements placed upon this system by the assumptions made in the accident evaluations of Chapter 15.

3. System Design
   The descriptive information concerning the design of the spray system, including any subsystems and supporting systems, is reviewed to familiarize the reviewer with the design and operation of the system. The review includes:

a. The descriptive information contained in SAR Sections 6.5.2.2, 6.5.2.4, 6.5.2.5, and 6.5.2.6 (to establish the basic design concept), the systems, subsystems, and support systems required to carry out the iodine scrubbing function of the system, and the components and instrumentation employed in these systems.

b. The process and instrumentation diagrams of SAR Section 6.5.2 or 6.2.2 (whichever contains the relevant information).

c. Layout drawings, (plans, elevations, isometrics) of the spray distribution headers, from SAR Section 6.5.2 or 6.2.2.

d. Plan views and elevations of the containment layout.

e. Process and instrumentation diagrams of any ventilation systems operational in the post-accident environment.

4. Testing and Inspections

Section 6.5.2.5 of the SAR is reviewed to establish the details of the pre-operational test to be performed for system verification and the post-operational tests and inspections to be performed for verification of the continued status of readiness of the spray system.

5. Technical Specifications

At the operating license stage, Sections 3.0 and 4.0 of Chapter 16 of the applicant's final safety analysis report are reviewed to establish permissible outage times and surveillance requirements.

II. ACCEPTANCE CRITERIA

1. General Design Criteria

Criterion 41 ("Containment Atmosphere Cleanup"), Criterion 42 ("Inspection of Containment Atmosphere Cleanup Systems"), and Criterion 43 ("Testing of Containment Atmosphere Cleanup Systems") of Appendix A of 10 CFR Part 50 are used as acceptance criteria.

2. Specific Design Requirements for Iodine Removal Function

a. System Operation

The containment spray system should be designed to be initiated automatically by an appropriate accident signal and should be capable of continuous operation thereafter until the design objectives of the system have been achieved. In all cases the operating period should not be less than two hours. In addition, the system should be capable of operation in the recirculation mode on demand, for a period of at least one month following the postulated accident.

b. Coverage of Containment Volume

In order to assure full coverage of the containment volume, the following should be observed:

(1) The spray nozzles should be located as high in the containment as practicable, to maximize the spray drop fall distance.

(2) The layout of the spray nozzles and distribution headers should be such that the cross-sectional area of the containment covered by the spray is maximized and that a nearly homogeneous distribution of spray in the containment volume is produced. At the operating deck level, at least 95% of the cross section of the containment should be covered. Unsprayed regions in the upper containment and, in particular, an unsprayed annulus adjacent to the containment liner should be avoided wherever possible.

(3) In designing the layout of the spray nozzle positions and orientations, the effect of the post-accident atmosphere should be considered, including the effects of post-accident conditions that result in the maximum possible atmosphere density.

c. Promotion of Containment Mixing

Because the effectiveness of the containment spray system depends on a well-mixed containment atmosphere, all design features enhancing post-accident mixing should be considered. Where necessary, forced air ventilation should be provided to avoid stagnant air regions.

d. Spray Nozzles

The nozzles used in the containment spray system should be of a design that minimizes the possibility of clogging while producing drop sizes effective for iodine absorption. The nozzles should not have internal moving parts such as swirl vanes, turbulence promoters, etc. They should not have orifices or internal restrictions which would narrow the flow passage to less than 1/4 inch in diameter.

e. Injection Spray Solution

The partition of iodine between liquid and gas phases is enhanced by the alkalinity of the solution. The spray system should be design such that the spray solution maintains the highest possible pH, within materials compatibility constraints. This requirement is satisfied by a spray pH in the range of 8.5 to 11.0. Iodine scrubbing credit is given for spray solutions whose chemistry, including any additives, has been demonstrated to be effective for iodine absorption and retention under post-accident conditions. Both theoretical and experimental verification are required. The following solutions have been shown to meet these requirements:

(1) Boric acid solution, 1500 - 2500 ppm boron.

(2) Boric acid solution buffered with NaOH to a pH of 8.5 - 11.0.

(3) Boric acid solution with a minimum of 1.0% by weight of thiosulfate, buffered with NaOH to a pH of 8.5 - 11.0.

(4) Trace level hydrazine (50 ppm nominal) solution.

f. Containment Sump Mixing

The containment sump should be designed to promote mixing of emergency core cooling system (ECCS) and spray solutions. Drains to the engineered safety features (ESF) sump should be provided for all regions of the containment which would collect a significant quantity of the spray solution. Alternatively, allowance should be made for "dead" volumes in the determination of sump pH and the quantities of additives injected.

g. Containment Sump Solutions

The pH of the aqueous solution collected in the containment sump after completion of injection of containment spray and ECCS water, and all additives for reactivity control, fission product removal, or other purpose, should be maintained at a high level. The equilibrium sump pH, after mixing and dilution with the primary coolant and ECCS injection, should be above 8.5 for sodium hydroxide and sodium thiosulfate sprays. A pH value exceeding 8.5 provides assurance that significant evolution of iodine does not occur.

h. Storage of Additives

The design should provide facilities for the long-term storage of all spray additives. These facilities should be designed such that the additives required to achieve the design objectives of the system are stored in a state of continual readiness whenever the reactor is critical during the design life of the plant. The storage facilities should be designed such that freezing, precipitation, chemical reaction, and decomposition of additives are prevented. For NaOH storage tanks, heat tracing of tanks and piping is required whenever exposure to temperatures below 40°F is predicted. An inert cover gas should be provided for solutions with an NaOH concentration of 30% by weight or higher.

i. Single Failure

The system should be able to function effectively and meet all the above criteria with a single failure of an active component in the spray system, any of its subsystems, or any of its support systems. The system is considered functional with respect to iodine removal if it is capable of delivering the design spray flow rate with the additive concentration within the acceptable range as determined above.

3. Testing

Tests should be performed to demonstrate the spray systems, as installed, meet all design requirements for an effective iodine scrubbing function. Such tests should include verification of:

a. Freedom of the containment spray nozzles from obstructions.

b. Capability of the system to deliver the required spray flow.

c. Capability of the system to deliver the required spray additives within the specified range of concentrations.

4. Technical Specifications

The technical specifications should specify appropriate limiting conditions for operation (LCO's), tests, and inspections to provide assurance that the system is capable of its design function whenever the reactor is critical. These specifications should include:

a. The operability requirements for the system, including all active and passive devices, as a limiting condition for operation (with acceptable outage times). The following should be specifically included:

(1) Containment spray pumps.

(2) Additive pumps (if any).

(3) Additive mixing devices (if any).

(4) Additive quantity and concentration in the additive storage tanks.

(5) Nitrogen or other inert gas pressure in the additive storage tanks.

b. Periodic inspection and sampling of the contents of the additive tanks to confirm that the additive quantity and concentrations are within the limits established by the system design.

c. Periodic testing and exercising of the active components of the system and verification that essential piping and passive devices are free of obstructions.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The review of the fission product removal function of the containment spray system follows the procedure outlined below.

The reviewer determines whether the containment spray system is used for fission product removal purposes. (Chapter 15 of the SAR should be reviewed to establish whether a fission product removal function for the containment spray system is assumed in accident dose evaluations.)

If the containment spray system is not used for dose mitigation purposes, i.e., if it is used for heat removal only, no further review is required. (The CSB reviews the heat removal and hydrogen mixing aspects of the containment spray system.) If the containment spray system is designed to reduce the concentrations of fission products in the containment, the capability of the system to function effectively as a fission product removal system is reviewed. If, as a result of the review, system modifications are required, the CSB is informed of the required modifications for integration with any other requirements placed on the containment spray system.

1. <u>System Design</u>

   Review of the system design includes an examination of the components and design features necessary to carry out the iodine scrubbing function, including:

   a. <u>Spray Chemistry</u>

      The forms of iodine for which spray removal credit is claimed in the accident analyses (SAR Chapter 15) are established. Containment spray systems may be designed for removal of iodine in the vapor (elemental) form, in the form of organic compounds, and in the form of iodine adsorbed on airborne particulate matter.

      The systems or subsystems required to carry out the iodine scrubbing function of the containment spray, such as the spray system, recirculation system, spray additive system, and water source are identified.

      The conceptual designs of the systems involved are reviewed:

      (1) To determine the chemical additive and to ascertain the effectiveness of the additive for elemental and organic iodine removal by comparison with additives of proven effectiveness (see acceptance criteria in Section II) or by review of theoretical and experimental verifications supplied for new additives.

      (2) To ascertain that the range of additive concentrations is within the limits listed in the acceptance criteria of Section II or that adequate justification is supplied for the iodine removal and retention effectiveness, radiolytic and pyrolytic decomposition, corrosion, and solidification and precipitation behavior of the chemical additives for the range of concentrations encountered. The concentrations in the storage facility, the chemical addition lines, the spray solution injection, the containment sump solution, and the recirculation spray solution should be examined. The extremes of the additive concentrations should be determined with the most adverse combination of ECCS, spray, and additive pumps (if any) assumed to be operating, and a single active failure of pumps or valves should be considered.

   b. <u>System Operation</u>

      The time and method of system initiation, including additive addition, is reviewed to confirm that the acceptance criteria of Section II are met. Automatic initiation of spray and spray additive flow, without mechanical delays or manual overrides, is required. The system operation should be continuous until the iodine removal objectives of the system are met. If a switch-over from the injection to a recirculation mode of operation is required, the reviewer should confirm that all requirements listed in the acceptance criteria, particularly those concerning spray coverage and solution pH, are met during the recirculation phase.

c. Spray Distribution and Containment Mixing

The number and layout of the spray headers used to distribute the spray flow in the containment are reviewed. The reviewer verifies that the layout of the headers assures coverage of essentially the entire cross-section of the containment with spray, under minimum spray flow conditions. The effect of the high temperature and pressure conditions in the containment on the spray droplet trajectories should be taken into account in determining the area covered by the spray.

