U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 7.1 INTRODUCTION

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)

I. AREAS OF REVIEW

Section 7.1 of the applicant's safety analysis report (SAR) contains information pertaining to safety-related instrumentation and control systems, their design bases, and the applicable acceptance criteria. EICSB reviews this information as detailed in III of this plan, and also determines the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format (Item 4.1 of Ref. 1).

The secondary review branches (APCSB, CSB, RSB) review the safety-related system tabulations for completeness, i.e., to verify that all safety-related systems within their respective areas of primary review responsibility have been identified. If systems other than those identified are deemed to be safety-related, this information is transmitted to EICSB.

This review plan also includes evaluation of the proposed technical specifications given in SAR Chapter 16 to assure that they are adequate with regard to safety system settings, limiting conditions for operation, and periodic surveillance testing of instrumentation and controls.

II. ACCEPTANCE CRITERIA

The identification of safety-related systems is acceptable when it can be concluded that the integrated response of these systems assures the safety of the plant in normal operation, anticipated operational transients, and postulated accidents.

Table 7-1, "Acceptance Criteria for Controls," lists the criteria currently applicable to safety-related instrumentation and control systems (acceptance criteria for safety-related electric power systems are listed in Table 8-1). Conformance to these criteria does not necessarily establish the adequacy of the functional performance and reliability of these systems. However, omission of any of the criteria will in most cases be an indication of system inadequacy. Therefore, the identification of the criteria applicable to safety-related instrumentation and control systems is acceptable if it includes all of the criteria listed in Table 7-1, and if the SAR contains a statement to the effect that these criteria are implemented, at the operating license (OL) stage, or will be implemented, at the construction permit (CP) stage, in the design of these systems.

The fundamental bases for acceptance of the proposed technical specifications are that the limiting conditions for operation are such that sufficient equipment is required to be available for operation to meet the single failure criterion; that equipment outages that are permissible for a short period of time still leave available sufficient equipment to provide the protective function assuming no failures; and that the provisions of the technical specifications are compatible with the safety analyses.

III. REVIEW PROCEDURES

Safety-related systems fall into three categories: basic safety systems, auxiliary supporting systems, and other systems important to safety.

Basic safety systems are those that directly perform a protective function. Examples are the reactor trip system, the emergency core cooling system, the containment isolation system, and the containment spray system. The reactor trip system provides reactor protection by fast insertion of negative reactivity (control rods) when plant conditions approach design safety limits. All the other systems listed are engineered safety features (ESF) systems; their function is to mitigate the consequences of postulated design basis accidents.

Auxiliary supporting systems are those that must function to enable operation of the basic safety systems. Component cooling systems, service water systems, ventilation systems, and electric power systems which serve ESF and reactor trip components are examples of auxiliary supporting systems. These systems must meet the same criteria as the basic safety systems they support.

Other systems important to safety are those systems which operate to reduce the probability of occurrence of specific accidents, or to maintain the plant (including other safety systems) within the envelope of operating conditions postulated in the accident analyses as being required to assure full protection capability. Examples of this type of system are the cold loop startup control (interlocks) system, the accumulator tank isolation valve control (interlocks, position indication, alarms) system, and the plant status and alarm systems that provide the operator with the information necessary for initiating manual protective action. These systems are primarily instrumentation and control systems characterized by having a functional interface with the operator. The same safety criteria apply. However, in application to this type of system, the criteria are usually further defined by regulatory guides and in branch technical positions of the EICSB.

The EICSB review encompasses all of the electric power, instrumentation, and control systems associated with all three categories of safety-related systems described above, with particular emphasis on the elements which constitute the protection system (as defined in IEEE Std 279-1971) and the Class IE electric systems (as defined in IEEE Std 308-1971). The safety-related electric power systems are covered in the standard review plans for Chapter 8 of the SAR. The standard review plans for SAR Chapter 7 are concerned only with the safety-related instrumentation and control systems.

The review of SAR Section 7.1 and applicable portions of the plant technical specifications is performed as follows:

1. EICSB will establish that all safety-related systems are identified, and that this identification does not conflict with the more detailed information provided in other sections of the SAR, particularly in Chapters 6 and 8 and in subsequent sections of Chapter 7. The definitions of safety-related systems presented above should be used as an aid in assessing the completeness of the identification. The secondary review branches (APCSB, CSB, RSB) will confirm the identification of all safety-related systems within their respective areas of primary review responsibility. If systems other than those identified are deemed to be safety-related, this information should be transmitted to EICSB. Particular care should be exercised to assure that all systems postulated in the accident analyses (Chapter 15) as being required for safety are identified as safety-related systems.

2. EICSB verifies that other systems described in the SAR (particularly in Chapters 5, 6, 8, 9, 10, 11, and 15) but not identified by the applicant in Section 7.1 are not required for safety. The reviewer should obtain concurrence from the secondary review branches with regard to systems considered to be safety-related by EICSB, but which have not been identified as such by the applicant. Written requests for evaluation should be made to the secondary review branches when there are novel designs or significant differences of opinion.

3. EICSB verifies that the safety-related systems are categorized by supplier, i.e., those designed and supplied by the nuclear steam system supplier and those designed or supplied by others.

4. EICSB verifies that systems identical to those of reference plants that have recently received construction permits or operating licenses and those that differ from the reference plants are so identified.

5. EICSB verifies that for those systems that are different from those of the reference plants, the differences are described and justified to the extent necessary for an evaluation of their safety significance.

6. EICSB confirms that the criteria identified as being applicable to the design of safety-related instrumentation and control systems include those criteria listed in Table 7-1. This identification meets the applicable requirements of General

Design Criterion 1, "Quality Standards and Records," of Appendix A of 10 CFR Part 50. General Design Criterion 1 also 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." Therefore, the SAR should include (1) a discussion regarding the applicability of each criterion listed, and (2) a statement to the effect that the criteria are implemented (OL) or will be implemented (CP) in the design of safety-related instrumentation and control systems.

7. EICSB verifies that technical design bases are provided (reference to other sections of the SAR is acceptable) for all the various functions of the protection system.

8. Applicable portions of the proposed plant technical specifications (SAR Chapter 16) are reviewed by EICSB and the secondary review branches to:
   a. Confirm the suitability of the safety limits, limiting safety system settings, and the limiting conditions for operation.
   b. Verify that the frequency and scope of periodic surveillance requirements are adequate.

For a CP review, it is only necessary to confirm that the applicant has identified those variables, conditions, or other items which have been determined to be probable subjects of the technical specifications (See 10 CFR §50.34(a)(5).). The applicant's justification for the selection of those items is evaluated with special attention to any that may significantly influence the final design. The specific provisions of the proposed technical specifications are not approved during the CP review. However, any specific provisions which are known to be unacceptable or which may influence acceptance of the preliminary design of the plant should be brought to the applicant's attention and, if approrpiate, included in that portion of the staff's safety evaluation report pertaining to the design of the affected systems.

For an operating license review, the proposed technical specifications are reviewed and evaluated in depth in accordance with the requirements of 10 CFR §50.36. For the EICSB areas of review, a check is made that the limiting conditions for operation (LCO) agree with the surveillance requirements, i.e., for each system or component that is the subject of a LCO, there must be corresponding surveillance requirements. Each system or component that performs a function for which credit is taken in the accident analyses should be the subject of an LCO. The limiting safety system settings should be in accordance with the values assumed in the accident analyses, including appropriate allowances for instrument error, drift, etc. If the acceptance of the design of a particular system is based upon required plant conditions or particular operating procedures, such requirements should be included in the final technical specifications and, if appropriate, noted in that portion of the staff's safety evaluation report pertaining to the design of the affected system.

IV. EVALUATION FINDINGS

EICSB 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 applicant has identified the safety-related instrumentation and control systems and the applicable safety criteria and has documented his intent to design and implement these systems in accordance with the criteria. It is concluded that implementation of these systems in accordance with the criteria provides assurance that the plant will perform as designed in normal operation, anticipated operational transits, and postulated accident conditions, and meets the applicable requirements of General Design Criterion 1."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."*

*All references for this plan are included in Standard Review Plan Table 7-1.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 7.2 REACTOR TRIP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Core Performance Branch (CPB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

EICSB reviews Section 7.2 of the applicant's safety analysis report (SAR), which describes the reactor trip system (RTS). The reactor trip system, which is part of the reactor protection system, includes those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices, that are required to initiate reactor shutdown. The RTS is designed to initiate automatically the reactivity control system (control rods), to assure that specified acceptable fuel design limits are not exceeded. It also includes those safety-related portions of control systems, the actions of which inhibit or limit the response of the reactivity control system to ensure that fuel design limits and safety limits are not exceeded.

Although the design configurations of RTS's for nuclear reactors vary significantly, it is possible by use of the diagram in Figure 7.2-1 to define the RTS of each nuclear steam supply system (NSSS) to the extent necessary for the purpose of identifying the EICSB primary review responsibility.

As shown in Figure 7.2-1, the RTS includes several sensors (usually 4) to measure each parameter such as neutron flux, primary system pressure, reactor outlet temperature, etc. These parameters are detected by sensors of various principles and types that provide electrical signals, mostly at low current or voltage levels. The sensors are located at many locations throughout the plant. It is necessary to determine that each location is suitable for the type of sensor used and that its transmission circuitry (channel) is properly routed to the RTS cabinets in which the electronic signal conditioning equipment is located. Most often, sensors are mounted on local racks and panels. Their arrangement should be considered in the review. For example, consideration should be given to the routing of sensing lines from the process system taps to the sensors, the sensor mountings on racks, and the arrangement of local racks and panels within the plant. The paths of transmission circuitry include routing through containment electrical penetrations and into the cable spreading room. These regions deserve special review attention with regard to ascertaining RTS compliance with the acceptance criteria of Section II of this review plan.

The reactor trip system cabinets that include signal conditioning equipment, logic arrangements, test circuitry, indicators, alarms, and other features are the focal point of the RTS. The cabinets are usually located in the control room area. In addition to reviewing the cabinets and their contents against the acceptance criteria, the reviewer must show that the cabinets are not vulnerable to significant degradation from external influences. Other significant RTS cabinets include those that contain the system trip actuation devices themselves. The actuation devices and the power circuitry to the actuated devices (control rod drives) are also within the scope of the EICSB review; however, the control rod poison sections and the control drives are reviewed by others.

The power supply for the RTS is included in the EICSB review to the extent that the review must show that loss of power would not result in RTS failure to function. The review need not address the capability of the power supplies, usually motor-generator sets, to supply power. However, uniqueness of voltage and frequency requirements for certain RTS motor-generator sets and power supplies must be considered.

Testability of the RTS must be reviewed to ensure that the entire system is fully testable. The EICSB reviewer must ascertain that the test circuitry and test methods used do not compromise the independence of redundant circuits and equipment and do, in fact, enhance RTS reliability. This concern is particularly significant for newer solid state designs incorporating automatic test features.

Another review area of significance includes the interlock circuits that are provided to inhibit control rod motion. These are actuated from safety-related control system sensors such as those that monitor control rod position, turbine trip, etc. Also, protective interlocks actuated from loop isolation valve switches that are used to reset RTS parameter trip levels to more conservative values must be reviewed, along with manual selector switches that are also used to reset protection system trip levels as required for other modes of operation than the normal full power operating mode. These are shown schematically in Figure 7.2-1.

A review of measures involving reactor shutdown that are required to satisfy the design requirements for "anticipated transient without scram" (ATWS) events is included in this section. These measures, for the most part, have not been defined by nuclear steam system suppliers. The EICSB review of the proposed measures will be conducted to assure compliance with the staff technical report WASH-1270, Section V (Ref. 5). The criteria for measures required to make ATWS acceptable are currently under development and will be promulgated as branch technical positions and subsequently as regulatory guides.

The descriptive information, including electrical single line diagrams, electrical schematics (for the operating license stage only), logic diagrams, and physical arrangement drawings are reviewed. The objectives are to determine, on the basis of the most recent diagrams available, that the RTS satisfies the acceptance criteria and to determine that the RTS will perform its intended function during accident conditions and other transient conditions identified in the safety analysis report (SAR) accident analyses. This capability must be maintained during all plant operating modes including start-up power operation, shutdown, and refueling, as defined by the technical specifications.

The depth of review for a plant at the construction permit (CP) stage is limited. For a construction permit, design criteria and preliminary designs are reviewed in order to establish a basis for acceptance. The level of detail need only be sufficient to provide reasonable assurance that the final design will conform to the design bases and that the design bases themselves provide an adequate margin for plant safety. For an operating license (OL), the final design diagrams and results of analyses are reviewed to determine that the required safety functions can be accomplished.

The review is also to include evaluation of the proposed technical specifications to assure their adequacy. Refer to Standard Review Plan (SRP) 7.1 for the considerations involved.

In summary, the primary review area within the scope of the EICSB for SAR Section 7.2 includes:

1. The descriptive information, design bases, and analyses for the reactor trip system.

2. The descriptive information and design bases for supporting systems interfacing with and essential to the operation of the reactor trip system.

In cases where the design is similar to that of plants previously reviewed, the reviewer may determine that it is not necessary to review every facet of the design but may instead select and place emphasis on the most critical areas. Conversely, when concepts that have not previously been reviewed by the staff are received for review, evaluation considerations beyond those outlined in this section may be applied as necessary to assure that the proposed designs will function properly and meet all applicable requirements.