The layout of the containment and forced ventilation systems (safety-grade) operating after the loss-of-coolant accident (LOCA) are reviewed to determine any areas of the containment free volume that are not sprayed. The rate of mixing (either through forced ventilation or by demonstrated convective mixing) between the sprayed and unsprayed compartments of the containment is evaluated. The containment may be considered a single, well-mixed volume provided the spray covers regions comprising at least 90% of the containment volume and provided a ventilation system is available for adequate mixing of any unsprayed compartments.

d. Spray Nozzles

The design of the spray nozzles is reviewed to confirm that the spray nozzles are not subject to clogging from debris entering the recirculation system through the sump screens.

e. Sump Mixing

The mixing of the spray water containing the chemical additive and water without additive (such as spilling ECCS coolant) in the containment sump is reviewed. The areas of the containment which are exposed to the spray but are without direct drains to the recirculation sump (such as the refueling cavity) are considered. The reviewer confirms that the required sump concentrations are achieved within the appropriate time intervals. In addition, the long-term sump pH requirements (see Standard Review Plan 6.1.3) are considered for systems with low pH values.

The equilibrium partitioning of iodine between the sump liquid and the containment atmosphere is examined for the extremes of the additive concentrations determined above, in combination with the range of temperatures possible in the containment atmosphere and the sump solution. The minimum iodine partition coefficient determined for these conditions forms the basis of the ultimate iodine decontamination factor allowed in the staff's analysis, described below. (See Ref. 3 for a theoretical examination of iodine partition coefficients.)

f. Storage of Additives

The design of the additive storage tanks is reviewed to establish whether heat tracing is required to prevent freezing or precipitation in the tanks. The reviewer determines whether an inert cover gas is provided for the tanks to prevent reactions of the additive with air, such as the formation of sodium carbonate by the reaction of sodium hydroxide and carbon dioxide. Alternatively, the reviewer verifies by a conservative analysis that an inert cover gas is not required.

g. Single Failure

The system schematics are reviewed by inspection, postulating single failures of any active component in the system, including inadvertent operation of valves that are not locked open. The review is performed with respect to the iodine removal function, considering conditions that could result in too fast as well as too slow an additive injection.

2. Testing

At the construction permit stage, the containment spray concept and the proposed tests of the system are reviewed to confirm the feasibility of verifying the design functions by appropriate testing. At the operating license stage, the proposed tests of the system and its components are reviewed to verify that the tests will demonstrate that the system, as installed, is capable of performing, within the bounds established in the description and evaluation of the system, all functions essential for effective iodine removal following postulated accidents.

3. Technical Specifications

The technical specifications are reviewed to verify that the system, as designed, is capable of meeting the design requirements and that it remains in a state of readiness whenever the reactor is critical.

a. Limiting Conditions for Operation

The LCO's should require the operability of the containment spray pumps, all associated valves and piping, the spray additive tanks including the appropriate quantity of additives, and any metering pumps or mixing devices.

b. Tests

Pre-operational testing of the system, including the additive tanks, pumps, (if any), piping, and valves is required, as discussed above. In particular, the preoperational testing should verify that the system, as installed, is capable of delivering a well-mixed solution containing all additives with concentrations falling within the design margins assumed in the dose analyses of Chapter 15 of the SAR.

Periodic testing and exercising of all active components should include the spray pumps, metering pumps (if any), and valves. Confirmation that passive components, such as all essential spray and spray additive piping, and any passive mixing devices are free of obstructions should be made periodically. The contents of the spray additive tanks should be sampled and analyzed periodically to verify that the concentrations are within the established limits, that no concentration gradients exist, and that no precipitates have formed.

4. Evaluation

A calculation of the iodine removal effectiveness of the system is performed to establish the degree of iodine dose mitigation by the containment spray following the postulated accident. The mathematical model used for this calculation is designed to reflect the conclusions reached in the preceding steps of the review. The parameters determined include:

a. The maximum elemental iodine decontamination factor (DF) for the containment atmosphere achieved by the spray system, as determined from the equilibrium iodine partition coefficient. The DF is at present limited to a maximum value of 100 for sodium hydroxide and hydrazine systems and a value of 1000 for systems using appropriately buffered thiosulfate additive.

b. The removal of iodine from the containment atmosphere is considered a first-order removal process. The removal coefficient (lambda) for this process is determined for each of the sprayed regions of the containment, or, alternatively, an equivalent lambda for the entire free volume of the containment is determined. (See reference 4 for an acceptable model for the calculation of lambda.) The removal constant determined by this method is at present limited to a maximum value of 10 per hour for elemental iodine when used in conjunction with the instantaneous containment plate-out factor of 2 implied in Regulatory Guides 1.3 and 1.4.

IV. EVALUATION FINDINGS

If the AAB finds, on completion of the review, that the containment spray and spray additive system (if any) is effective for iodine removal, the following can be reported in the staff's safety evaluation report (SER):

"The concept upon which the proposed system is based has been demonstrated to be effective for iodine absorption and retention under post-accident conditions.

"The proposed system design is an acceptable application of this concept. The proposed pre-operational tests, post-operational testing and surveillance, and proposed limiting conditions of operation for the spray system provide adequate assurance that the iodine scrubbing function of the containment spray system will meet or exceed the effectiveness assumed in the accident evaluation."

In addition, the staff's evaluation of the iodine removal effectiveness of the containment spray system includes the following parameters, which are also used in the thyroid dose calculations of a postulated loss-of-coolant accident:

a. A first-order removal constant of λ_1 (1/hr) for elemental iodine, λ_2 (1/hr) for organic iodine, and λ_3 (1/hr) for particulate iodine.

b. An effective volume for the spray of V (ft^3).

c. A maximum decontamination factor (DF) for elemental iodine.

V. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

2. 10 CFR Part 50, Appendix A, General Design Criteria 41, 42, and 43.

3. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part IV. Calculation of Iodine Partition Coefficients," ORNL-TM-2412 Part IV, Oak Ridge National Laboratory (1970).

4. A. K. Postma and W. F. Pasedag, "Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containments," WASH-1329, U.S. Atomic Energy Commission, June 1974.

5. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

6. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2.

7. General references for this SRP are listed in the following sections of the AAB "Bibliography for Filters, Sprays, and Iodine," maintained in the AAB office:

   Section IV, Design Considerations of Reactor Containment Spray Systems
   Section V, General Fission Product Behavior
   Section VI, Hydrazine Sprays
   Section IX, Iodine Species and Physical Properties
   Section XIII, Properties of Iodine-Water Systems
   Section XV, Spray Removal Systems
   Section XVI, Progress Reports - BMI
   Section XVII, Progress Reports - ORNL
   Section XVIII, Progress Reports - PNL
   Section XIX, Conferences

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.5.3 FISSION PRODUCT CONTROL SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The descriptions of the primary and secondary containments and of the containment penetrations are reviewed to (a) provide a basis for developing the mathematical model for design basis accident (DBA) dose computations, (b) verify that the values of certain key parameters are within pre-established limits, (c) confirm the applicability of important modeling assumptions, and (d) verify the functional capability of the secondary containment ventilation systems. The parameters which must be established and the systems whose functions must be reviewed or understood by the reviewer are outlined below. Many of these areas are the responsibility of other branches and are reviewed by the AAB to provide a general knowledge of the containment systems and their operation following a loss-of-coolant accident (LOCA).

1. Primary Containment Design

The following areas are reviewed:

a. Containment type, e.g., free-standing steel shell, reinforced steel-lined concrete, as described in Sections 3.8.1 or 3.8.2 and 6.2.1 of the applicant's safety analysis report (SAR). The containment type should be known so that the reviewer understands the degree to which positive pressure periods in the secondary containment may be affected by design basis accident heat loads on the primary containment. The need for containment vacuum relief valves may also be indicated by containment type. The CSB has responsibility for evaluating the pressure transient of the primary containment and for reviewing the vacuum relief valve design, where appropriate.

b. Pressure suppression devices, e.g., sprays, subatmospheric operation, suppression pool, ice condenser, as described in Sections 6.2.1 and 6.2.2 of the SAR. The existence and operation of pressure suppression devices should be determined since their existence and performance control peak containment pressure and containment leakage rate. The CSB is responsible for evaluating the peak containment pressure and containment leakage rate.

c. Fission product cleanup, e.g., sprays with chemical additives, internal ESF filter systems, ice condenser, as described in SAR Sections 6.5.1, 6.5.2, and 6.5.4. Knowledge of these systems is necessary for modeling the system for dose calculations.

d. General design characteristics, e.g., design leakage rate, free volume, fan flow rate across operating floor (ice condenser), peak containment pressure, time into a design basis accident for initiation and rate of hydrogen purge through the containment purge system when this is exhausted into the secondary containment system. (See SAR Sections 9.4, 6.2.5, and Tables 6-1 through 6-4 as appropriate.) Some of these parameters are required for the dose calculations; others are required in establishing the model to be used.

Hydrogen purge time and purge rate are interface areas with the CSB, as detailed in Section III of this standard review plan (SRP). Verification of other design data may require interfaces with the CSB, the APCSB, or the SEB as noted in Section III.