To assure that the auxiliary supporting systems that are essential to RTS operation will adequately maintain the required environmental conditions in areas of the plant where RTS equipment is located, APCSB support is required in the evaluation of cooling systems, heating and air conditioning systems, etc. The APCSB provides assistance in determining that the RTS will be capable of performing its function with auxiliary supporting systems degraded to their limiting conditions for operation. The auxiliary systems are described in SAR Chapters 9 and 10, for which APCSB has primary review responsibility.

Assistance is required from the CPB in reviewing the reactivity control aspects of the RTS, including negative reactivity available in control rods, allowable reactivity insertion or withdrawal rates, and reactivity distributions throughout plant life. The CPB reviews the placement of neutron sensors with regard to measurement of the flux spatial dependence, the flux magnitude, and calibration effects for all operating modes throughout core life. CPB assistance is also required to establish technical specifications for core protection instrumentation with regard to limiting conditions for operation and limiting safety system settings. The plant nuclear design is discussed in SAR Section 4.3, for which CPB has primary review responsibility.

EICSB requires support from the MEB to review seismic qualification tests and supporting analysis for RTS equipment. The MEB review responsibilities in this regard are discussed in SRP 3.10.

The RTS design and construction must be carried out in accordance with the quality assurance requirements of 10 CFR Part 50, General Design Criterion 1 and Appendix B. QAB assistance is required to make this determination. QAB also determines that the quality assurance program documentation required of applicants and the proposed QA/QC organizations are acceptable. The QAB review responsibilities are discussed in SRP 17.1 and 17.2.

To assure that the location, number, and ranges of sensors provided to monitor the performance of the reactor heat transfer systems and related equipment are adequate, the EICSB requires RSB support. RSB assistance is also required to establish technical specification requirements for heat transfer system instrumentation with regard to limiting conditions for operation and limiting safety system settings. The RSB primary review responsibilities are discussed in SRP 4.4, 6.3, and 15.

II. ACCEPTANCE CRITERIA

In general, the reactor trip system is acceptable if it includes adequate redundancy; meets the single failure criterion; has the capacity and capability to safely and reliably shut down the reactor; is fully testable; is capable of functioning during and after design basis events and accidents; and satisfies applicable requirements of the regulations and the recommendations of Institute of Electrical and Electronic Engineers (IEEE) standards, regulatory guides, and branch technical positions. Section V of this plan lists those regulations, standards, guides, and positions used by the reviewer as aids in ascertaining that the above criteria have been met. Section III of this plan discusses the application of these evaluation guides to the review.

The general design criteria and IEEE Std 279-1971 set forth requirements that must be met by all RTS designs. Supporting auxiliary systems must also satisfy these requirements. Appendix A to this plan provides the reviewer with a summary of the use of IEEE Std 279 in the review.

The regulatory guides and branch technical positions set forth acceptable methods of implementing criteria and are not requirements. They serve to resolve problems by proposing particular solutions. Industry standards and topical reports referenced in a SAR may be used as a basis for approval of a design. However, acceptability of the standards and topical reports referenced, but not previously reviewed, must be determined in order to complete the review of the SAR.

Acceptance criteria for specific areas of RTS design are as follows (a complete listing of these criteria is included in Table 7-1, attached to the Chapter 7 standard review plans):

1. System Redundancy Requirements
   General Design Criteria 20 through 29 set forth requirements with regard to functional redundancy considerations. General Design Criteria 2, 3, and 4 set forth the external considerations that must be reviewed to assure that redundancy is not compromised. IEEE Std 279 and IEEE Std 379 are also useful to the reviewer in determining redundancy requirements for the RTS.

2. System Conformance with the Single Failure Criterion
   The General Design Criteria applicable to the preceding discussion on system redundancy requirements (II.1, above) apply equally to system conformance to the single failure criterion. In addition to the general requirements of these regulations, Regulatory Guide 1.53 (as it relates to IEEE Std 379) and IEEE Std 279, paragraphs 4.2, 4.7.3, 4.7.4, 4.7.4.1, 4.7.4.2, and 4.11 explicitly address the single failure criterion and form the basis for judging system conformance to the single failure criterion. Also, see Appendix A to this plan for additional guidance.

3. System Capability and Reliability
   The general requirements for RTS capability are included in General Design Criteria 20 through 29. With the exception of RTS response time, the analyses performed by the CPB and described in SRP 4.3 serve as the basic acceptance criteria for capability. The basis for system response time acceptance is established in the SAR, usually in Chapters 7 and 15. RTS reliability considerations and their conformance to General Design Criterion 21 are based on analyses, as documented in NSSS topical reports, and on testing and operating experience with given hardware.

4. System Testability
   The criteria used to judge system testability and conformance with General Design Criterion 21 are basically those contained in IEEE Std 279, IEEE Std 338, and Regulatory Guide 1.22. In addition, initial qualification of the system must be found acceptable on the basis of IEEE Std 336, IEEE Std 344 (as modified by Branch Technical Position EICSB 10), and Regulatory Guide 1.68 with regard to surveillance. Also, an acceptable design must satisfy Regulatory Guide 1.47 as augmented by Branch Technical Position EICSB 21.

5. System Capability During and Following Design Basis Events
   The method used to assure that the RTS will be capable of performing its protective function during and following design basis accidents is that of equipment qualification for the conditions postulated to accompany the events.

   General Design Criteria 2, 3, and 4 identify events of concern and state acceptance objectives. IEEE Std 344 (as modified by Regulatory Guide 1.29 and Branch Technical Position EICSB 10) for seismic qualification, IEEE Std 317, IEEE Std 323, and

IEEE Std 336 for environmental qualification provide the acceptance criteria. IEEE Std 336 is augmented by Regulatory Guide 1.30, IEEE Std 317 is augmented by Regulatory Guide 1.63, and IEEE Std 323 is augmented by Regulatory Guide 1.89.

6. Identification of Control Panels, Racks, Equipment, Cables, and Cable Trays
The method used for identifying RTS cables and cable trays as safety-related equipment in the plant, and the identification scheme used to distinguish between redundant equipment, racks, panels, cables, and cable trays are acceptable if found to be in accordance with Section 5.1.2 of Regulatory Guide 1.75. IEEE Std 279, paragraph 4.22 also addresses identification criteria.

7. Separation of Equipment, Cables, and Cable Trays
Regulatory Guide 1.75 provides a basis for review and acceptance of the separation criteria presented in the SAR.

8. Vital Supporting Systems
The auxiliary systems that are required to assure RTS functional capability should satisfy the same acceptance criteria as the RTS.

9. Technical Specifications
The acceptance criteria for technical specifications are identified in 10 CFR §50.34 and 50.36. Usually the most recently licensed plant of the type being reviewed serves as a model for the technical specifications. Standard technical specifications are also in preparation at this time. Refer to SRP 7.1 for technical specification considerations.

For those areas of review identified in Section I of this plan as being the responsibility of other branches, the acceptance criteria and their application are included in the appropriate sections of the applicable standard review plans. There are criteria that are used by both primary and secondary review branches as the basis for accepting a design. As they relate to the RTS, some of these criteria and their application are presented below.

In assuring the adequacy of the seismic design of Category I instrumentation and electrical equipment, both the MEB and EICSB perform reviews to ascertain that the proposed design satisfies IEEE Std 344 as supplemented by Branch Technical Position EICSB 10.

To assure that the requirements of General Design Criterion 1 and Appendix B of 10 CFR Part 50 are met in the reactor trip system, the quality assurance program for the RTS Class IE instrumentation and electrical equipment must satisfy the requirements of IEEE Std 336, as augmented by Regulatory Guide 1.30.

III. REVIEW PROCEDURES

The main objectives in the review of the reactor trip system are to determine that this system includes the required redundancy, satisfies electrical and physical independence requirements and the single failure criterion, has the capability and reliability required, is testable, is capable of performing its function during and following design basis events, and can safely shut down the reactor in conformance with all the general design criteria requirements for RTS and the requirements documented in the accident analysis chapter of the safety analysis report.

In the construction permit (CP) review, the descriptive information, including system safety design bases and their relationship to the acceptance criteria, preliminary analyses, electrical single line diagrams, preliminary physical arrangement drawings, functional logic diagrams, and functional piping and instrument diagrams (P&IDs) are examined to determine that there is reasonable assurance that the final design will meet the above objectives. Included in this review, the design criteria for establishing trip setpoints must be evaluated to show conformance to the following guidelines:

(1) The range selection for instrumentation shall be such as to exceed the expected range of the process variable being monitored.
(2) The accuracy of all the safety trip points will not be numerically larger than the accuracy that was assumed in the accident analysis.
(3) The trip setpoints should be located in that portion of the instrument's range which is most accurate and must be located in a region with the required accuracy.
(4) All safety trip points will be chosen to allow for the normal expected instrument system setpoint drift such that the technical specification limit will not be exceeded.
(5) Verification of the above criteria shall be demonstrated as a part of the qualification test program required by IEEE Std 323-1974.

At the operating license (OL) stage of review, these objectives are verified in the review of final electrical schematics and physical arrangement drawings. In addition, a site visit is conducted to assure that the design objectives have, in fact, been implemented in accordance with the design bases and criteria. Appendix 7-B to the Chapter 7 standard review plans contains a typical site visit agenda.

This section describes the method and reasoning to be employed by the reviewer in making a determination as to RTS acceptability. For the purpose of illustration, the RTS system as presented in Figure 7.2-1 is shown as being comprised as two identical, redundant subsystems.

Prior to reviewing Section 7.2 of the SAR, the following background information should be briefly reviewed, in addition to the balance of Chapter 7:

Chapter 1 of the SAR, to become familiar with the general operation of the plant, from both the safety and the operational standpoints.

Chapter 4, the reactor design, in particular the nuclear design, Section 4.3, and the thermal and hydraulic design, Section 4.4.

Chapter 5, on the design of the reactor coolant system, Sections 5.1, 5.2.1, and 5.2.2.

Chapter 6, to note the engineered safety feature provisions.

Chapter 15, to become familiar with the representative types of events for which analyses have been documented. In particular, the effects of failures of the protective functions, and the assumptions and initial conditions that form the bases of the accident analyses are noted.

Chapter 16, to become familiar with limiting conditions for operation, limiting safety system settings (i.e., trip setpoints), and surveillance requirements that pertain to the RTS.

Chapter 17, to note the quality assurance considerations addressed.

The single most relevant document used in the review of the RTS is IEEE Std 279. Conformance of the RTS to the design requirements stated in Sections 3 and 4 of this standard, together with conformance to the requirements of the general design criteria and the functional requirements derived from the accident analyses, will result in an acceptable design. Guidance on the use of IEEE Std 279 is provided in Appendix A of this plan. The general methodology by which the reviewer conducts his review is outlined below by addressing "key concerns" such as redundancy, independence, single failures, capability, and testing.

1. <u>System Redundancy Requirements</u>

With the assistance of the CPB and the RSB, as needed, EICSB determines that the system redundancy requirements are satisfied. Generally, a minimum degree of redundancy of one satisfies RTS requirements. Most RTS parameters are monitored by four sensor channels and only two of four channels are required to initiate the RTS logic channel protective action.

Where it is determined that the spatial dependence of a parameter requires several sensor channels to assure core protection, the redundancy requirements are determined for the individual case. Once design adequacy is established, the reviewer must relate the design requirement to the limiting conditions for operation in the technical specifications. In certain designs, for example, adequate monitoring of core power requires a minimum number of sensors arranged in a given configuration to permit unrestricted power operation. When, because of system degradation, the minimum number of sensors are not available, operation must be restricted. This aspect of redundancy must be dealt with in coordination with the CPB to establish conditions of restricted operation.

Another area where the redundancy requirement of the RTS may have to be defined on an individual core basis is the allowable power operation for reactor coolant systems that have loop isolation valves. Here, the redundancy of the instrumentation provided on the reactor coolant system piping, and on steam generators in the case of pressurized water reactors (PWR's), must be reviewed with the RSB to determine whether the reactor system instrumentation redundancy (not channel redundancy) requirement has been degraded below that on which the accident analyses are based if isolation valves are closed.

With regard to redundancy requirements considered strictly from an electrical point of view, it is only necessary to assure that at least two redundant logic trains (minimum degree of redundancy of one) are provided to initiate reactor trip. From this standpoint the review may be reduced to a simple analysis in which redundant paths from sensors to logic and to actuation devices are identified to assure that the RTS functional requirements are met. It is pointed out that redundancy may be accomplished by equipment that is diverse in principle so long as the same level of protection is provided.

In this discussion on RTS redundancy, it is appropriate to reference Figure 7.2-1. Notice that for required protective functions, the RTS sensors, initiation devices, logic matrices, and actuation and actuated devices all must be redundant. Also note that modules of one channel must not affect those of another channel.

Another area that must be reviewed with regard to redundancy has to do with the measures to be included in nuclear power plants to deal with ATWS events. These measures must be reviewed to assure that they are unaffected by failures that could disable the RTS.

2. <u>System Conformance with the Single Failure Criterion</u>
In evaluating the adequacy of the RTS system in meeting the single failure criterion, both electrical and physical independence must be considered.

   a. <u>Electrical Independence</u>
   To assure electrical independence, the design bases governing the electrical independence of redundant sensors, logic elements, and actuation channels are required to satisfy not only paragraph 4.6 of IEEE Std 279, which states that, "channels that provide signals for the same protection function shall be independent, and the likelihood of interaction between channels is considered," but also, the requirement of paragraph 4.7.2. This paragraph requires that, "the transmission of signals from protection system channels that are used for other purposes, (non-protective) such as control or readout and indication, are properly isolated to ensure that no credible failure at the output of an isolation device shall prevent the associated protective channel from meeting performance requirements." Examples of credible failures at the output of isolation devices are provided in paragraph 4.7.2.

   b. <u>Physical Independence</u>
   To assure physical independence, the design bases governing the physical separation of redundant equipment including sensors, cables, cable trays, racks, panels, and control boards are required to be in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems." This regulatory guide sets forth acceptance criteria for the physical separation of circuits and electrical equipment that is included in the RTS.

   Another review objective is to determine whether the RTS is located in seismic Category I structures. In certain designs, RTS sensors may be located in a non-seismic Category I structures such as the turbine building. For these special cases, the reviewer must assure that the most reasonable installation of sensors and circuits is provided in regard to physical protection against damage from a seismic event. Further guidance is provided by Branch Technical Position EICSB 15.

c. <u>Single Failure Criterion</u>
To assess the RTS acceptability with regard to the single failure criterion, IEEE Std 379 and Regulatory Guide 1.53 are used. Again, as was the case for redundancy requirements, review for compliance with the single failure criterion may be reduced to an analysis in which it is determined that the system can perform all protective functions concurrent with failure of any sensor, logic circuitry and components that meet the single failure cirterion. IEEE Std 279, paragraph 4.2, provides an additional example of single faiulre criterion application.

With regard to power requirements, the RTS must be reviewed to assure that no failure of a power supply will result in maintaining power to the system such that the protective function (trip) of the RTS is negated (fail-safe design). For example, loss of power to a sensor channel should cause a channel trip. Similarly, a loss of power to a logic element or actuator channel should result in a trip. Exception to this latter rule may be taken so long as the single failure criterion is satisfied and the power sources required are designed as Class IE power systems.

The RTS logic matrices should be reviewed to determine whether redundant circuitry includes the contacts of relays or switches in mutually redundant logic ciruits. This task can be accomplished during the OL detailed drawing review. When violations of the single failure criterion are found, they are to be identified to the applicant and corrected. The staff safety evaluation report should discuss the final disposition of designs that are revised to satisfy the acceptance criterion.

The RTS equipment arrangement must be reviewed to assure that no single credible event will result in a loss of redundant circuits or equipment. This matter is discussed further in III.5.b, below.

3. <u>Identification of Control Boards, Equipment, Cables, and Cable Trays</u>
To determine that the identification scheme used for Class IE equipment, cables, and raceways in the plant and Class IE internal wiring in the control boards is consistent with Regulatory Guide 1.75, the criteria proposed for identifying Class IE wiring cables, and cable trays are reviewed. This includes such criteria as those for distinguishing between safety-related cable trays of different channels, non-Class IE cable which is run through Class IE cable trays, and non-Class IE cable which is not physically associated with any Class IE division. IEEE Std 279, paragraph 4.22, also discusses identification. Color coding is a preferred method of identification. In

multi-unit paths that share source spaces, it is particularly important to retain unit identifications along with channel identification.

4. <u>System Testing and Inoperability Surveillance</u>

The proposed preoperational and initial startup test programs for the RTS and its supporting systems are reviewed to verify that the proposed programs are consistent with the requirements set forth in IEEE Std 279, IEEE Std 308 (as augmented by Regulatory Guide 1.32), and Regulatory Guides 1.22 and 1.68.

The descriptive information as supplemented by functional logic diagrams (CP and OL) and electrical schematics (OL) are reviewed to verify that the design has the necessary provisions to permit testing of the RTS on a periodic basis when the reactor is in operation. The reviewer is guided by the recommendations set forth in Regulatory Guide 1.22 and IEEE Std 279, paragraph 4.10, in arriving at an acceptable method of periodic testing of actuation devices (e.g., solenoids, breakers) and actuated equipment (control rods). The same guidance is used to review testability of all modules, relays, permissives, bypasses, and safety-related control devices.

The descriptive information (CP and OL) and the design implementation as depicted on electrical drawings (OL) of the means proposed for automatically indicating, at the system level, bypassed or deliberately inoperable RTS protection channels are reviewed to ascertain that the design is consistent with Regulatory Guide 1.47 as supplemented by Branch Technical Position EICSB 21 and with IEEE Std 279, paragraph 4.13.

5. <u>Other Matters</u>

a. The Technical Specification considerations for the RTS are outlined in SRP 7.1

b. The APCSB reviews supporting systems such as heating and ventilating component cooling water, service water, etc. to assure that failure of these supporting systems will not result in loss of RTS function as result of a degraded environment. It is necessary to assure that those systems required to maintain environmental conditions within the envelope for which the RTS equipment and circuits were designed and qualified be monitored for performance. Examples of such systems include control room and switchgear room heating and ventilating systems.

   THE APCSB should also assist in determining hazardous conditions that might follow failure of non-safety equipment in regions where RTS components and circuits are located. Specific failures must include, as a minimum, the following: fire, missiles, flooding, jet impingment from pipe breaks, and damage that may be caused by failure of non-seismic Category I structures and components. The EICSB relates these conditions to the ability of the RTS to retain functional capability.

c. To assure that the RTS provides adequate core protection, the CPB should confirm that the accident analyses of SAR Chapter 15 have addressed the requirements of IEEE Std 279, Section 3, "Design Basis." To accomplish this task, it is necessary to confirm that the accident analyses have taken into consideration such matters as spatial dependences, operational limits and margins, transient ranges, system response times, and signal and instrument accuracies.

d. The MEB has primary responsibility for assuring that the seismic design of Category I instrumentation and electrical equipment satisfies appropriate requirements. These include IEEE Std 344 and Branch Technical Position EICSB 10. EICSB supplements the MEB by reviewing the description of the seismic qualification test program (CP) and the results of such tests and analyses (OL) that demonstrate the capability of Class IE instrumentation, control devices, and associated circuits to withstand the effects of seismic event. An integrated review is required.

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 reactor trip system includes the initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices utilized to implement reactor shutdown. The scope of the review included the descriptive information (CP and OL), functional logic diagrams (CP and OL), functional instrumentation and electrical diagrams (CP and OL), and preliminary (CP) and final (OL) physical arrangement drawings and schematics. The review has included the applicant's design bases and their relation to the proposed design for the reactor trip system. The review has also included the proposed means for identification of cables and equipment, periodic testing capability, and the qualification test program (CP) and the results (OL) for demonstrating the suitability of the reactor trip system.

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the reactor trip system and vital 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. These are listed in Table 7-1.

"On the basis of our review we have concluded that the reactor trip system conforms to applicable regulations, guides, technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (EICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

4. Regulatory Staff, "Technical Report on Anticipated Transients Without Scram," WASH-1270, U.S. Atomic Energy Commission, September 1973.

APPENDIX A
STANDARD REVIEW PLAN 7.2
USE OF IEEE STD 279 IN THE REVIEW OF THE RTS

This appendix discusses the requirements of IEEE Std 279-1971, Section 4, as they are used in the review of the RTS.

1. Section 4.1 - This section requires that the RTS perform automatically and with precision and reliability. These requirements must be met over the full range of transient and steady-state conditions. The system must meet these requirements in any environmental condition expected during plant operation in which the applicant's accident analyses take credit for the function performed by the RTS. Other criteria which set forth similar requirements are GDC 2, 4, 10, 13, 20, 21, and 29.

    a. Automatic initiation is required for all protective functions. Manual initiation is also provided and is a requirement. (See Section 4.17 and Regulatory Guide 1.62.)

    b. The precision required in the RTS is that assumed in the accident analyses. (Precision requirements are identified in Section 3.9 of IEEE Std 279.)

    c. Quantitative reliability information for RTS often presented in NSSS topical reports. The reliability requirements for RTS are primarily satisfied by related reactor operating experience. The staff is actively pursuing the question of RTS reliability and the potential for RTS loss of function. (See Reference 5, Section V of SRP 7.2.)

    d. The requirements for precise and reliable operation suggest that the RTS design should avoid unnecessary complexity. "Unnecessary complexity" is a difficult judgement: the reviewer should discuss his concerns with the system designer in detail, and should consult with the section leader and branch chief on this matter.

2. Section 4.2 - This section requires that the reviewer examine several different aspects of each single failure to determine its effect.

    a. The first step in a single failure analysis is to identify components that are not seismic Category I, those that are not qualified for accident environments, and those that serve both safety and non-safety systems. Each of the non-qualified and non-safety grade components and systems are assumed to fail to function if failure adversely affects RTS performance and are assumed to function if functioning adversely affects RTS performance.

b. The consequences of events for which the RTS is designed to provide a protective function are examined. All failures in the RTS that can be predicted to occur as a result of the events are assumed to occur if such events adversely affect RTS performance. In general, the lack of equipment qualification may serve as a basis to assume failures.

c. After assuming the failures of non-safety grade, non-qualified equipment and those failures in the RTS caused by an event, any other single failure is arbitrarily assumed and the resultant performance of the RTS is analyzed to assure that the minimum protective function will be performed.

d. The single failure criterion applies to all electric equipment. No distinction is made between active and passive components.

e. IEEE Std 379 and Regulatory Guide 1.53 are used for additional insight to single failure criterion analysis.

3. Section 4.3 - There are no specific criteria to judge the quality of the equipment used in the RTS. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the RTS.

4. Section 4.4 - It is verified that each component and module has been qualified for normal, upset (i.e., operational transient), and accident environments at its installed location. This applies to all normal and upset conditions, but only to those accident conditions where the components and modules provide a protective function. The components must provide the accuracy, range, and response times required by the accident analyses. SRP 3.10 and 3.11 discuss equipment qualification.

5. Section 4.5 - No credit should be given for "safe" failure modes in meeting this requirement. The comments of Section 4.4 apply. For example, if the most probable effect of a given accident is a loss of energy supply to the RTS, it does not matter, in meeting this requirement, whether or not the loss of energy causes the RTS to perform its protective function. Even though GDC 23 requires that the RTS be designed to "fail safe," acceptance of the RTS design shall not be based on an accident causing a failure, even if that accident-induced failure accomplishes the protective function.

6. Section 4.6 - The requirement for channel independence applies to all portions of the RTS that are designated as redundant channels. Independence is maintained in a number of ways. Physical independence is attained by physical separation and physical barriers. Electrical independence is achieved by isolation devices and utilization of separate power sources and other circuit devices. Verification of compliance with physical separation requirements may be made by comparing the design to Regulatory Guide 1.75 recommendations.

7. Section 4.7 - Control and protection system interaction involves more than examining their electrical isolation and interconnection. The functional performance of control systems must be reviewed to the extent that it is determined that a control system cannot prevent

proper action of a protection system. This section of IEEE Std 279, with regard to isolation devices and multiple failures resulting from a credible single event, is explained by example in the document.

8. Section 4.8 - This requirement is self-explanatory. A protection system that requires loss of flow protection would normally derive its signal from flow sensors. A designer might elect to use an indirect parameter such as a pressure signal or pump speed. The reviewer should review the system to determine whether the indirect parameter would be valid at all times.

   Even a directly measured variable should be reviewed and its response to postulated events compared with the credit taken for the parameter in the events for which it provides protection.

9. Section 4.9 - The most common method used to verify the availability of the RTS input sensors is by cross checking between redundant channels that have readout available. When only two channels of readout are provided, evaluate the applicant's analysis of the effect of the operator choosing the incorrect readout as a basis for operator actions.

   When non-indicating sensors are used, check the test procedure to see whether a bypass indication is provided when the sensor is disabled. Of course, this latter approach should also be applied to indicating sensors when the design necessitates.

10. Section 4.10 - The extent of test and calibration capability that is provided bears heavily on whether the design meets the single failure criterion.

    a. Any failure that is not detectable must be considered concurrently with any postulated, detectable, single failure.

    b. Periodic testing should duplicate, as closely as practical, the overall performance required of the RTS. The test should confirm operability of both the automatic and manual circuitry. This capability must be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be reviewed in detail to confirm that the tests do, in fact, overlap from one test segment to another.

    c. Test frequencies are acceptable if identical to frequencies recently approved on other <u>identical</u> plants. Any changes made in the design or test procedures are not an adequate basis for reducing test frequencies until after experience is gained and the results submitted for review.

    d. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. Check that periodic tests conducted during power operation use only permanently installed test equipment. Also see Regulatory Guide 1.22 and Branch Technical Positions EICSB 22, 24, and 25.

11. Section 4.11 - It is verified that tests can be conducted without initiating a protective action at the system level, and that tests can be conducted without preventing the initiation of a protective action at the system level. In general, it is an operational rather than a safety problem if testing causes the initiation of a protective action. For those parts of the RTS with a degree of redundance greater than one, testing should not require bypass of the channel level protective action. For one-out-of-two systems, the channel protective action may be bypassed only if initiation of the protective action would disrupt plant operation. The bypassed channel must remain operable and operating. In these cases, verify that an interlock is provided that prevents, even with a single failure in the interlock circuits, bypassing both channels and that the single bypass is indicated. See Regulatory Guide 1.22 and Branch Technical Position EICSB 24.