2. Secondary Containment Design

The following areas are reviewed:

a. Containment type, e.g., metal siding, reinforced concrete. (See SAR Section 3.8.4.) The type of secondary containment structure may indicate the effect of varying wind speed (possible exfiltration) and the probable leak tightness of the secondary containment. The SEB has responsibility for reviewing the structural design of the containment. Leak tightness and leakage testing are the responsibility of the CSB.

b. Physical layout, e.g., volume completely surrounding primary containment, auxiliary building regions treated, main steam tunnel treated (in boiling water reactors), main steam line leakage control system provided (BWR's), drawings or plan views defining secondary containment boundary, clarification of which regions are treated by cleanup systems. (See SAR Sections 6.2.3, 6.5.3, and 9.3.) Knowledge of what regions are treated as part of the secondary containment is essential to establish the mathematical model for dose calculations.

c. Fission product removal or hol up system design, e.g., regions treated by each system, piping and instrumentation rawings of each system and its operation, fan flow rates, recirculation rate, filter locations and efficiencies, system redundancy, actuation signals, time to reduce region pressures below atmospheric, potential for exfiltration under varying wind conditions, filter cooling capability, placement of ducting. (See SAR Sections 6.2.3, 6.5.1, and 6.5.3.) The reviewer is responsible for determining that each system can perform its functions as claimed to reduce fission produ t release following a postulated design basis accident. Information on fission product removal systems may be provided by other AAB reviewers or by the ETSB (filter system). Knowledge of these systems is necessary for modeling the system for the dose calculation. The CSB has responsibility for evaluating the pressure transient in the secondary containment to verify secondary containment region pressures following a design basis accident and for reviewing bypass leakage paths. The MEB has responsibility for evaluating the structural design of the ventilation system.

d. General design characteristics, e.g., negative pressure maintenance during normal operation, free volumes of regions, and leakage rates. (See SAR Sections 6.2.3, 6.5.3, and 9.4.) Knowledge of these parameters is also necessary for developing the mathematical model. The APCSB has responsibility for evaluating systems which maintain negative pressure in secondary containment regions during normal operation. The CSB has responsibility for evaluating secondary containment leakage rates.

II. ACCEPTANCE CRITERIA

In establishing the model to be used for estimating the radiological consequences of a design basis loss-of-coolant accident and determining the acceptability of the secondary containment ventilation systems, the following acceptance criteria are used by the AAB.

1. Primary Containment

Primary containment design leakage rates for which credit is given should not be less than 0.1%/day due to difficulties in measuring lower leakage rates. No upper limit has been established for this parameter except, where feasible (e.g., where very high leakage rates could be allowed), leakage rates should be reduced to obtain computed doses from design basis accidents that are well within 10 CFR Part 100 guidelines (e.g., 150 rem thyroid).

2. Secondary Containment

To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and its volume should be held at a minimum negative pressure differential of 0.25 inch (water), when compared with adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to assure site boundary exposures less than those calculated for the design basis accidents even if exfiltration occurs. (For a very leaky secondary containment, the CSB requests the AAB to perform a special exfiltration analysis.) Metal siding structures are acceptable if they can meet all leakage test requirements under varying wind conditions.

Other criteria include specifications for:

a. Mixing test for any recirculation system installed.

b. Intake and return headers on recirculation systems. These should be placed as far away from each other as is practical. The return header should provide a wide distribution over the confinement volume. The purpose of this placement is to assure some degree of mixing of the return flow in the secondary containment volume before it is again drawn into the system intake. With judicious placement, up to 50% mixing may be assumed, but a claim for greater than 50% mixing must be supported by adequate test data or a testing program which the applicant proposes to follow, once the system is built, to prove the claim. Spacing between intake and return headers is reviewed on a case-by-case basis. Adjustments in the mixing fraction to less than 50% may be indicated by some designs. Past practice has been to allow mixing in 50% of the volume between (and within 10 or 20 feet of) the inlet and outlet headers if both have distributed openings or if one has distributed openings and the other is at the top of the containment.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The purpose of the review of a dual containment system is to define a model to be used in DBA (specifically, the LOCA) dose calculations, to check that the values of certain key parameters are within pre-established limits, to confirm the correctness of important modeling assumptions, and to verify the functional capability of the secondary containment ventilation systems. Specific system design areas may not be reviewed in detail (filters, sprays, leakage rates, etc.), but the reviewer is responsible for reviewing all related ventilation systems and for selecting a representative dose model for DBA calculations. Therefore, the reviewer covers various areas (containment design, positive pressure periods, filters, etc.) for continuity rather than detail. Digital computer codes (Ref. 1) are used to perform the dose calculations.

All statements referring to "operation" in the following discussion mean operation following a postulated design basis LOCA. Normal operation is so identified.

Where a review area is not the primary responsibility of the AAB, it is assumed that appropriate acceptance criteria are used by the responsible branch and when these criteria are not met, the inadequacies are identified by that branch and the AAB is informed so that appropriate modifications of the model may be made. These areas include:

- Primary containment leakage rate, bypass leakage, and testing of these (CSB).

- Hydrogen purge systems (CSB).

- Secondary containment vacuum maintenance systems (normal operation) (APCSB).

- Secondary containment pressure response (post-accident) (CSB).

- Containment isolation (CSB).

- Structural design of containments (SEB) and systems (MEB).

- Engineered safety feature filter systems (ETSB).

1. Primary Containment Design

   a. The primary containment design is studied to familiarize the reviewer with the overall construction (free-standing steel shell; reinforced, steel-lined concrete) and anticipated performance capability of the primary containment (subatmospheric or ice condenser containment, leakage rate limits, etc.). Certain parameters, such as design leakage rate, containment free volume, the existence of internal fission product cleanup systems, should be noted for later use. (See example of worksheet, Table 6.5.3-1.) The performance capability of the internal fission product cleanup systems (if any) should be verified. (See SAR Sections 6.5.1, 6.5.2, and 6.5.4.)

   b. The curve indicating containment pressure versus time following the accident should be studied. Historically, pressurized water reactor (PWR) containment design leakage rates have been reduced by a factor of two after one day. (See Ref. 3.) If the long-term pressure transient shows the containment pressure is not reduced to one-half within 24 hours, the CSB confirms the validity of the leakage rate before it is used in the dose analysis. On BWR containment systems (including MARK III), the containment design leakage rate is to be used for all time periods following the accident (See Ref. 2) unless advised otherwise by the CSB. For those containments designed to reach subatmospheric pressure at some time less than 30 days after the accident, the CSB verifies the time required to reach subatmospheric pressure. Verification is by buckslip, a copy of which is retained in the AAB site analyst's workbook.

      Performance of the hydrogen purge system is reviewed for the purpose of modeling the purge dose calculation. On some systems, the hydrogen purge lines are vented to the recirculation return line of the secondary containment ventilation system. Initiation time for the hydrogen purge and the purge rate are obtained from the CSB.

2. Secondary Containment Design

   a. The secondary containment design is reviewed to determine how it should be modeled for the dose calculations. The ability of the structures to withstand the safe shutdown earthquake or to meet the tornado criteria is the responsibility of the SEB, but the reviewer checks the applicant's SAR to determine what criteria the structures are designed to meet. The reviewer also ascertains that the applicant has considered the question of potential exfiltration from regions of the secondary containment under varying wind conditions, especially if the structure has a leakage rate greater than 100%/day. The anticipated leakage rate from each region is noted (see example of worksheet, Table 6.5.3-2), and special attention paid to accuracy of the proposed leakage testing if the leakage rates are less than 10% per day. (No facility reviewed to date has a proposed secondary containment leakage rate of less than 10% per day. Experience indicates that 10% per day may be difficult to achieve in actual practice.)

b. The boundary of the secondary containment is determined; it should completely enclose the primary containment. Usually, the secondary containment boundary is composed of more than one region, e.g., a shield building (concrete) or enclosure building (metal siding) around the primary containment and all or parts (emergency core cooling pump rooms, etc.) of the auxiliary building. (See Figures 6.5.3-1 through 3 for example diagrams.) These regions may be treated by one or more ventilation systems as shown on Figures 6.5.3-1 and 6.5.3-2.

c. For PWR containments and BWR MARK III containments, the annular region between the shield building or enclosure building and the primary containment may be held at a negative pressure relative to adjacent areas by a vacuum exhaust system during normal operation. Since this system is used during normal operation, it may appear in the SAR under auxiliary systems. The exhaust system may also treat the auxiliary building regions which are part of the secondary containment; but if these regions are maintained at a negative pressure during normal operation, it is most likely done with the auxiliary building ventilation system. Both the vacuum exhaust and auxiliary building ventilation systems fall under the purview of the APCSB. The systems' ability to maintain negative pressures of sufficient margin under varying wind conditions and operational modes prior to a design basis accident is verified by the APCSB. The AAB reviewer is responsible for reviewing the design of systems maintaining negative pressure following a design basis accident. If an adequate negative differential pressure margin (0.25 inch water gauge) is maintained for all times into the accident (from the time the accident happens), then no positive pressure time period need be assumed in the dose model. All positive pressure periods in the secondary containment regions are treated as direct outleakage periods following an accident, and no credit is given for filters or recirculation systems. The CSB verifies the positive pressure periods. The large reactor buildings around older BWR containments are usually maintained at a negative pressure during normal operation, and the dose model used for these cases has not assumed any positive pressure period.

d. The exhaust systems used to maintain the negative pressure differential following the accident should be sized to meet the negative pressure criterion for the inleakage rate and the conservatively calculated heat load for the regions treated by each, and analyses to this effect should be presented by the applicant. The pressure response analyses are reviewed by the CSB. The functional capability of the filter design associated with the exhaust system is reviewed by the ETSB under Standard Review Plan (SRP) 6.5.1. The reviewer should establish that the ESF filter systems are being reviewed by the ETSB. The exhaust systems may be one of several designs. Common designs are:

(1) Straight exhaust through charcoal and HEPA filters. Primary containment leakage to these regions is assumed to go directly to the filter with no mixing or holdup in the region being filtered. (See Figures 6.5.3-3 and -4.)