12. Section 4.12 - The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may be required to take action to prevent the unnecessary initiation of a protective action and this is acceptable. In no circumstance should a design be approved where action or inaction of the reactor operator is required to make available the protective actions needed in any operational or shutdown mode of the plant.

13. Section 4.13 - See Regulatory Guide 1.47 and Branch Technical Position EICSB 21 for an explanation of this requirement as it pertains to the RTS.

14. Section 4.14 - In practice, administrative control is used as the basis for assuring that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.

15. Section 4.15 - This requirement is similar to Section 4.12. The phrase "positive means" can be interpreted as either automatic or manual. In the case of manual means, the design must be such that no action or inaction on the part of the reactor operator will prevent the more restrictive set point from being available. It is acceptable for the design to be such that incorrect action or inaction by the operator will cause an unnecessary protective action or prevent placing the plant in an operating mode for which there is inadequate protection.

16. Section 4.16 - "Completion of a protective action" must be defined by the applicant for the RTS. This information should be supplied as a part of the design basis information required by Section 3.0 of IEEE Std 279.

    Generally, completion consists of causing negative reactivity to be inserted. Verify that once initiated, the protective action will continue to completion. Termination by deliberate actions of the operator should never inhibit the protective action.

17. Section 4.17 - Regulatory Guide 1.62 describes an acceptable method of implementing the requirement for manual initiation of protective actions. For those designs that take no credit (in the accident analyses) for manual initiation of protective actions, conformance

with Regulatory Guide 1.62 is an adequate basis for acceptance. In practice, the requirements of IEEE Std 279 are applied to all equipment used by the operator to detect the need for the protective action, to accomplish the protection action, and to confirm completion of the protective action. However, it first should be established that automatic initiation need not or cannot be provided. Cost is not sufficient justification for the lack of automatic initiation.

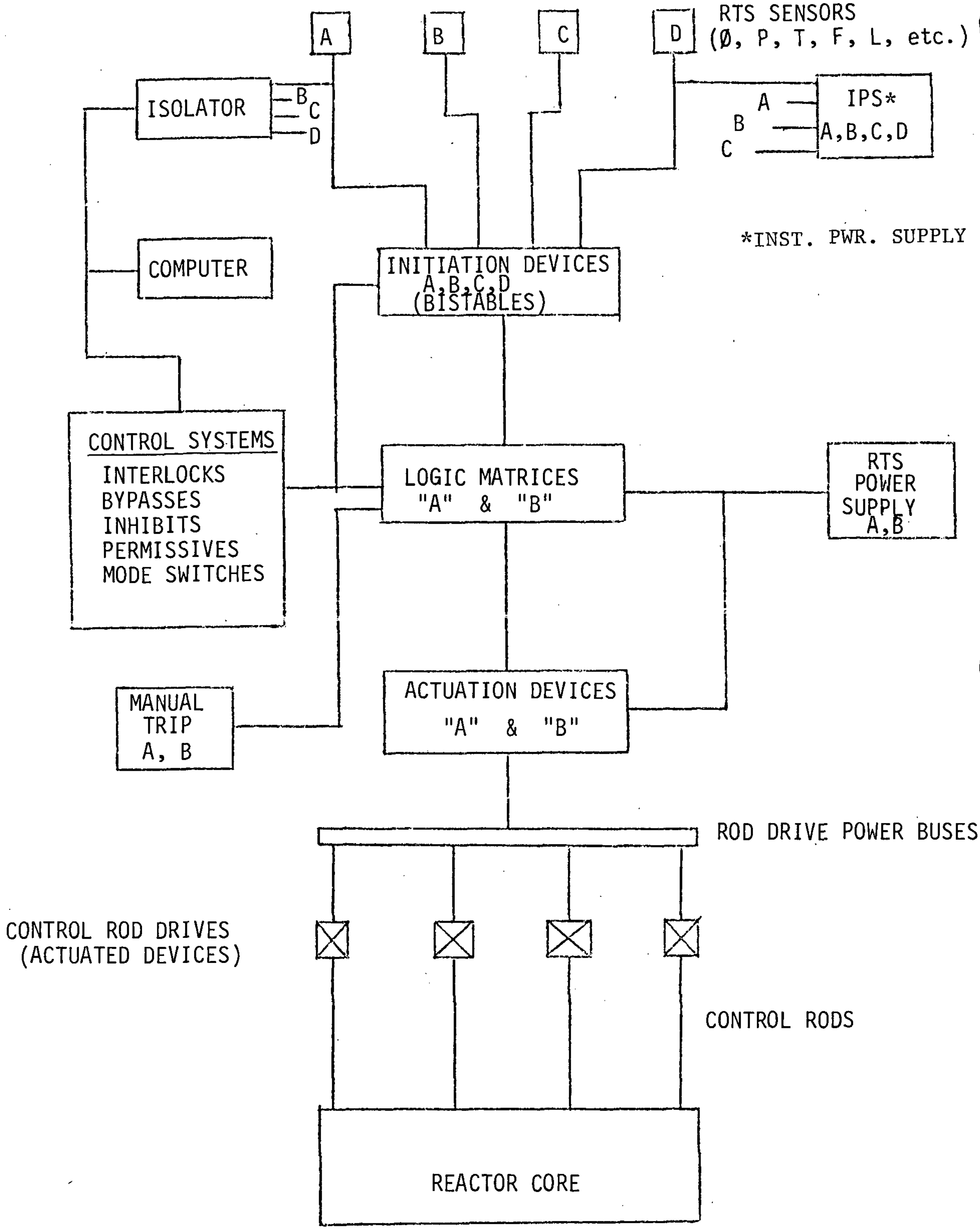
18. Section 4.18 - See procedure above for Section 4.14.

19. Section 4.19 - The method of identification of status at the channel level may be accomplished by lights, indicators, and annunciators.

20. Section 4.20 - The method used to establish adequacy of information readout would include a review of the RTS system inputs to annunciators and event recorders. Engineering judgement serves as the basis for acceptance.

21. Section 4.22 - This requirement is self-explanatory. The preferred identification method is color coding of components, cables, and cabinets. See also Regulatory Guide 1.75.

FIGURE 7.2-1
REACTOR TRIP SYSTEM
(Typical)



U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 7.3 ENGINEERED SAFETY FEATURE SYSTEMS

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)

I. AREAS OF REVIEW

This standard review plan (SRP) covers the portion of the protection system used to initiate and control operation of the engineered safety feature systems and essential auxiliary supporting systems. This portion of the protection system is called the engineered safety feature actuation system (ESFAS).

Typical engineered safety feature (ESF) systems are:

- Containment and Reactor Vessel Isolation Systems
- Emergency Core Cooling Systems (ECCS)
- Containment Heat Removal and Depressurization Systems
- Pressurized Water Reactor (PWR) Auxiliary Feedwater Systems (See SRP 7.4 for review of the safe shutdown functions of this system)
- Boiling Water Reactor (BWR) Standby Gas Treatment Systems
- Containment Air Purification and Cleanup Systems
- Containment Combustible Gas Control Systems

Typical essential auxiliary supporting systems are:

- Electric Power Systems (See Chapter 8 for review plans for these systems)
- Diesel Generator Fuel Storage and Transfer Systems
- Instrument Air Systems
- Heating, Ventilating, and Air Conditioning (HVAC) Systems for ESF Areas
- Essential Service Water Systems

The descriptive information, functional control diagrams, piping and instrument diagrams, electrical schematics (operating license stage only), and physical arrangement drawings, as presented in the applicant's safety analysis report (SAR), are reviewed. The objectives are to determine that the engineered safety feature actuation system satisfies applicable design criteria and will perform as intended during all plant operating conditions and accident conditions for which its function is required. The most significant difference between the review performed for a construction permit (CP) application and that performed for an operating license (OL) application is that the CP review can be based on a preliminary design. The depth of detailed information need only be "sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety (Ref. 1)." In addition, "a construction permit...will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit (Ref. 2)."

The review of the information presented and referenced in Section 7.3 of an SAR is primarily directed to the engineered safety feature actuation system (ESFAS), i.e., the instrumentation and controls used to initiate and control the operation of the engineered safety features. The scope of the EICSB review of Section 7.3 of an SAR includes:

1. The descriptive information, including single line diagrams (CP) and schematic diagrams (OL) pertaining to the ESFAS. The ESFAS includes all electric and electromechanical equipment involved in detecting a plant condition requiring operation of an ESF system and in initiating the operation of the ESF system.

2. The descriptive information pertaining to the instrumentation and control systems for those auxiliary supporting systems that are essential to the operation of either the ESFAS or the ESF systems.

3. The applicant's proposed design criteria for the ESFAS and the instrumentation and controls of essential auxiliary supporting systems.

4. The applicant's analysis of the adequacy of the proposed design criteria and design bases for the ESFAS and the instrumentation and controls of auxiliary supporting systems.

5. The applicant's analyses of how the design of the ESFAS and auxiliary supporting systems conform to the design criteria for these systems.

The RSB and the CSB review, for those ESF systems within their review responsibilities, the following aspects of ESFAS:

(1) The adequacy of the monitored variables, i.e., the suitability of parameters, such as pressure, for initiating operation of a given ESF system.

(2) The acceptability of the proposed trip set points.

The APCSB will advise EICSB of any corrections to the SAR descriptions of auxiliary supporting systems essential to ESF systems and of time intervals available to initiate operation of auxiliary supporting systems.

II. ACCEPTANCE CRITERIA

Acceptance criteria for the review areas of this plan are referenced in Table 7-1 (Ref. 3), which lists the general design criteria (GDC), industry standards, regulatory guides, and branch technical positions that are applicable to the ESFAS and the instrumentation and controls of essential auxiliary supporting systems. These documents either establish design requirements or describe acceptable methods of implementing design requirements. In each of these categories, some documents set forth mandatory design criteria and others describe acceptable methods of design.

The GDC and IEEE Std 279-1971 set forth requirements that must be met by all designs for the ESFAS. In addition, these are also used for essential auxiliary supporting system instrumentation and controls. One purpose of the review is to verify that the applicant has committed to designing the ESFAS and the essential auxiliary supporting system instrumentation and controls in accordance with these mandatory criteria.

The regulatory guides are not mandatory and only set forth acceptable methods of implementing the mandatory criteria. The branch technical positions are used when a particular design problem has an identified and acceptable solution; they also are not mandatory.

Industry standards that are not endorsed by regulatory guides or incorporated in regulations or technical positions, or that have not been previously used and accepted in the licensing process, must be reviewed before they can be accepted as a sole basis for approval of a design. They are useful as guidance for identifying the subjects of importance to be considered in the review of the ESFAS. In all cases, the primary basis for acceptance of an ESFAS design is conformance to the mandatory criteria of the regulations.

III. REVIEW PROCEDURES

This section describes the general procedures to be followed in reviewing the ESFAS. For simplicity, it is written for the ESFAS for a single ESF system comprised of two identical, redundant subsystems. The same procedure should be applied to each ESF system and to each essential auxiliary supporting system.

Background information of interest in the review of the ESFAS is found in a number of SAR sections. A list of these is given below for reference purposes. Most of these reference sections also provide background information for other review plans in Chapter 7.

Chapter 1 of the SAR: for familiarization with the general operation of the plant, both safety and non-safety aspects.

Chapter 3: for a general understanding of the principal architectural and engineering designs of those structures, components, equipment, and systems important to safety.

Section 3.1: for exceptions to criteria applicable to the ESFAS, and for structures suitable for housing ESFAS equipment.

Chapters 4 and 5: for an understanding of the reactor and the reactor coolant system and its interconnections with the ESF systems.

Chapter 6: for the design bases, design features, and functional performance requirements of the ESF system.

Chapter 7: for a detailed understanding of the design and operation of the ESFAS.

Chapter 9: for the design bases, design features, and functional performance requirements of essential auxiliary supporting systems.

Chapter 15: for the courses of accidents for which the ESF system provides protective functions, the effects of failures of the protective functions, and the assumptions and initial conditions that form the bases of the accident analyses.

Chapter 16: for the proposed limiting conditions for operation for the ESF and the ESFAS.

It should be noted that reference to the above sections of the SAR is made to gain an understanding of the purpose of the ESF and an understanding of how the ESF system and the ESFAS are designed and are supposed to function. No "evaluation" should be made of these sections, i.e., the SAR description is taken at face value.

The next step is to evaluate the design against the requirements of IEEE Std 279-1971. This procedure is detailed in Appendix A to this plan. The procedures in Appendix A address only those design requirements that are specific in nature. For example, paragraph 4.9 of IEEE Std 279-1971 requires that the design include means for checking the availability of each system input sensor during operation. Appendix A outlines a straightforward procedure that can be used to determine whether or not this requirement is met.

Appendix A discusses the requirements of IEEE Std 279-1971 and how they are used in the review of the ESFAS and the essential auxiliary supporting systems instrumentation and controls. Although the primary emphasis is on the equipment comprising the ESFAS, the reviewer should consider the protective functions on a systems level. It serves little purpose to approve an ESFAS design unless that design is compatible with the ESF systems and auxiliary supporting systems and unless the design and the accident analyses are compatible. It is not sufficient to judge the adequacy of the ESFAS only on the basis that the design meets the specific requirements of IEEE Std 279-1971. It is also necessary to judge the functional relationship between the ESFAS and the ESF systems themselves.

Other requirements for the ESFAS and the instrumentation and controls of essential auxiliary supporting systems are listed in Table 7-1. Many of these requirements are general in nature and this permits various designs to meet them. For example, GDC 20 requires, in part, that the protection system be designed to sense accident conditions and to initiate the operation of (ESF) systems important to safety. A cursory examination of the descriptive information would be sufficient to determine whether or not the ESFAS is designed to sense accident conditions and initiate the ESF systems. Such general requirements are not detailed here as to review procedures. Specific design features and approaches are described in the EICSB technical positions in Appendix 7-A to Chapter 7 of the review plans.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

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:

"7.3 Engineered Safety Feature Actuation Systems (ESFAS)

The engineered safety feature actuation systems include the instrumentation and controls used to detect a plant condition requiring operation of an engineered safety feature (ESF) system, to initiate action of the ESF, and to control its operation. The scope of review of the ESFAS for the ________________ plant included single line diagrams (CP and OL) and schematic diagrams (OL) and descriptive information for the ESFAS and for those auxiliary supporting systems that are essential to the operation of either the ESFAS or the engineered safety feature systems themselves. The review has included the applicant's proposed design criteria and design bases for the ESFAS and the instrumentation and controls of auxiliary supporting systems, and his analysis of the adequacy of those criteria and bases. The review also has included the applicant's analyses of the manner in which the design of the ESFAS and auxiliary supporting systems conform to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the engineered safety feature actuation systems and necessary auxiliary 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. These are listed in Table 7-1.

"The staff concludes that the design of the engineered safety feature actuation systems conform to all applicable regulations, guides, branch technical positions, and industry standards and is acceptable."

V.. REFERENCES

1. 10 CFR §50.34(a)(3)(iii), "Contents of Applications; Technical Information. Preliminary Safety Analysis Report."

2. 10 CFR §50.35(b), "Issuance of Construction Permits."

3. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

4. Standard Review Plan Appendix 7-A, "Branch Technical Positions (EICSB)."

APPENDIX A

STANDARD REVIEW PLAN 7.3

USE OF IEEE STD 279 IN THE REVIEW OF THE ESFAS AND INSTRUMENTATION AND CONTROLS OF ESSENTIAL AUXILIARY SUPPORTING SYSTEMS

This appendix discusses the requirements of IEEE Std 279-1971, Section 4, as they are used in the review of the ESFAS and instrumentation and controls of essential auxiliary supporting systems.

1. Section 4.1 - This section requires that the ESFAS perform automatically and with precision and reliability. These requirements must be met over the full range of transient and steady-state conditions of the energy supply and environment during all plant conditions in which the applicant's accident analyses take credit for functions performed by the ESFAS. Other criteria which set forth similar requirements are: GDC 2, 4, 10, 13, 20, 21, and 29.

   a. Automatic initiation is required for all protective functions that must be started within a short time of the indicated need for the function. Although GDC 20 appears to require automatic initiation of all protective functions, initiation solely by manual means has been acceptable. However, automatic initiation is preferable for all protective functions, even though they are not needed (according to the accident analyses) for a relatively long time. Where the protective action is initiated solely by manual means, all the actions that need or may need to be performed by the operator during the time interval are reviewed, as are the applicant's basis for not providing automatic initiation. In this latter regard, the cost of automatic initiation is not, of itself, sufficient justification for using manual initiation. If the reviewer's judgement is that manual initiation is sufficiently reliable, then the equipment used by the operator to detect the need for the protection function, and to verify that the protective function has been completed, it must also meet all the requirements applicable to automatically initiated protective functions. See also Branch Technical Position (BTP) EICSB 20.

   b. The precision required in the ESFAS is at least that assumed in the accident analyses.

   c. There are no quantitative requirements established for the reliability of the ESFAS. The design is reviewed to identify any unusual or unique equipment that has not previously been used in nuclear plants. The "type testing" (as defined in IEEE Std 323-1974) that demonstrates such equipment is capable of performing its function is reviewed. The design is also reviewed to assure that no unnecessary interlocks, time delays, or other complexities are introduced in the ESFAS circuits. Where such features do exist, the applicant's design bases and performance analyses should be reviewed to determine that the reliability of the ESFAS is not significantly reduced by the inclusion of such features.

2. Section 4.2 - This is the most fundamental of all the requirements that the ESFAS must meet. It is inherent in other criteria such as GDC 21, 22, 24, 34, 35, 38, 41, 44, 54, 55, and 56.

In evaluating ESFAS conformance with this requirement, the reviewer must examine several different aspects of each single failure to determine its effect. The time of occurrence of the failure and the plant conditions prevailing at that time can significantly alter the effects of any single failure.

a. The first step in a single failure analysis is to identify components that are not seismic Category I, those that are not qualified for accident and post-accident environments, and those that serve both safety and non-safety systems and whose failure can affect the performance of or create the need for the EFSAS. Each of the non-qualified and non-safety grade systems and components are assumed to fail to function if failure adversely affects ESFAS performance and are assumed to function if functioning adversely affects ESFAS performance.

b. Next, the consequences of the events for which the ESFAS is designed to provide protective functions are examined. All failures that can be predicted to occur as a direct or consequential result of an event are assumed to occur if such failures adversely affect ESFAS performance. In general, lack of adequate environmental or seismic qualification testing is sufficient basis to assume a direct or consequential failure of equipment.

c. After assuming the failures of non-safety grade, non-qualified equipment and those failures caused by an event, any other single failure in the ESFAS or its auxiliary supporting systems is arbitrarily assumed and the resultant performance of the ESFAS is analyzed to assure that the minimum protective function will be performed.

d. In choosing the postulated failure to be analyzed, no distinction is made between active and passive components in electrical systems. Further, electrical equipment serving mechanical components that are not required to function in a given event is treated the same as electrical equipment serving "active" mechanical components, i.e., those that must function. (See also BTP EICSB 18.)

e. The meaning of redundancy is discussed in IEEE Std 379 and Regulatory Guide 1.53. Basically, to be considered redundant, there must be no communication, either directly or indirectly, between two systems that can perform the same function. Thus, two systems, each of which can perform a protective function, are not redundant (and therefore do not meet the single failure criterion) if the failure of one system affects in any way the performance of the other system. This includes starting (or not starting) one system by sensing the failure (or operation) of the other system.

3. Section 4.3 - There are at present no specific criteria to judge the quality of the equipment used in the ESFAS. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the ESFAS.

4. Section 4.4 - Standard Review Plans 3.10 and 3.11 discuss the evaluation of equipment qualification. In reviewing the ESFAS, check that each component or module of the ESFAS has

been qualified for normal, accident, and post-accident environments at its installed location. This applies to all normal conditions but only to those accident conditions where the component or module provides a protective function.

5. Section 4.5 - This requirement is similar to Section 4.4 discussed above. No credit should be given for "safe" failure modes in meeting this requirement. For example, if the most probable effect of a given accident is a loss of energy supply to an ESFAS, it does not matter, in meeting this requirement, whether or not the loss of energy causes the ESFAS to perform its protective function. Even though GDC 23 requires that the ESFAS be designed to "fail-safe," acceptance of the ESFAS design should not be based on an accident causing a failure, even if that accident-induced failure accomplishes the protective function.

6. Section 4.6 - The requirement for channel independence applies to all portions of the ESFAS that are designated as redundant channels. Verification of compliance with this requirement and the recommendations of Regulatory Guide 1.75 and IEEE Std 384-1974 concentrates on points of interface between redundant ESFAS components and interfaces between the redundant portions of the ESFAS and non-safety grade systems. For example, switches common to redundant portions of the ESFAS are reviewed for physical independence between redundant switch sections and for the effects on redundant systems caused by a single malpositioned switch. Also reviewed are the functional performances of isolation devices to assure that no failure in non-safety circuits can disable safety functions.

7. Section 4.7 - The interaction of control systems and the ESFAS involves more than examining the electrical interconnection of control systems with the ESFAS. The functional performance of appropriate control systems must also be reviewed to determine whether their effect on plant conditions can indirectly affect the performance of the ESFAS or the ESF. For example, if a cooling water system is used to supply both safety and non-safety equipment, the controls for the cooling water system must be examined to determine whether failure could lead to insufficient cooling water being supplied to the ESF or the ESFAS during an accident. (Also see Branch Technical Position (BTP) EICSB 27.)

   Note that if failure of a system serving both safety and non-safety systems can lead to a condition requiring action by the safety system, then in addition to the failure creating the need for safety action, the ESFAS must be designed to withstand any other simultaneous single failure.

8. Section 4.8 - This requirement is self-explanatory. In addition, it must be verified that the measured variable is the variable that is used in the accident analyses.

9. Section 4.9 - The most common method used to verify the availability of the ESFAS input sensors is by cross checking between redundant channels that have readout available. When only two channels of readout are provided, evaluate the applicant's analysis of the effect of the operator choosing the incorrect readout as a basis for this action.

   Where non-indicating sensors are used, check the test procedure to see whether a bypass indication is provided when the sensor is disconnected from the process system.

10. Section 4.10 - The extent of test and calibration capability that is provided bears heavily on whether the design meets the single failure criterion.

a. Any failure that is not detectable must be considered concurrently with any postulated, detectable, single failure.

b. Periodic testing should duplicate, as closely as practical, the integrated performance required from the ESFAS, ESF systems, and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in fact, overlap from one test segment to another. For example, closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.

c. Test frequencies are acceptable if identical to frequencies recently approved on other identical plants. Any changes made in design or test procedure are not an adequate basis for reducing test frequencies until after experience is gained and the results submitted for review.

d. Test procedures that require disconnecting wires, installing jumpers, or other similar modifications of the installed equipment are not acceptable test procedures for use during power operation. Check that periodic tests conducted during power operation use only permanently installed test equipment. See also Regulatory Guide 1.22 and BTP EICSB 22, 24, and 25.

11. Section 4.11 - Verify that tests can be conducted without initiating a protective action at the system level, and that tests can be conducted without preventing the initiation of a protective action at the system level. In general, it is an operational rather than a safety problem if testing causes the initiation of a protective action. For those parts of the ESFAS with a degree of redundancy greater than one, testing should not require bypass of the channel level protective action. For one-out-of-two systems, one channel may be bypassed only if initiation of the protective action would disrupt plant operation and the other channel remains operable. In these cases, verify that an interlock is provided that prevents, even with a single failure in the interlock circuits, bypassing both channels and that the single bypass is indicated. See also Regulatory Guide 1.22 and BTP EICSB 24.

12. Section 4.12 - The requirement for automatic removal of operational bypasses means that the reactor operator shall have no role in such removal. The operator may be required to take action to prevent the unnecessary initiation of a protective action and this is acceptable. In no circumstances should a design be approved where action or inaction of the reactor operator is required to make available the protective actions needed in any operational or shutdown mode of the plant.

13. Section 4.13 - See Reg. Guide 1.47 and BTP EICSB 21 for an explanation of this requirement as it pertains to the ESFAS, ESF systems, and auxiliary supporting systems.

14. Section 4.14 - In practice, administrative control is used as the basis for assuring that access to the means for bypassing is limited to qualified plant personnel and that permission of the control room operator is obtained to gain access.

15. Section 4.15 - This requirement is similar to Section 4.12. The phrase "positive means" can be interpreted as either automatic or manual. In the case of manual means, the design must be such that no action or inaction on the part of the reactor operator will prevent the more restrictive set point from being available. It is acceptable for the design to be such that incorrect action or inaction by the operator will cause an unnecessary protective action or prevent placing the plant in an operating mode for which there is inadequate protection (as defined by the accident analyses). See BTP EICSB 12 for specific guidance on set point changes required with a reactor coolant pump out of service.

16. Section 4.16 - For the ESFAS, "completion of a protective action" must be defined by the applicant for each ESF system. This information should be supplied as part of the design basis information required by Section 3.0 of IEEE Std 279-1971.

    Generally, completion consists of starting or energizing the components in the ESF system. Verify that once initiated, the protective action will continue until terminated by deliberate actions of the operator and that operator action cannot prevent the initiation of the protective action when the ESFAS determines the need for that action. Exception: "pull-to-lock" control switches have been acceptable even though their manipulation could prevent the protective action from going to completion.

17. Section 4.17 - Regulatory Guide 1.62 describes an acceptable method of implementing the requirement for manual initiation of protective actions. For those designs that take no credit (in the accident analysis) for manual initiation of protective actions, conformance with Regulatory Guide 1.62 is an adequate basis for acceptance.

    For those protective actions which are initiated solely by manual means, there are no specific criteria to judge acceptance at present. In practice, the requirements of IEEE Std 279 are applied to all equipment used by the operator to detect the need for the protective action, to accomplish the protection action, and to confirm completion of the protective actions. However, it first should be established that automatic initiation need not or cannot be provided. Cost is not sufficient justification for the lack of automatic initiation. In judging the adequacy of any manual initiation features, the other tasks that the operator may be required to perform should be determined and then a judgment made as to whether it is reasonable to rely on the operator to perform all necessary actions. In most situations, automatic actuation, backed up by provisions for manual initiation or manual termination, is more reliable than manual initiation alone, no matter how much time is available to take the protective action.

18. Section 4.18 - See procedure above for Section 4.14.

19. Sections 4.19 and 4.20 - Other than the requirements for indication and identification of channel and system level protective actions, there are no specific implementation guidelines

by which to judge the adequacy of a design with respect to the requirements for status indication. Evaluate the applicant's discussion of how the ESFAS designs conform to these requirements. Acceptance is based on the reviewers's engineering judgement.