(2) Recirculation system with split in flow (some exhausted through filters and some recirculated to the region being treated). Primary containment leakage to the region being treated is assumed to be directly to the intake of the recirculation fan. There, a fraction of it (the ratio of exhaust to total flow) is exhausted through the filters; the balance is then assumed to return to the region being treated. The placement of the system intake and return headers is examined to determine that return flow from the fans does not have a direct path to the intake again. (See Figures 6.5.3-5 and -6.) Credit for mixing in 50% of the region is given if the header placement is satisfactory.

(3) Other variations on the recirculation system are (a) filters in the recirculation line, (b) filters in both the recirculation line and the exhaust line, and (c) high exhaust flow to reduce the negative pressure to several inches water gauge, and then no exhaust with recirculation only for some time period.

The sizing of the system fans for the volumes they are maintaining at a negative pressure may be critical in determining the ratio of exhaust flow to recirculation flow. Past history shows secondary containment structures are considerably more leaky than applicants anticipated (2 to 5 times as great as anticipated), and fan exhaust flows have been increased after testing to account for this. (When identical flow rates are predicted for two volumes which differ by a factor of 10 or more, it is difficult to believe that the negative pressure differential will be the same for both volumes.) The flow rates, negative pressure differential, and volumes are noted and the appropriate AAB reviewer and CSB reviewer (pressure response only) consulted for verification before performing dose calculations.

The systems should be reviewed to determine volumes treated, system operation, fan flow rates, and filter efficiencies. All the applicant's claims should be verified by appropriate staff members as noted on Table 6.5.3-2. Leakage fractions from the primary containment to each volume should be identified and stated in the technical specifications. Completeness of information, adequacy of technical specifications and testing methods, and the adequacy and maintenance of the integrity of the secondary containment negative pressure considering failures of non-seismic piping or ducting are verified by the CSB.

IV. EVALUATION FINDINGS

The reviewer defines a dose model for the LOCA dose calculations and prepares a table of all the data for the primary and secondary containments to be used in the calculation. The recommended form for tabulation is given in Table 6.5.3-3. This table should include the information needed to model hydrogen purge dose calculations. In addition, the reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

"The fission product control systems include all structures, ducting, valves, and fans which are used to control leakage of fission products following a postulated design basis accident. The scope of review of these systems included piping and instrumentation diagrams and general arrangement diagrams showing flow in the fission product control systems and areas treated by each system, and descriptive information about each system. The review has included the applicant's proposed design criteria and design bases for each system and the applicant's analysis of the adequacy of those criteria and bases. The applicant's analyses of the manner in which the designs of the fission product control systems conform to the proposed design criteria have also been reviewed.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the fission product control systems and necessary auxiliary supporting systems to the Commission's regulations, and to Regulatory Guide 1.3 (or 1.4), staff technical positions, and industry standards.

"The staff concludes that the designs of the fission product control systems conform to all applicable regulations, guides, staff positions, and industry standards, and are acceptable."

V. REFERENCES

1. Computer codes are currently under development. Documentation will be published as a NUREG report.

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2.

Table 6.5.3-1

Example Worksheet:
Primary Containment Information

Data Description	Parameter Value	Staff Verification
Type of Structure		SEB
Primary Containment Design Leak Rate		CSB
Bypass Leakage Fraction to Volumes		CSB
1.		
2.		
3.		
Primary Containment Free Volume		CSB
Primary Containment Subatmospheric Operation		CSB
Primary Containment Internal Fission Product Removal Systems:		AAB
Ice Condenser		
Spray System		
Filter System		
Other		
H_2 Purge Mode (e.g., direct, to recirculation systems, to annulus)		CSB
Purge Initiation Time		
Purge Rate		
Primary Containment Purge:		CSB
Used During Normal Operation		
Valve Arrangement		

Table 6.5.3-2

Example Worksheet:
Secondary Containment Information

Data Description	Parameter Value	Staff Verification
For each Secondary Containment Region:		
Type of Structure		SEB
Free Volume		CSB
Mixing Fraction		AAB
Design Leak Rate		CSB
Annulus Width (where applicable)		CSB
For each Ventilation System:		
Total Recirculation Flow		AAB
Exhaust Flow		AAB
Filter Placement		AAB
Filter Efficiencies		ETSB
Header Placement		AAB
Time Sequence for Operation Following an Accident or		CSB
Operation of System Prior to an Accident if Used During Normal Operation		APCSB

Table 6.5.3-3

Evaluation Findings

Primary Containment Leak Rate

Primary Containment Free Volume

Primary Containment Internal Fission Product Removal System

Primary Containment Subatmospheric Operation

Primary Containment Leakage Paths

Secondary Containment Free Volume

Secondary Containment Total System Flow

Secondary Containment Exhaust Flow

Secondary Containment Mixing Fraction

Secondary Containment Filter Efficiencies

Time Sequence for Operation of Fission Product Removal or Holdup Systems in Total Containment System Following a Postulated Accident

H_2 Purge:

- Initiation Time
- Purge Rate
- Purge Model

Dual Containment

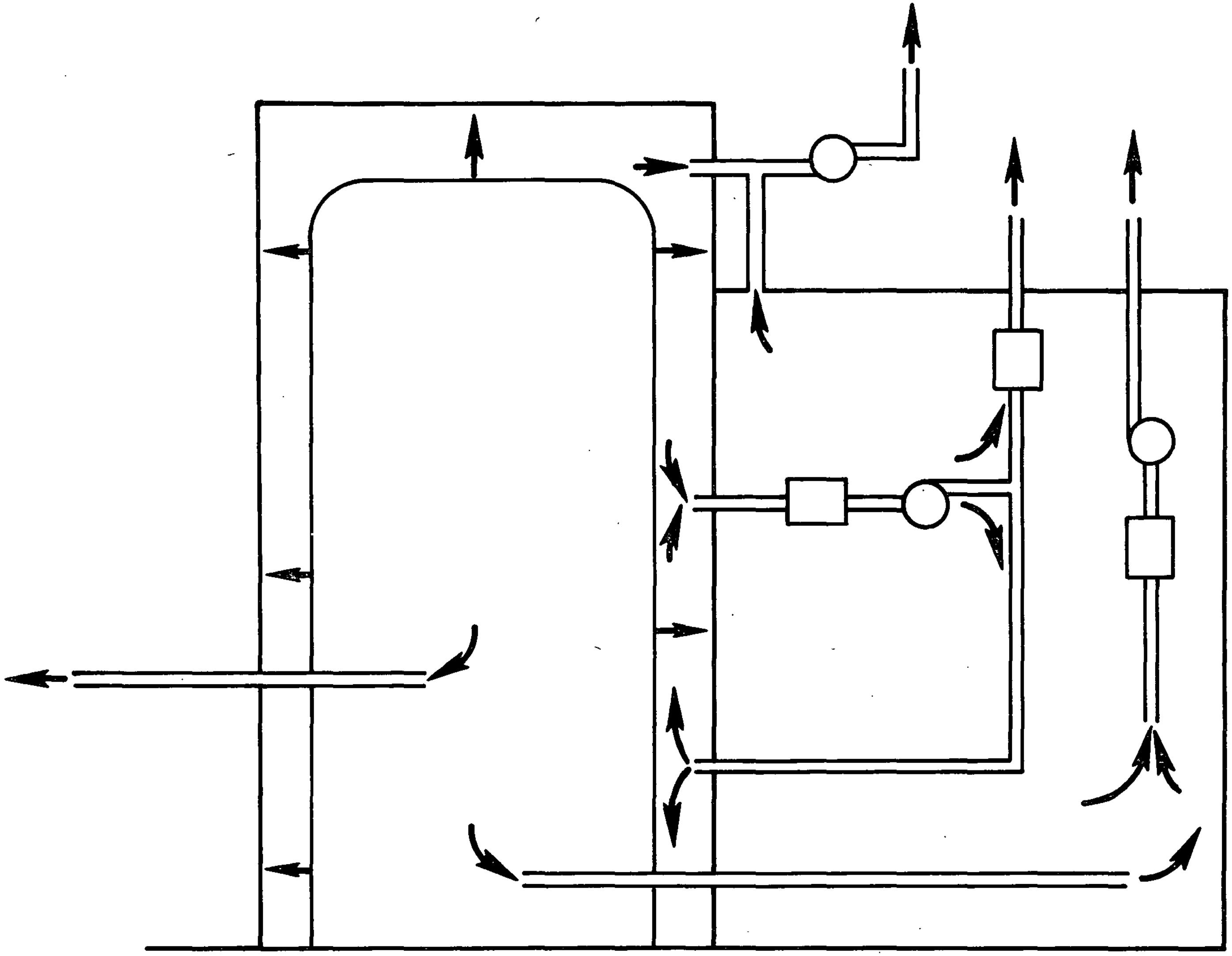


FIGURE 6.5.3-1

PWR Dual Containment

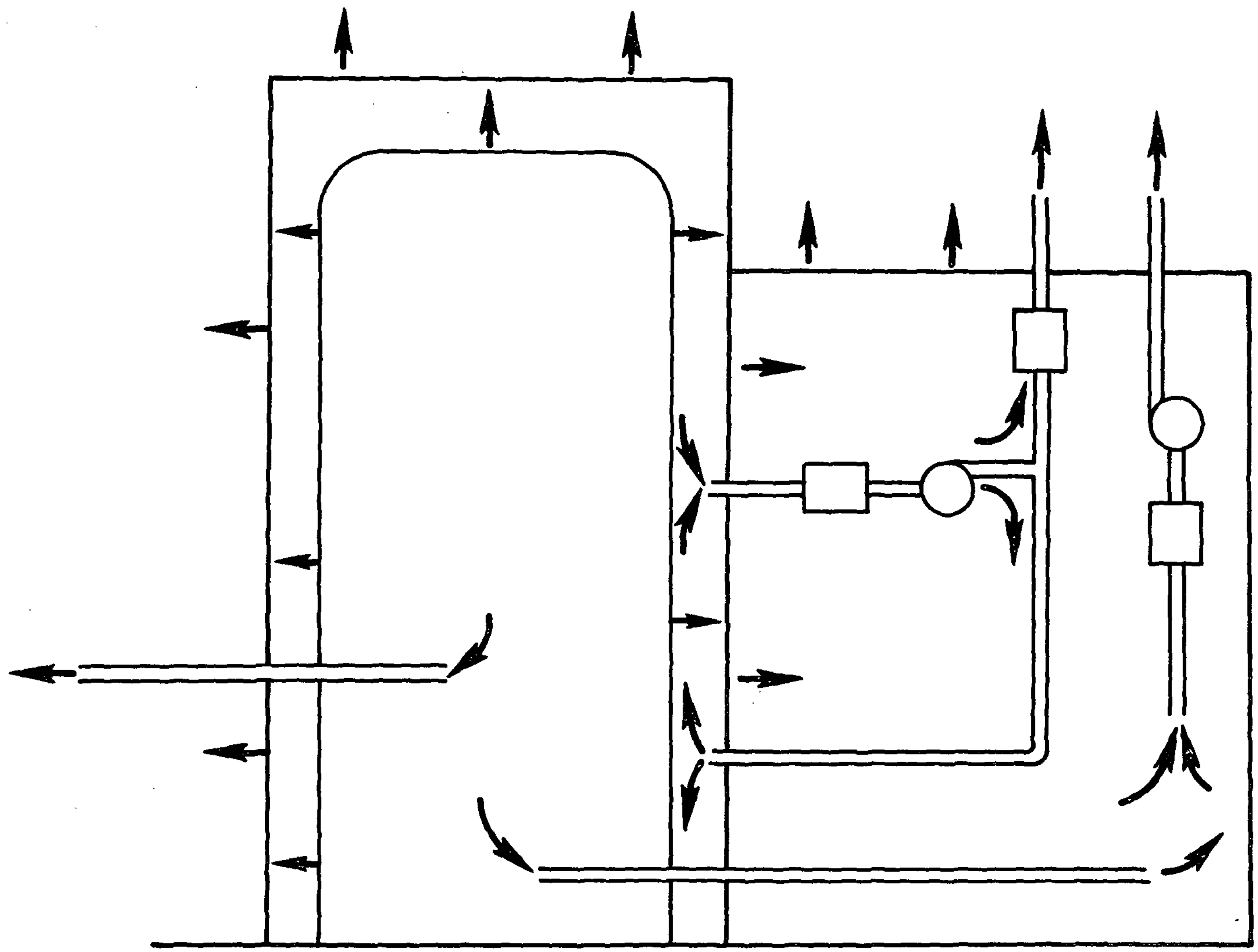
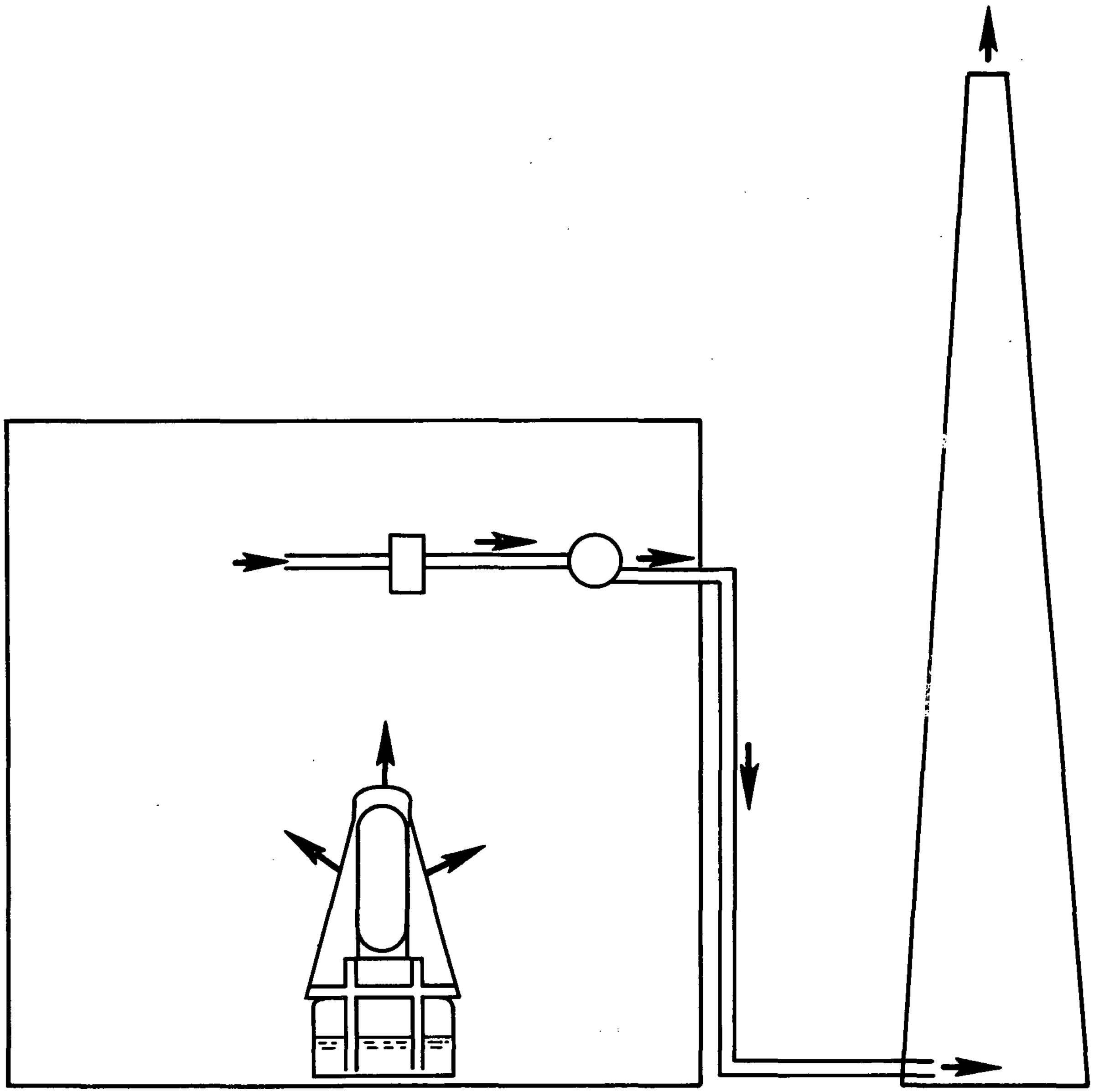
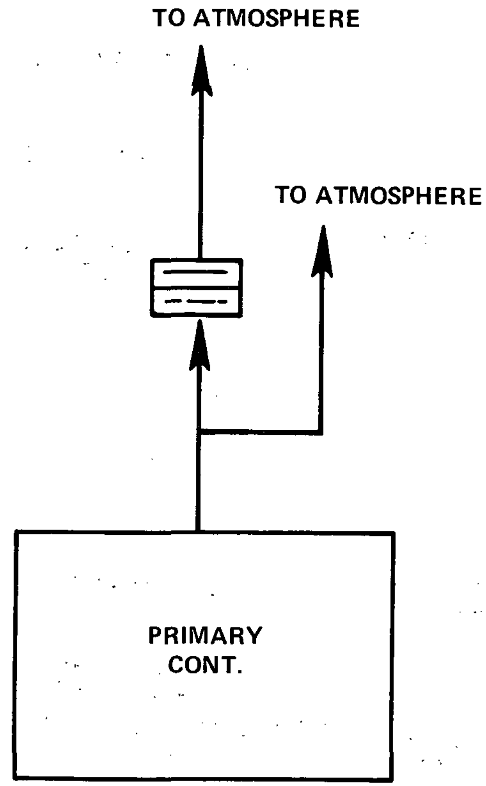