See also SRP 7.5 for a discussion of review procedures for safety-related display instrumentation.

20. Section 4.22 - This requirement is self-explanatory. The preferred identification method is color coding of components, cables, and cabinets. See also Regulatory Guide 1.75.

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SECTION 7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Reactor Systems Branch (RSB)
Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The systems reviewed under this plan are those instrumentation and control systems associated with parts of the nuclear steam supply system (NSSS) used to achieve and maintain a safe shutdown condition of the plant. The specific arrangement of these parts of both the primary and secondary loops of the NSSS depends on the type of plant (pressurized water reactor, PWR; boiling water reactor, BWR; etc.) as well as on individual plant design features, and the conditions under which the safe shutdown has to be achieved and maintained. There are two kinds of shutdown conditions; hot shutdown and cold shutdown. A hot shutdown is a stable condition of the reactor achieved shortly after a programmed or emergency shutdown (scram) of the plant has taken place. A cold shutdown is a stable condition of the plant achieved after the residual heat removal process has brought the primary coolant temperature below 200°F. In either case, it is necessary that reactivity control systems maintain a subcritical condition of the core and that residual heat removal systems operate to maintain adequate cooling of the core. For a precise definition of both shutdown conditions for a specific plant, see Chapter 16, "Technical Specifications," in the applicant's safety analysis report (SAR).

Examples of systems required for achieving and maintaining a safe shutdown are the auxiliary feedwater system, the residual heat removal system, and the boric acid transfer system (for PWR's).

The review of the instrumentation and control systems associated with the various parts of the NSSS required for safe shutdown, along with the equipment required for their proper alignment from the main control room or from other locations outside the control room, is the responsibility of the EICSB. The review includes the sensors, initiating circuitry, logic bypasses, interlocks, redundancy features, and actuated devices of those systems and auxiliaries which provide the necessary instrumentation and control functions. The EICSB review should confirm that the systems required for safe shutdown, as defined above, and their supporting systems will perform design functions as required for plant shutdown and conform to all applicable acceptance criteria.

The descriptive information contained in the SAR, including single line diagrams, electrical schematics, piping and instrumentation diagrams (P&ID's), and physical arrangement diagrams are reviewed to ascertain that the systems required for safe shutdown meet the acceptance criteria listed in Section II of this plan. For a construction permit (CP) review, a commitment to meet these criteria, together with a preliminary design, can suffice in cases where the design of these systems has not been completed. For an operating license (OL) review, however, the actual design and its implementation should be verified to meet these criteria.

The EICSB review includes the following specific items:

1. The redundancy of power sources, logic, and instrumentation provided for the operation and status monitoring of systems required for safe shutdown. This requires the review of the descriptive information contained in the SAR, functional diagrams, electrical schematics, and P&ID's.

2. The ability of systems required for safe shutdown to function after sustaining a single failure. This requires the review of the descriptive information and diagrams as in (1) above, and in addition the drawings showing the physical layout of the instrumentation, control equipment, and cabling, the design criteria for physical separation of redundant electrical equipment and cabling, the design criteria for providing control and motive power to these systems, the control arrangements for controlled electrically-operated valves, and provisions for sharing of electrical systems between units in multi-unit plants.

3. The criteria for design of instrumentation and electrical equipment, cabling, cable trays, and structures housing parts of the systems required for safe shutdown.

4. The environmental qualification of the electrical and instrumentation equipment and cabling.

5. The online testability of the systems and indication of bypassed or inoperable status of the systems required for safe shutdown.

The APCSB should evaluate the adequacy of those auxiliary systems required for the proper operation of the systems required for safe shutdown. These include systems concerned with compressed air requirements, reactor coolant chemistry, boron concentration, lighting, air conditioning, etc. In particular, the APCSB should determine that the piping, ducting, and valving of redundant vital auxiliary supporting systems meet the single failure criterion. In addition, the APCSB should review the physical arrangement of components and structures related to the systems required for safe shutdown and their supporting systems, and determine that single events will not disable these systems.

The CSB should review the containment ventilation and atmosphere control systems provided to maintain required environmental conditions for electrical and instrumentation equipment associated with the systems required for safe shutdown and located inside containment.

The MEB review should confirm that the seismic qualification of instrumentation and electrical systems is acceptable. This includes the design criteria and testing methods and procedures employed in the seismic design and installation of Category I instrumentation and electrical equipment.

The RSB should review the systems identified as required for safe shutdown, and confirm that the configuration and design bases of these systems are correct, and that all design paramaters such as temperature, pressure, flow rate, and reactivity can be controlled within acceptable limits. This information should be provided to the EICSB. For situations where shutdown is to be accomplished from locations outside of the main control room, the RSB review should establish the adequacy of needed systems and any differences in system alignment or operation that are required to achieve and maintain safe shutdown.

The QAB review should verify that the quality assurance program proposed by the applicant includes the systems required for safe shutdown.

II. ACCEPTANCE CRITERIA

The design materials, qualifications testing, and surveillance of systems required for safe shutdown are covered by several general design criteria (GDC), IEEE standards, regulatory guides, and branch technical positions. A list of applicable criteria, standards, guides, and technical positions is given in Table 7-1 and Appendix 7-A, attached to the standard review plans for Chapter 7.

The instrumentation and control systems required for safe shutdown are acceptable when it is determined that these systems satisfy the following requirements:

1. They have the required redundancy.

2. They meet the single failure criterion.

3. They have the required capacity and reliability to perform intended safety functions on demand.

4. They are capable of functioning during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

5. They are testable during reactor operation.

The criteria listed in Table 7-1 are utilized as the bases for determining that these requirements are met. How these criteria are applied during the review process is discussed in Section III of this plan. The applicability of the acceptance criteria to the review of the systems required for safe shutdown is as follows:

1. System Redundancy Requirements
   GDC 26, 33, 34, and IEEE Std 279 specify the requirements that systems required for safe shutdown, among others, must meet with regard to all operating conditions (such as loss of offsite power), so that they can perform their safety function assuming a single failure. If a determination is made that the systems required for safe shutdown meet the requirements of these criteria, they are acceptable in this regard. Electrical and physical independence requirements as discussed in Standard Review Plans (SRP) 7.2 and 7.3 should be met.

2. Conformance with the Single Failure Criterion
   IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 provide recommendations and guidance for meeting the single failure criterion. Regarding the application of the single failure criterion to the design of manually-controlled electrically-operated valves, the acceptability of proposed designs is based on Branch Technical Position EICSB 18.

3. Identification of Cables, Cable Trays, and Instrument Panels
   The method used for identifying power and signal cables and cable trays as safety-related equipment, and the identification scheme used to distinguish between redundant cables, cable trays, and instrument panels should be in accordance with the recommendations of Sections 5.1.2 and 5.6.3 of Regulatory Guide 1.75, "Physical Independence of Electric Systems," and Section 4.2.2 of IEEE Std 279. Color coding is a preferred method of identification.

4. Vital Supporting Systems
   The instrumentation, control, and electric equipment associated with the auxiliary systems that support the systems required for safe shutdown should meet the same acceptance criteria as for the systems they support.

5. System Testing, Quality Assurance, and Surveillance
   GDC 1, 21, IEEE Std 279, IEEE Std 336, and Regulatory Guides 1.22, 1.47, and 1.68 contain the applicable acceptance criteria with regard to preoperation and periodic testing, quality assurance, and design provisions for indicating the availability of systems required for safe shutdown and essential auxiliary supporting systems.

For the areas of review identified in Section I as responsibilities of other branches, the applicable acceptance criteria are included in the corresponding review plans.

III. REVIEW PROCEDURES

The main objectives of the review of systems required for safe shutdown are to determine that the design of these systems includes the required redundancy; meets the single failure criterion; provides the required capacity and reliability to perform intended safety functions on demand; and provides the capability to function during and after design basis events such as earthquakes and anticipated operational occurrences.

For a CP review, the descriptive information contained in the preliminary safety analysis report (PSAR), including the design bases and their justification with regard to the acceptance criteria, electrical single line drawings, and P&ID's are reviewed to determine that the basic design features and the commitments made provide assurance that the final design will meet the acceptance criteria. During tne OL stage of review, it is verified that the acceptance criteria are met through review of the final electrical and instrumentation drawings and the physical layout drawings, and a site visit, during which a spot-check verification of the design is performed. In order to verify that the acceptance criteria are satisfied, the review is performed in accordance with the following specific procedures.

A major portion of the systems required for safe shutdown are also used as engineered safety feature (ESF) systems, as discussed in SRP 7.3. A major portion of the systems required for safe shutdown are also used as engineered safety feature (ESF) systems, as discussed in SRP 7.3. This plan includes the safe shutdown systems configurations which are not part of ESF systems or result from a realignment of ESF systems. The RSB and APCSB confirm the acceptability of the proposed configuration and the redundancy required for systems required for safe shutdown as specified in GDC 26, 33, 34. The descriptive information, including the electrical one-line diagrams and P&ID's (for CP and OL reviews) and electrical schematics (for the OL review) should be reviewed to verify that the necessary redundancy is provided. This should include instrumentation channels used to sense vital parameters such as temperature, pressure, water level, etc.; the associated logic and actuated devices; and the motive and control power sources.

Conformance with the single failure criterion as specified by IEEE Std 279 and Regulatory Guide 1.53 is verified by review of the same information as for redundancy and may be done, to some extent by necessity, at the same time. The guidance provided by Regulatory Guide 1.53 is excellent for ascertaining that a given design is single failure proof. A particularly important but subtle point to check is one cited in Position 4 of Regulatory Guide 1.53, wherein a single d-c source supplies control power for one channel of system logic and for the redundant actuator circuit.

Certain areas of review need close coordination between primary and secondary review branches in order to make a determination that a specific aspect of the design meets the applicable criteria. Seismic qualification of Class IE equipment, flood protection of safety-related systems and components, and effects of high energy fluid line breaks inside containment or near safety-related equipment are the major areas for which branch coordination is essential in evaluating the acceptability of a given design feature.

For a multi-unit plant where electrical systems are shared, thus resulting in more and complex interaction modes, a fault-tree and decision-tree analysis may be required from the applicant to show that a single failure or a single event resulting in multiple failures will not result in unacceptable consequences with respect to the capability of systems required for safe shutdown to perform safety functions when required. Additional guidance with regard to the single failure criterion as it relates to shared electric power systems is given in the review plans for Chapter 8.

For the case of manually-controlled electrically-operated valves in these systems, the acceptability of the proposed design is based on satisfying Branch Technical Position EICSB 18. This position basically states that it is acceptable to disconnect electric power to a safety-related valve as means of removing the possibility of an active failure of that valve.

Regulatory Guide 1.75 provides guidance for satisfying the acceptance criteria with respect to the identification of power and signal cables, cable trays, and instrument panels related to systems required for safe shutdown. The criteria for identification and separation of redundant systems discussed in Regulatory Guide 1.75 are presented in sufficient detail to make their application self-explanatory.

GDC 1, 21, 22, and 23, IEEE Std 279, IEEE Std 336, and Regulatory Guides 1.22, 1.47, and 1.68 provide the requirements that the design of systems required for safe shutdown must meet with regard to preoperational and periodic inservice testing. The primary review responsibility for the preoperational testing is with the QAB. Periodic testing and downtime restrictions are specified in the technical specifications. The review procedures for technical specifications are covered in SRP 7.1.

Another important area to be reviewed is the remote or local control stations that are required by GDC 19 for the safe shutdown of the plant in case the main control room becomes uninhabitable. Plant designs should provide for control stations in locations removed from the main control room that may be used for manual control and alignment operations needed to achieve and maintain a hot shutdown and subsequently to be able to achieve a cold shutdown. Equipment required for safe shutdown should be operable from local control panels. Access to these local control panels should be under strict administrative controls. The design of these control stations should provide appropriate readouts so that the operator can monitor the status of the shutdown. Typical readouts are steam generator level, steam generator pressure, pressurizer pressure, pressurizer level, and auxiliary feedwater flow.

The remote control stations and the equipment used to maintain safe shutdown should be designed to accommodate a single failure. Equipment located at these stations which is required for safe shutdown should be capable of operating independently (without interaction) of the equipment in the main control room. The design should be such as to prevent a single failure in the main control room or the cable spreading room from defeating the capability for affecting safe shutdown from the remote control stations, and vice versa. The remote control station equipment should be designed to the same standards as the corresponding equipment in the main control room, including appropriate IEEE criteria. Control transfer devices should be located away from the main control room and cable spreading areas, and their actuation should cause an alarm in the control room.

An important part of the review is the engineering drawing review at the OL stage. The drawing review should:

1. Verify that a complete set of drawings has been submitted that includes logic diagrams, P&IDs, and location layout drawings for these systems.

2. Verify that the submitted drawings represent the actual system designs and layouts for the particular plant, and that those drawings submitted as "typical" of a system are so identified.

3. Verify that the design and layout meet the applicable criteria listed under Section II of this plan.

The environmental qualification of components and cabling of systems required for safe shutdown should be the same as for the ESF systems discussed in SRP 7.3.

An applicant may choose to take exceptions to some of the acceptance criteria in the branch technical positions, guides, IEEE standards (other than IEEE Std 279, which is a mandatory requirement) and propose alternate ways of meeting the GDC requirements (which are mandatory). Any exceptions to the criteria are evaluated on an individual case basis. Exceptions are judged on the basis of the proposed design providing an equivalent level of safety and conservatism.

In general, the applicant will have design criteria that supplement or clarify the mandatory criteria. In the evaluation of such criteria, the reviewer can use the guidance listed above to determine whether the applicant's design criteria are adequate.

For the purpose of the EICSB review, no distinction should be made between the design criteria for systems required for safe shutdown and the criteria for the instrumentation and controls for essential auxiliary supporting systems.

Certain system designs and design features are submitted on a generic basis in the form of topical reports. Reference to a topical report is an acceptable alternative to submitting information in an application for a CP or an OL. Generally, topical reports pertain to standardized systems and qualification tests. If a referenced topical report has been accepted after staff review, the subjects of the report should not be reviewed again in connection with a particular application. If the referenced topical report has not been reviewed up to the time of the application review, it should be reviewed and treated in the same manner as the SAR itself. It may be necessary to assure conformance to requirements by getting additional information and justifications from the applicant. If the topical report has been rejected, then the applicant should be so advised and requested to submit information or design changes that are acceptable.

References other than topical reports should be obtained from the library or other sources, or the applicant asked to supply a copy.

A site visit and inspection should be performed before the evaluation findings are written for an OL. A site inspection should include a spot-check verification that the design and layout criteria reviewed during the drawing review are actually implemented at the hardware assembly stage. A site visit should be coordinated with the licensing project manager and the regional office that has jurisdiction over the geographic area in which the plant is located. Specific items to be considered include:

1. Separation and identification of redundant safety-related instrumentation channels, cabling, cable trays, and instrument rack terminations.

2. Separation of actuating switches in control panels for redundant safety-related equipment such as inboard and outboard isolation valves, coolant pumps, diesel-generator sets, etc.

3. Testability provisions and calibration procedures for instrumentation channels required for safe shutdown.

4. Adequacy of local control panels for remote shutdown, especially with regard to sufficient monitoring channels and actuating devices that the operators would need to perform and maintain a safe shutdown.

For a full outline of topics for a site visit, see Appendix 7-B to this Chapter.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been submitted and the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The review of systems required for safe shutdown includes the sensors, initiating circuitry, logic elements, interlocks, redundancy features, actuated devices, and auxiliaries that provide the instrumentation and control functions that prevent the reactor from returning to criticality and provide means for adequate residual heat removal from the core, containment, and other vital components and systems.

"The scope of review of systems required for safe shutdown for the plant included single line diagrams (CP and OL) and schematic diagrams (OL) and descriptive information for these systems and for auxiliary systems essential for their operation. The review has included the applicant's proposed design criteria, design bases, and analyses. The review has also included the applicant's analyses of the manner in which the design of these systems and their auxiliary supporting systems conform to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for systems required for safe shutdown and essential supporting auxiliaries to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"The staff concludes that the design of systems required for safe shutdown conforms to the applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (EICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

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SECTION 7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Reactor Systems Branch (RSB)
Core Performance Branch (CPB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

Information presented in the applicant's safety analysis report (SAR) is reviewed by the staff to determine that the design of safety-related display instrumentation (SRDI) required for safe functioning of the plant during operating and accident conditions is in conformance with applicable regulations, guides, branch technical positions, and industry standards and is consistent with the accident analysis assumptions of Chapter 15 of the SAR. For construction permit (CP) applications, the applicant's descriptive information for the SRDI should include commitments to meet applicable requirements and should present full justification for any exceptions taken.

For operating license (OL) reviews, the information presented should include the following:

1. Tables of system variables and components to be indicated and recorded (including accuracies and ranges of instruments).

2. Functional control diagrams or other means of illustrating the redundancy of monitored variable and component sensors and channels, the capability for sensor checks, and the means for verifying operability of monitoring system channels.

3. Electrical distribution diagrams illustrating electrical isolation of redundant sensors and channels.

4. Physical layout drawings illustrating separation of redundant indicating instruments.

5. Component and module quality and performance documentation, with particular emphasis on equipment used for post-accident monitoring.

6. Descriptions of the means for identifying redundant elements (such as cable, cable tray, component, module, and interconnecting wiring identifications).

7. Schematic and control panel display diagrams illustrating system level automatic bypass indication for deliberately bypassed safety-related components or systems.

Other EICSB areas of review associated with SRDI systems that are covered elsewhere are as follows:

1. Environmental design and qualification testing of electrical equipment are addressed in Standard Review Plan 3.11.

2. Technical specification requirements imposed upon the operation of the SRDI are discussed in Standard Review Plan 7.1.

The RSB identifies any changes or corrections to the listing of engineered safety feature and reactor coolant system variables and components that require indication, by examining the tables of SAR Section 7.5 that describe the information display (including accuracy and range requirements of indicating instrumentation) required by the operator to perform manual safety functions. The CPB identifies any changes or corrections to the listing of reactor variables that require indication, and the APCSB identifies "balance of plant" variables and components that require indication and, where necessary, states the required locations of the indicators.

The MEB reviews, in SAR Section 3.10, the criteria for seismic qualification and the test and analysis procedures and methods to assure the operability of the SRDI.

The QAB reviews, in SAR Chapter 17, the quality assurance procedures to be used by the applicant in the design, construction, installation, and maintenance of the SRDI.

II. ACCEPTANCE CRITERIA

The safety-related display instrumentation design is acceptable when it can be concluded that it conforms to the criteria listed in Table 7-1 and that the operator will be provided with sufficient information to perform required manual safety functions should such action be necessary. Specific points with regard to these criteria are detailed below.

1. The SRDI should cover appropriate variables, consistent with the assumptions for accident analyses and with the information needs of the operators in normal, transient, and accident conditions. The design of the SRDI should conform to the recommendations of Branch Technical Position EICSB 23. The accuracy and range of indicating instrumentation should be consistent with the assumptions of the accident analyses. Any exceptions to these requirements will be referred to the appropriate branch for resolution on an individual case basis.

2. All monitoring channels should be redundant, to assure that wrong indication due to device malfunction will not cause false action or inaction on the part of the operator. Identification malfunctions can be identified by cross checking between redundant channels.

3. Redundant channels of indicating instrumentation should be isolated physically and electrically to assure that a single failure will not result in complete loss of information about a monitored variable. Single failures might include such possible faults as shorting or opening circuits or interconnecting signal or power cables. It also includes single credible malfunctions or events that might cause a number of subsequent component, module, or channel failures. The post-accident SRDI should be capable of operating from onsite power. If signals from the post-accident monitoring equipment are used for control, the required isolation devices will be classified as part of the post-accident monitoring instrumentation. No credible failure at the output of an isolation device should prevent the associated monitoring channel from meeting minimum performance requirements considered in the design bases.

4. Capability should be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation. An acceptable way of accomplishing this would be by:
   a. Perturbating the monitored variable and observing the resulting indications.
   b. Introducing and varying a substitute input to the sensor of the same nature as the measured variable.
   c. Cross checking between channels that bear a known relationship to each other and that have readouts available.

   For channels which monitor a normally static parameter, provisions should be made to allow periodic testing in accordance with Regulatory Guide 1.22, thereby verifying channel operability.

5. An indication system should be provided covering bypassed or deliberately inoperable conditions of safety systems. Guidelines for the indication system are provided in Regulatory Guide 1.47 and Branch Technical Position EICSB 21.

6. Cables, cable trays, components, modules, and interconnecting wiring should be identified. The method used for identification and the scheme used to distinguish between redundant cables, cable trays, components, modules, and interconnecting wiring are acceptable if they are in accordance with the recommendations of Regulatory Guide 1.75.

7. Components and modules should be of a quality consistent with the reliability requirements for safety-related systems. An acceptable quality would be that of components and modules that have been previously used in similar service conditions and have demonstrated low maintenance requirements and failure rates. Other means to demonstrate acceptable quality would be through analysis and testing of components and modules, in accordance with criteria cited in Table 7-1.

8. In order to assure that the requirements of General Design Criterion 1, "Quality Standards and Records," are met in the SRDI, the quality assurance program must satisfy the requirements of IEEE Std 336-1971, as amplified by Regulatory Guide 1.30.

9. For those areas of review identified in Section I of this plan as being the responsibility of other branches, the acceptance criteria are included in the applicable sections of the review plans of those branches.

III. REVIEW PROCEDURES

The objectives in the review of the SRDI are to determine that the plant display instrumentation is designed, constructed, and installed in accordance with the design criteria outlined in Section II of this plan. In the CP review, the descriptive information, including the design bases and their relation to the criteria, preliminary analyses, piping and instrumentation diagrams (P&ID's), functional control diagrams, preliminary electrical diagrams, and preliminary physical arrangement drawings are examined to determine that there is reasonable assurance that the final implementation will meet all criteria. At the OL stage, the objectives are verified by review of the tables of variables and components to be monitored, indicated, and recorded; functional control diagrams, P&ID's, and electrical distribution diagrams; physical layout drawings; component and module quality considerations; the identification scheme for redundant systems; and the procedures for maintenance and checking of the availability of each system.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

The review steps are as follows:

1. Based on information provided by the RSB, CPB, CSB, and APCSB with regard to variables that need to be monitored and on Branch Technical Position EICSB 21, the list of monitored variables (if available) in the SAR is checked for sufficiency. In addition, the accuracy and range of the monitors are checked against the plant accident analyses as noted in II.1, above.

2. Functional control diagrams and P&ID's are reviewed to establish that the redundancy is sufficient, so that false indication due to malfunction of an indicating device should not lead to an undesirable manual action. In reviewing the P&ID's, the reviewer verifies that redundant sensors for each monitored variable are identified. After establishing sensor redundancy, the functional control diagrams are reviewed to ascertain that redundancy is maintained through the system logic down to the indicating devices.

3. Since independence from offsite power is required for post-accident SRDI, emphasis is placed on the electrical distribution system supplying power to post-accident SRDI.

Electrical distribution diagrams are reviewed to establish that redundant instrument channels are supplied from redundant electrical distribution channels of the emergency power supply. In addition, electrical schematic diagrams (as appropriate) are reviewed to ascertain that there is no interconnecting wiring between redundant channels whose failure (open or short circuit) could cause the simultaneous loss of redundant channels. Also, through the schematic diagrams, the reviewer ascertains that devices that isolate signals used for both safety indication and control are properly identified as part of the safety system and that a failure at the output of the isolation device does not prevent the associated monitoring channel from performing its safety function. Qualification of isolation devices is covered in the review of Sections 3.10 and 3.11 of the SAR.

4. Physical layout drawings (such as control room panel layouts, local panel layouts, sensor locations, instrument cabinet layout drawings, penetration drawings, and cable routing drawings) are reviewed to establish that physical independence is maintained between redundant channels of the SRDI. The control room panel layout drawings are examined to determine that the minimum separation distance between redundant equipment and circuits internal to the control boards is in accordance with Section 5.6 of Regulatory Guide 1.75. Local panel layout drawings are examined on the same basis. Sensor location drawings are examined to determine that the connections to the process system are sufficiently separated, in accordance with Section 5.8 of Regulatory Guide 1.75, to assure functional capability despite any single design basis event. The separation recommendations of Section 5.6 of this guide also apply to instrument cabinets (the layout drawings are examined to determine that the minimum separation distance between redundant equipment and circuits internal to the cabinets is provided). The procedure for review of penetration drawings and cable routing drawings is discussed in SRP 8.3.

5. With regard to the quality of components, there are at present no specific criteria to judge the quality of equipment used in the SRDI. However, Appendix B to 10 CFR Part 50 provides some guidance from which a judgment may be made of the quality of equipment required for the SRDI.

6. The procedure for reviewing the identification scheme proposed by the applicant to distinguish between redundant reactor protection system elements (including SRDI) is described in SRP 7.3.

7. The applicant's final design and installation of the SRDI is examined (schematic diagrams, wiring diagrams, installation drawings, etc.) to determine that the system includes the capability of periodic tests or checks to assure availability during operation.

IV. EVALUATION FINDINGS

The reviewer confirms that sufficient information has been provided and the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The safety-related display instrumentation provides the operator with information on the status of the plant to allow manual safety actions to be performed whenever necessary. The scope of review of safety-related display instrumentation included tables of system variables and component states to be indicated, functional control diagrams (CP and OL), electrical and physical layout drawings (OL), and descriptive information. The review has included the applicant's proposed design criteria and design bases, including that for indication of bypassed or inoperable safety-related systems. The review also has included the applicant's analyses of the manner in which the design of safety-related display instrumentation conforms to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for safety-related display instrumentation to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 7-1.

"The staff concludes that the design of safety-related display instrumentation for the ________________ plant conforms to applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (EICSB)."

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SECTION 7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Core Performance Branch (CPB)
Reactor Systems Branch (RSB)
Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The group of instrumentation systems reviewed under this plan are those required for safety that are not identified as part of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. They consist to a large extent of groups of interlocks intended to protect other vital systems from potentially damaging transients during normal operation and under accident conditions. Examples of such systems are cold water interlocks, refueling interlocks, interlocks that prevent overpressurization of low pressure systems, reactor vessel instrumentation, and accumulator valve interlocks. They also include the process and effluent radiological monitors which should be reviewed for the adequacy of their seismic design, redundancy and emergency power (See SRP 11.5).