FIGURE 6.5.3-2



BWR With Stack

FIGURE 6.5.3-3



Standard Model

FIGURE 6.5.3-4

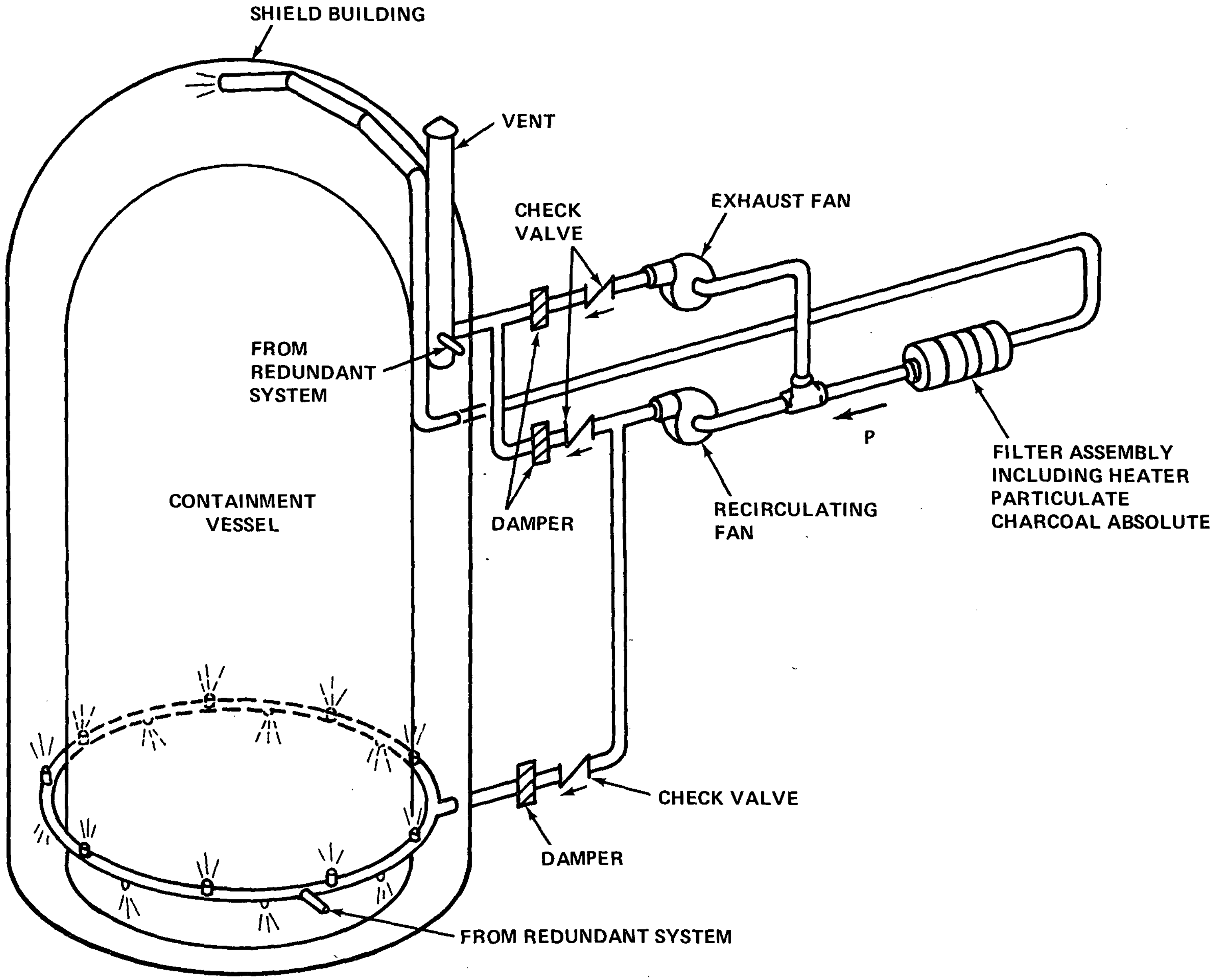
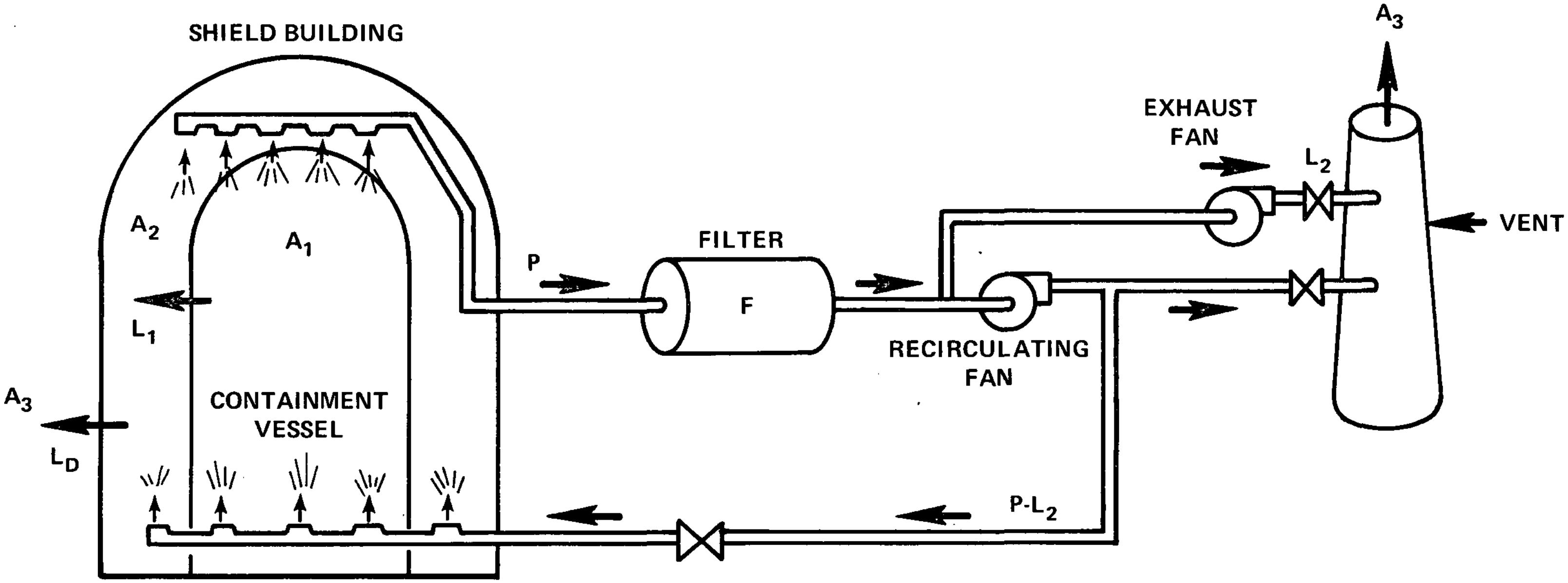

SHIELD BUILDING
VENT
CHECK
VALVE
EXHAUST FAN
FROM
REDUNDANT
SYSTEM
P
FILTER ASSEMBLY
INCLUDING HEATER
PARTICULATE
CHARCOAL ABSOLUTE
CONTAINMENT
VESSEL
DAMPER
RECIRCULATING
FAN
CHECK VALVE
DAMPER
FROM REDUNDANT SYSTEM

FIGURE 6.5.3-5



System Schematic

FIGURE 6.5.3-6

11/24/75

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) are reviewed:

1. Fission Product Removal Requirement for the Ice Condenser System

   Sections of the SAR related to accident analysis, dose calculations, and fission product removal and control are reviewed to establish whether fission product scrubbing of the containment atmosphere is required for mitigation of offsite doses following a postulated accident. This review usually covers SAR Sections 6.2, 6.5.4 and 15.2.X.X.

2. Design Bases

   The design bases for the fission product removal function of the ice condenser system are reviewed to determine whether they are consistent with the requirements placed upon this system by the assumptions made in the accident evaluations of SAR Chapter 15.

3. System Design

   The descriptive information concerning the portions of the ice condenser system design important to its fission product removal function is reviewed to familiarize the reviewer with the design and post-accident functioning of the ice condenser. This includes:

   a. The basic design concept, the systems, subsystems, and support systems required to carry out the fission product cleanup function of the ice condenser.

   b. Descriptive information and figures from SAR Section 6.2, as related to:

      (1) The time required to establish a steady flow of predictable magnitude of an air-steam-iodine mixture through the ice beds.

      (2) The time of melt-out of the ice beds.

4. Testing and Inspections

The details of the applicant's proposed preoperational test to be performed for system verification and operational tests and inspections to verify the continued status of readiness of the iodine removal capacity of the ice condenser systems are reviewed.

5. Technical Specifications

At the operating license stage, Sections 3 and 4 of SAR Chapter 16 are reviewed to establish surveillance requirements for the sodium hydroxide concentrations in the ice.

II. Acceptance Criteria

The acceptance criteria for the fission product cleanup function of the ice condenser system are:

1. Ice Alkalinity

The ice condenser system is acceptable for elemental iodine removal if the ice contains a quantity of sodium hydroxide sufficient to assure that the water solution from ice melting has a pH of at least 9.0.

2. *Duration of Iodine Scrubbing Function*

The ice condenser is assumed to be effective for iodine removal only during that period following an assumed accident when a steady flow of predictable magnitude of the air-steam-iodine mixture has been established. At present, steady flow is assumed to commence with the operation of the post-accident mixing fans.

3. Tests and Inspections

Preoperational and inservice tests should assure that the proper ice alkalinity is maintained. Other inspections associated with the pressure suppression function will assure the adequacy of ice quantity and geometry.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The first step in the review of ice condenser fission product removal is to determine whether the ice condenser system is used for accident dose mitigation purposes. Chapter 15 of the SAR is reviewed to determine whether a dose reduction credit was assumed for the ice condenser. If no fission product removal credit is assumed in the accident analysis, no further review is required. (The heat removal aspects of the system are reviewed by the CSB.)

If the ice condenser system is used for iodine removal, the iodine removal effectiveness of the ice condenser system is reviewed. The review includes the following:

1. System Design and Evaluation

a. Ice Chemistry

The chemistry of the ice is usually modified to include sodium hydroxide in order to improve the iodine scrubbing effectiveness of the ice condenser system. If the concentration of the sodium hydroxide is such that the ice, after melting but prior to any dilution meets the pH requirements stated in the acceptance criteria of this review plan, the system is considered effective for elemental iodine removal. For ice condenser systems similar to those of the D. C. Cook and Sequoyah plants (with a steady-state flow rate of approximately 40,000 cfm) an efficiency of 30% per pass for elemental iodine is assigned. The system is considered ineffective for organic and particulate iodine removal.

b. Duration of Iodine Scrubbing Function

It is not feasible to specify the exact time of the fission product release following a postulated loss-of-coolant accident. In addition, the flow rates and air/steam fractions of the flow through the ice condenser vary significantly during and immediately following the accident. For dose calculation purposes, therefore, the following conservative assumptions are made:

(1) The iodine removal effectiveness of the ice condenser commences with the establishment of a steady-state air-steam flow by the air-steam return fans. (A single failure of one of the fans is assumed.)

(2) The initial concentration of iodine is assumed uniform throughout the entire containment. (This assumption may be modified in the future.)

(3) The effectiveness of the ice condenser as an iodine removal system is assumed to cease with the melt-out of the first ice bed.

c. Evaluation

The air-steam fan flow rate is used with the above assumptions in modeling fission product behavior for the loss-of-coolant accident (see Appendix A to Standard Review Plan 15.6.5).

2. Technical Specifications

The technical specifications are reviewed to assure that they require periodic inspection and sampling of the ice in order to confirm the continued state of readiness of the system, i.e., the system meets the chemistry requirements specified in the acceptance criteria of this review plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report:

"We have reviewed the fission product scrubbing function of the ice condenser and conclude that the addition of sodium hydroxide to the ice, as proposed by the applicant, will reduce the elemental iodine concentration of the steam-air mixture flowing through the ice beds following a loss-of-coolant accident. We estimate an elemental iodine removal efficiency of _______% per pass during the time period starting at________minutes after the accident and ending at__________minutes. The applicant's proposed program for preoperational and periodic surveillance tests will assure a continued state of readiness for the ice condenser iodine removal function."

V. REFERENCES

References for this standard review plan are listed in the following sections of the bibliography for filters, sprays, and iodine maintained in the AAB office:

Section V, General Fission Product Behavior.

Section VIII, Iodine Removal by the Ice Condenser.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

General Design Criterion 36, "Inspection of Emergency Core Cooling System;" Criterion 39, "Inspection of Containment Heat Removal System;" Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems;" and Criterion 45, "Inspection of Cooling Water System," of Appendix A to 10 CFR Part 50 require that the subject systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability.

The following areas relating to the inservice inspection (ISI) program for AEC Quality Group B and C (ASME Boiler and Pressure Vessel Code, Section III, Code Class 2 and 3) components are reviewed:

1. Components Subject to Examination
The descriptive information in the applicant's safety analysis report (SAR) is reviewed to establish that all the ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Subarticle NA-2140, Class 2 and Class 3 components are included in the ISI program. The Reactor Systems Branch verifies that the system classifications as Code Class 2 and 3 agree with Subarticle NA-2110 of Section III and with the definitions of the general design criteria.

2. Accessibility
The descriptive information, including drawings, is reviewed by the Materials Engineering Branch to establish that the Code Section XI, Subarticle IWA-1500, provisions for system accessibility are included in the applicant's layout and design of these systems.

3. Examination Techniques and Procedures
The required examination techniques and procedures, including the requirements of Subarticles IWC-2600 and IWD-2600 of Section XI, are reviewed for their conformance to Subarticle IWA-2200 of Section XI of the Code.

4. Inspection Intervals

The required examinations and inspections listed in the SAR are reviewed and compared to Table IWC-2600 and Subarticle IWD-2600 of Section XI to verify that they will be performed within the designated inspection interval and comply with subarticles IWC-2400 and IWD-2400 of Section XI to verify that they will be performed within the designated inspection interval and comply with Subarticles IWC-2400 and IWD-2400 of Section XI.

5. Examination Categories and Requirements

The technical specifications are compared to Table IWC-2520 and Subarticle IWD-2600 of Section XI to verify that the examination categories and ISI requirements of each category are in agreement.

6. Evaluation of Examination Results

The information concerning repair procedures is reviewed for compliance with Articles IWC-4000 and IWD-4000 of Section XI. Because Articles IWC-3000 and IWD-3000 are still in course of preparation, as an interim step, evaluation of examination results for Class 2 and 3 components is reviewed for compliance with Article IWB-3000 for Class 1 components.

7. System Pressure Tests

The pressure test program is reviewed for compliance with Subarticles IWC-5200 and IWD-5200 of Section XI to establish that leakage and signs of structural distress are inspected for on a periodic basis. In addition, for Class 2 components, a review is performed to determine that all applicable systems and components are pressure tested at acceptable combinations of temperature and pressure.

8. Augmented ISI to Protect Against Postulated Piping Failures

The augmented inservice inspection program to provide assurance against postulated piping failures of high energy fluid systems between containment isolation valves is reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Components Subject to Inspection

The applicant's definition of Code Class 2 and 3 components and systems subjects to an ISI program is acceptable if in agreement with the definitions of Code Section III, Subarticle NA-2140, and Section XI, Table IWC-2600 and Subarticle IWD-2600. The interpretation of code classifications by the applicant is subject to review by the Reactor Systems Branch for compliance with safety criteria pertaining to component classification. (Refer to NA-2110 of Section III.)

2. Accessibility

The design and arrangement of Class 2 and 3 systems are acceptable if the applicant includes allowances for adequate clearances to conduct the examinations specified in IWC-2600 and IWD-2600 at the frequency specified by IWC-2400 and IWD-2400. Special design considerations are given to those systems that are intended to be examined during normal reactor operation.

3. Examinatin Techniques and Procedures
Since acceptance criteria for the required examination techniques and procedures are identical to those for the reactor coolant pressure boundary (RCPB), Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing" (Ref. 3), Section II.3, "Examination Techniques and Procedures," is incorporated by reference in this plan.

4. Inspection Intervals
The inservice inspection program schedule given in the SAR is acceptable if the required examinations are completed during each ten-year interval, hereafter designated as the inspection interval, and as required by Subarticles IWC-2400 and IWD-2400 of Section XI.

5. Examination Categories and Requirements
The ISI examination categories and requirements for Class 2 and 3 components established by the SAR are acceptable if in agreement with the following criteria of Section XI: IWC-2520, IWC-2600, and IWD-2600. The areas subject to examination and the extent of examination for the Class 2 components shall comply with the requirements specified under the examination categories of Table IWC-2520.

6. Evaluation of Examinations Results
Articles IWC-3000 and IWD-3000 of Section XI concerning evaluation of examination results are still in preparation. The applicant's evaluation of examination results should be consistent with that for Code Class I components. Thus, Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing," Section II.6, "Evaluation of Examination Results," is considered acceptable (Ref. 3).

7. System Pressure Tests
The SAR program for Class 2 and 3 system pressure testing is acceptable if it meets the following criteria of IWC-5000 and IWD-5000 of Section XI:

a. Class 2 Systems

(1) The system hydrostatic test pressure shall be at least 1.25 times the system design pressure (P_D) and conducted at a test temperature not less than 100°F except as may be required to meet the test temperature requirements of IWA-5230.

(2) When system hydrostatic testing is required to be conducted at temperatures above 100°F in order to meet the fracture toughness criteria applicable to ferritic materials of which the system components are constructed, the test pressure may be reduced in accordance with the table given in IWC-5220.

(3) For components that are not required to function during reactor operation, the system test pressure shall not be less than 100 percent of the pressure developed during the conduct of a periodic system performance test.

(4) Open-ended portions of a system (e.g., suction line from a storage tank, or discharge line of a containment spray header) extending to the first shutoff valve may be exempted from the test requirements of IWC-2510.

b. Class 3 Systems

(1) The system test pressure shall be at least 1.10 times the system design pressure.

(2) Open-ended portions of a system (e.g., suction line from a storage tank) extending to the first shutoff valve may be exempted from the test requirements of IWD-5200.

c. Storage Tanks

The nominal hydrostatic pressure developed with the tank filled to its design capacity shall be acceptable as the system test pressure.

8. Augmented ISI to Protect Against Postulated Piping Failures

High energy fluid system piping between containment isolation valves should receive an augmented ISI as follows:

a. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the Code, Section XI, Division 1.

b. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval (IWA-2400, ASME Code Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

c. For those portions of high energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures.

d. The areas subject to examination should be defined in accordance with Examination Categories C-F and C-G for Class 2 piping welds in Tables IWC-2520.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review the following review procedure is followed:

1. Components Subject to Inspection

The applicant's components and system classifications under Class 2 and 3 are reviewed for agreement with Section II.1 of this plan. The interpretation of code classifications is the responsibility of the Reactor Systems Branch, should a discrepancy occur between the SAR and Section II.1.

The applicant's classification of Class 3 systems is reviewed for agreement with Section II.1 of this plan. Any safety-related, fluid-carrying components not included in Class 1 or Class 2 and not a part of the containment structure are included in Class 3.

2. Accessibility

The design and arrangement of Class 2 and 3 systems are reviewed in terms of accessibility for ISI to establish that the design meets the requirements of Section II.2 of this plan. No remote inspection program is required for Code Class 2 or 3 components.

3. Examination Techniques and Procedures

The reviewer verifies that the examination techniques as described by the SAR are the same as those specified in Section II.3 of this plan.

4. Inspection Intervals

The inservice inspection program for Class 2 and 3 components in the plant technical specifications is reviewed to establish that each area and component in the program is inspected on a schedule in agreement with Section II.4 of this plan.

5. Examination Categories and Requirements

The examination categories and parallel inspection requirements described or tabulated in the technical specifications are reviewed to establish that they are in agreement with Section II.5 of this plan. The technical specification table or description is acceptable if it follows Tables IWC-2520 and IWC-2600 in terms of headings and arrangement for Class 2 components and IWD-2600 for Class 3 components.

6. Evaluation of Examination Results

The reviewer verifies that the evaluation of examination results described in the SAR is in accord with Section II.6 of this plan.

7. System Pressure Test

The system pressure test program is acceptable if it meets the criteria of Section II.7 of this plan.

8. Augmented ISI to Protect Against Postulated Piping Failures

The reviewer verifies that the augmented inservice inspection program as described in the SAR meets the acceptance criteria identified in Section II.8 of this plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with the requirements of this review plan and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. In addition, Code Class 2 and 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

"Examples of Code Class 2 systems are: residual heat removal systems, portions of chemical and volume control systems (in PWR plants), portions of control rod drive systems (in BWR Plants), and engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are: component cooling water systems, and portions of radwaste systems. All of these systems transport fluids. The applicant has stated that the design of Code Class 2 and 3 systems meets the requirements of ASME Code Section XI. Compliance with the inservice inspections required by the Code and staff technical positions constitutes and acceptable basis for satisfying applicable requirements of General Design Criteria 36, 39, 42, and 45."

V. REVERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System;" Criterion 39, "Inspection of Containment Heat Removal System;" Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems;" and Criterion 45, "Inspection of Cooling Water System."

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Sections NA-2140 and NA-2110, and Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water Cooled Plants," American Society of Mechanical Engineers.

3. Standard Review Plan 5.2.4, "RCPB Inservice Inspection and Testing."

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SECTION 6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

Direct cycle boiling water reactor (BWR) plants have redundant quick-acting isolation valves on each main steam line from the reactor to the turbine. In the event of a loss-of-coolant accident (LOCA), any leakage of contaminated steam through these valves is controlled by a leakage control system.

The review of the main steam isolation valve leakage control system (MSIVLCS) is applicable to direct cycle BWR plants. The review covers the entire leakage control system including the source of the sealing medium, if any, and pumps, valves, and piping to the points of connection or interface with the main steam supply system. Emphasis is placed on the components of the leakage control system that are required to remain functional during design basis LOCA conditions.

1. APCSB reviews the ability of the MSIVLCS and essential subsystems to function during and subsequent to postulated LOCA conditions, including the loss of offsite power. The system is reviewed to determine that:

   a. A malfunction or failure of an active component of the system, or loss of the source of sealing fluid, if any, will not reduce the functional performance of the system.

   b. The failure of non-seismic Category I equipment or components will not have an adverse effect on the system or components.

   c. The capability of the system to perform its intended safety function is maintained assuming a single active failure of a main steam isolation valve.

2. The APCSB also reviews the design of the leakage control system with respect to the following:

a. The capability of the system to withstand the effects of the safe shutdown earthquake, including the source of sealing medium, if any.

b. The capability of the system to control main steam isolation valve leakage and preserve containment integrity under design basis LOCA conditions, including loss of offsite power.

c. The compatibility of initiation means and controls of the system with loading requirements on the emergency electrical buses, operator reaction times, and with actuation times available in view of the specified isolation valve leakage limits.

d. The requirements for interlocks to prevent inadvertent system operation.

e. The capability of the system design to permit functional testing of components, controls, and actuation devices during power operations to the extent practicable and complete functional testing during plant shutdown.

f. The capability of the system and main steam supply system components to withstand effects resulting from the use of a sealing medium, if any, such as thermal stresses, pressures associated with flashing, and thermal deformations, so that the structural integrity of the main steam lines and isolation valves will not be affected and that any deformation of valve internals will not induce excessive leakage through the values.

g. The design provisions incorporated to prevent or treat main steam isolation valve stem packing leakage or other direct leakage.

h. The instrumentation and control features necessary to accomplish the system function, including isolation of components of the system in the event of malfunctions.

i. The use of applicable codes and standards and assignment of appropriate seismic and quality group classifications.

3. The applicant's proposed technical specifications are reviewed at the operating license stage as they relate to areas covered in this plan.

Secondary review evaluations are performed by other branches and the results used by APCSB to complete the overall evaluation of the system. The evaluations provided by other branches are as follows. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE). The MEB reviews the seismic qualification of components and confirms that components and piping are designed in accordance with applicable codes and standards. The RSB determines that the assigned seismic and quality group classifications for system components are acceptable.

The MTEB verifies that inservice inspection requirements are met for system components and, upon request, verifies the compatability of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection and testing of all electrical components (sensing, control, and power) required for proper operation.

II. ACCEPTANCE CRITERIA

Acceptability of the MSIVLCS, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for determining the acceptability of the MSIVLCS is the degree of similarity of the design with that of previously reviewed plants.

The design of the MSIVLCS is acceptable if the integrated system design is in accordance with the following criteria:

1. General Design Criterion 2, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, as related to structures housing the system and the system itself being capable of withstanding the effects of externally and internally generated missiles.

3. General Design Criterion 54, as related to the capability for leak detection, isolation, and performance testing for system piping penetrating containment.

4. Regulatory Guide 1.96, as related to the design of the MSIVLCS.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria, design bases, and preliminary design meet the acceptance criteria given in Section II. For the review of operation license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design. The OL review includes a determination that the content and intent of the technical specifications prepared by the applicant are in agreement with the requirements for system testing, minimum performance, and surveillance developed by the staff. The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. The information provided in the SAR pertaining to the design basis and design criteria, the system piping and instrumentation diagrams (P&IDs), and the system description are reviewed to determine that they clearly delineate the following:

    a. The method used to accomplish the main steam isolation valve leakage control function and the system components essential for operation in design basis LOCA conditions.

b. Essential components of the leakage control system are correctly identified and are isolable from any non-essential portions of the system. The PID's are reviewed to verify that they clearly indicate the physical divisions between such portions and indicate any design classification changes. System drawings are reviewed to see that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements for the isolation valves.

c. Essential components of the leakage control system, including the isolation valves separating any non-essential portions of the system, and the seal fluid source (if used) are classified seismic Category I and Quality Group A or B, as specified in Regulatory Guide 1.96. Component and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above classifications have been included, and that the P&IDs indicate points of design classification changes.

d. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary design provisions to accomplish the testing program.

2. The reviewer determines that the safety function of the MSIVLCS will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes. The reviewer uses engineering judgment, the results of failure modes and effects analyses, and the results of reviews performed under other review plans to determine that the failure of non-essential portions of the system or of other systems not designed to seismic Category I standards and located close to essential portions of the system, or of non-seismic Category I structures close to essential portions of the system, will not preclude operation of the essential portions of the MSIVLCS. Reference to SAR sections describing site features, the general arrangement and layout drawings, and the tabulation of seismic design classifications for systems and structures will be necessary. Statements in the SAR that the above conditions are met are acceptable.

3. If the leakage control system is one using a fluid sealing medium:

a. The system design is reviewed to determine that the quantity of sealing fluid needed for an effective seal of the valves has been provided. Independent analyses, using the pump performance curves in the SAR, are made to assure that the design and the location of the pump and components are such as to maintain the appropriate net positive suction head (NPSH) requirements and provide a continuous supply of sealing fluid during the full course of an accident.

b. The system design is reviewed to determine that effects resulting from the sealing fluid, such as thermal stresses, pressures associated with flashing, thermal deformations, and other effects will not effect the structural integrity of the steam lines or the main steam isolation valves, or lead to excessive leakage of the valves. This portion of the review is done on a case-by-case basis. Acceptability may be based on a comparative analysis of system parameters and capabilities to similarly designed systems previously found acceptable. The APCSB also accepts the system design if a statement in the SAR commits to performing calculations or functional testing to demonstrate that the above conditions are met.

4. The MSIVLCS is reviewed to verify that instrumentation, controls, and interlocks designed to standards appropriate for an engineered safety feature are provided to actuate the system in the event of a design basis LOCA, and to prevent inadvertent actuation. Interlocks to prevent inadvertent operation of the leakage control system that are actuated by signals from the reactor protection, engineered safety feature, or containment isolation systems are acceptable. A statement in the SAR that such instrumentation, controls, and interlocks will be provided is acceptable for construction permit (CP) review.

5. The system performance requirements, P&IDs, MSIVLCS drawings, and the results of failure modes and effects analyses are reviewed to assure that the system can function following a design basis LOCA assuming a concurrent single active failure, including the failure of a single main steam isolation valve to close. The reviewer evaluates the analyses presented in the SAR to assure the function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum requirements are met for each failure condition over the required time spans. For each case the design is acceptable if minimum system functional requirements are met. The reviewer also provides the Accident Analysis Branch with an estimate of the quantity of fluid processed by the MSIVLCS, for use in calculating radiological consequences of a LOCA.

6. The leakage control system design is reviewed to verify that valve stem packing leakage or other direct leakage from the main steam isolation valves or other components outside containment is prevented or controlled. Such leakage could bypass the leakage control system and result in untreated releases to the environment. The means for prevention or control need not be part of the leakage control system itself, but should meet the same design standards.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The main steam isolation valve leakage control system (MSIVLCS) includes [the source of the sealing medium, (if used)] pumps, valves, and piping to the points of connection or interface with the main steam lines. The scope of review of the MSIVLCS for the ______________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases and the requirements for operation of the system during loss-of-coolent accident conditions. (CP)] [The review has determined that the design of the MSIVLCS and supporting systems is in conformance with the proposed design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the MSIVLCS and supporting systems to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, staff technical positions, and industry standards.

"The staff concludes that the design of the MSIVLCS conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Pehnomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

4. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."