The review of these systems encompasses the sensors, initiating circuits, logic elements, bypasses, interlocks, redundancy and diversity features, actuated devices, testing provisions, and equipment qualifications.

The EICSB has primary responsibility for the review of these systems. The review should confirm that these systems and essential supporting systems will perform design functions when required during all applicable operational and emergency conditions of the plant, and that the design of these systems conforms to all applicable acceptance criteria.

The descriptive information contained in the applicant's safety analysis report (SAR), including single line diagrams, electrical schematics, piping and instrumentation diagrams (P&IDs), and physical arrangement diagrams, is reviewed to ascertain that "other instrumentation systems required for safety" meet the acceptance criteria discussed in

Standard Review Plan (SRP) 7.1 and listed in Table 7-1. For a construction permit (CP) review, a commitment to meet these criteria can suffice in cases where the design of these systems has not been completed. For an operating license (OL) review, however, the actual design must be found to meet these criteria.

As a part of the primary review responsibility of the EICSB, it should be verified that:

1. The necessary redundancy of power sources, logic, and instrumentation are provided for the operation and status monitoring of "other instrumentation systems required for safety." This requires the review of the descriptive information contained in the SAR, functional diagrams, electrical schematics, and P&IDs.

2. The "other instrumentation systems required for safety" can perform necessary functions after sustaining a single failure. This requires review of the information as in (1) above, together with the drawings showing the physical layout of the electrical and instrumentation equipment and cabling. The review also involves verification that the design criteria for physical separation of redundant electrical equipment and cabling are acceptable, the design criteria for providing control and motive power to these systems are acceptable, and the single failure criterion has been included in the design considerations for manually-controlled electrically-operated valves.

3. The instrumentation and electrical equipment, cabling, cable trays related to, and structures housing parts of "other instrumentation systems required for safety" are designed in accordance with criteria required for Class IE and seismic Category I systems and structures, respectively. Also, proper identification of equipment, cabling, and cable trays to include color-coding in addition to alphanumeric markings is verified.

4. Environmental qualification of the electrical and instrumentation equipment and cabling has been established by tests and analyses showing that the equipment involved can perform needed safety-related functions in environments that may develop as a result of design basis accidents or anticipated operational occurrences.

   It should be established that the seismic qualification program is acceptable to the MEB as discussed in SRP 7.1 and later in this Section. It should be verified that all electrical and instrumentation equipment of "other instrumentation systems required for safety" have been included in the seismic qualification program.

5. On-line testability of the systems and indication of bypassed or inoperable status of the systems required for safety are provided.

The APCSB should evaluate the adequacy of those auxiliary systems required for the proper operation of "other instrumentation systems required for safety." These include compressed air systems, air conditioning systems, heat tracing systems, etc. In addition, the APCSB should review the physical arrangement of components and structures related to "other instrumentation systems required for safety" and supporting systems, and determine that single events will not disable redundant parts of these systems. The CPB will verify that boron dilution rates achievable, or the accidental startup of an unborated or cold reactor coolant loop, result in acceptable reactivity insertion rates as discussed in SRP 4.3.

The CSB should review the containment ventilation and atmosphere control systems provided to maintain environmental conditions required for operation of electrical and instrumentation equipment associated with "other instrumentation systems required for safety" and located inside containment.

The MEB review should confirm that the seismic qualification of the instrumentation and electrical systems is acceptable. This should include the seismic design criteria, analyses, testing procedures, and restraint measures employed in the seismic design and installation of Category I instrumentation and electrical equipment including trays, control room boards, and instrument racks and panels, as covered in SRP 3.10.

The RSB review should identify "other instrumentation systems required for safety" and confirm that the configuration and design bases of the systems are correct, and that design parameters such as temperature, pressure, flow rate, and reactivity can be controlled within acceptable limits. Information should be provided to the EICSB as to any corrections needed in the SAR and any exceptions to acceptance criteria taken by the applicant.

The QAB review should verify that the quality assurance program proposed by the applicant includes "other instrumentation systems required for safety."

II. ACCEPTANCE CRITERIA

The design, materials, qualification testing, and surveillance of "other instrumentation systems required for safety" are covered by several general design criteria (GDC), IEEE standards, regulatory guides, and branch technical positions which are applicable in whole or in part. A list of the applicable criteria, standards, guides, and branch positions is given in Table 7-1 and Appendix 7-A to this chapter.

The "other instrumentation systems required for safety" are acceptable when it is determined that these systems satisfy the following requirements:

1. They have the required redundancy.

2. They meet the single failure criterion.

3. They have the required capacity and reliability to perform intended safety functions on demand.

4. They are capable of functioning during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

5. They are testable during reactor operation.

The criteria listed in Table 7-1 are utilized as the bases for determining that these requirements are met and that the "other instrumentation systems required for safety" are acceptable. How these criteria are applied during the review process is discussed in Section III of this plan. Specific points with regard to the acceptance criteria are detailed below.

1. <u>System Redundancy Requirements</u>
   GDC 26 and 33 and IEEE Std 279 specify the requirements that "other instrumentation systems required for safety," among others, must meet with regard to all operating conditions (such as loss of offsite power), so that they can perform needed safety functions assuming a single failure. If a determination is made that these systems meet the requirements of these criteria, they are acceptable with regard to redundancy requirements.

2. <u>Conformance With the Single Failure Criterion</u>
   IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 provide that safety systems should be capable of performing needed safety functions after sustaining a single failure. Regarding the application of the single failure criterion to the design of manually-controlled electrically-operated valves in safety systems, the acceptability of proposed designs is based on Branch Technical Position EICSB 18. This position states that it is acceptable to disconnect electric power to a safety-related valve as means of designing against an active valve malfunction.

3. <u>Identification of Cables and Cable Trays</u>
   The method used for identifying power and signal cables and cable trays as safety-related equipment, and the identification scheme used to distinguish between redundant cables, cable trays, and instrument panels should be in accordance with the recommendations of Regulatory Guide 1.75.

4. <u>Vital Supporting Systems</u>
   The instrumentation, control, and electric equipment associated with auxiliary systems that support "other systems required for safety" should meet the same acceptance criteria as the systems they support.

5. <u>Testing, Quality Assurance, and System Availability Surveillance</u>
   GDC 1 and 21; IEEE Stds 279, 336, and 338; and Regulatory Guides 1.22, 1.47, and 1.68 contain the applicable acceptance criteria with regard to preoperational and periodic testing, quality assurance, and design provisions for indicating the availability of "other instrumentation systems required for safety."

For the areas of review identified in Section I as review responsibilities of other branches, the acceptance criteria are included in the corresponding standard review plans.

III. REVIEW PROCEDURES

The review is conducted to ascertain that the designs of "other instrumentation systems required for safety" (or design commitments in the case of CP's) are acceptable in terms of the acceptance criteria listed in Section II. The main objectives of the review of these systems are to determine that they include the required redundancy, meet the single failure criterion, provide the required capacity and reliability to perform intended safety functions on demand, and can function during and after certain design basis events such as earthquakes, accidents, and anticipated operational occurrences.

For a CP application, the descriptive information contained in the preliminary safety analysis report (PSAR), including the design bases and their justification with regard to the acceptance criteria, accident analyses, electrical single line and P&ID's, are reviewed to determine that the basic design features and the commitments made at this stage provide assurance that the final design will meet the acceptance criteria. During the OL review, it is verified that the acceptance criteria are met through review of the final electrical and instrumentation drawings and the physical layout drawings, and a site visit during which a spot-check verification of the design is performed.

The various elements of the review are carried out as follows:

1. The descriptive information in the SAR, including the electrical one-line and P&ID's (for CP and OL reviews), and electrical schematics (for the OL review), is reviewed to verify that the necessary redundancy is provided. This review includes instrumentation channels used to sense vital parameters such as temperature, pressure, water level, etc., the associated logic and actuated devices, and the motive and control sources.

2. Conformance with the single failure criterion as specified by IEEE Std 279, IEEE Std 379, and Regulatory Guide 1.53 is verified by review of the same information as for redundancy and may be done, to some degree by necessity, at the same time. The guidance provided by Regulatory Guide 1.53 is excellent for ascertaining that a given design is single failure proof. A particularly important point to check is one cited in Position 4 of Regulatory Guide 1.53, where a single d-c source supplies control power for one channel of system logic and for the redundant actuator circuit.

3. For a multi-unit design where electrical systems are shared, resulting in more and complex interaction modes, a fault-tree and decision-tree analysis may be requested from the applicant to show that single failures, or single events resulting in multiple failures, will not result in unacceptable consequences with respect to the capability of "other instrumentation systems required for safety" to perform safety functions when required. Additional guidance with regard to the single failure criterion is given in SRP 7.2 and 7.3.

4. For manually-controlled electrically-operated valves in safety-related systems, the acceptability of proposed designs is based on Branch Technical Position EICSB 18. This position basically states that it is acceptable to disconnect electric power to a safety-related valve as means of removing the possibility of an active failure of that valve.

5. Regulatory Guide 1.75, and more specifically, Sections 5.1.2 and 5.6.3 provide guidance for satisfying the acceptance criteria with respect to the identification of power and signal cables, cable trays, and instrument panels related to "other instrumentation systems required for safety." The criteria for identification and separation of redundant systems as discussed in Regulatory Guide 1.75 are presented in sufficient detail to make their application self-explanatory. GDC 1 and 21; IEEE Stds 279, 336 and 338; and Regulatory Guides 1.22, 147, and 1.68 provide the requirements that the design of these systems must meet with regard to preoperational and periodic testing. The primary review responsiblity for preoperational testing is with the QAB. Periodic and downtime restrictions are specified in the technical specifications. The review procedures for technical specifications are covered in SRP 7.1.

6. The process of aligning various systems for certain modes of operation may involve the interconnection of high pressure and low pressure systems. During normal operation, these systems must be isolated from one another. For example, the residual heat removal (RHR) system of some reactor designs is interfaced with the high pressure reactor coolant system. There should be two isolation valves in series, with diverse interlocks that will prevent operation of these valves unless the primary reactor coolant pressure is below a predetermined value. For a detailed description of the isolation requirements, see Branch Technical Position EICSB 3.

7. The main steam line radiation monitoring system in boiling water reactors is provided to monitor the gross release of fission products in the reactor coolant and initiate protective action if the level of such release exceeds a predetermined level. The reviewer should assure that the instrumentation channels provided for this purpose are divided into two redundant and independent groups. Also, the two groups should be powered from independent power channels of the emergency power system.

   Normally, four gamma-sensitive channels are provided to monitor the radiation level in the main steam lines. The reviewer should assure that the geometric arrangement and physical location of these is such that a fission product release will be detected with any number of main steam lines in operation, and that it will be detected at the earliest possible time following a fuel failure. It is important that the failure of any one of these four channels will not result in an inadvertent action. The initiating logic should be checked to make sure that this is the case. The reviewer should verify that the design has provisions for testing and that operability can be adequately tested.

8. The reviewer should verify that the "other instrumentation systems required for safety" have been qualified to operate under normal, operational transient, accident,

and post-accident environmental conditions and that they satisfy the recommendations of IEEE Std 323. The reviewer also verifies that equipment and structures related to these systems are seismically qualified or designed, and the seismic qualification and analysis program submitted by the applicant is acceptable to the MEB and EICSB. The environmental qualification of components and cabling of these systems should be the same as for the systems discussed in SRP 7.3 and 3.11.

9. An important part of the review is the engineering drawing review. A drawing review should include the following:

   a. Verification that a complete set of drawings has been submitted that includes logic diagrams, P&ID's, and location layout drawings for these systems.

   b. Verification that the submitted drawings represent the actual system designs and layouts for the particular plant, and that those intended to be "typical" of a system are so identified.

   c. Verification that the design and layout meet the applicable criteria listed in Section II of this plan.

10. A site visit and inspection should be performed before the evaluation findings are written for OL reviews. A site inspection should include spot-check verifications that the design and layout criteria are actually implemented at the hardware assembly stage. A site visit should be coordinated with the licensing project manager and the regional office that has jurisdiction over the plant. Items to investigate during the visit include:

   a. Separation and identification of redundant safety-related instrumentation channels, cabling, cable trays, and instrument rack terminations.

   b. Separation of actuating switches in control panels for redundant safety-related equipment such as inboard and outboard isolation valves, coolant pumps, diesel-generator sets, etc.

   c. Testing provisions and calibration procedures for instrumentation channels required for safety.

See Appendix 7-B to this chapter for a complete outline of items to be covered in site visits.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

EICSB verifies that sufficient information has been submitted and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The other instrumentation systems required for safety" consist of safety-related instrumentation systems not identified as parts of the reactor protection system, engineered safety features systems, safety-related display instrumentation systems, or systems required for safe shutdown. They are, to a large extent, groups of interlocks intended to protect other vital systems from potentially damaging transients during normal operating and accident conditions.

"Their review encompasses the sensors, initiating units, logic, bypasses, interlocks, redundancy and diversity features, actuated devices, testing provisions, and equipment qualifications. The review includes single line diagrams (CP & OL), schematic diagrams (OL), and descriptive information on this group of systems and supporting auxiliaries that are essential for their operation. The review has included the applicant's proposed design criteria and design bases and analyses of the manner in which the design of these systems conform to the proposed design criteria and are adequate.

"The basis for acceptance in the staff review of these systems has been conformance of the applicant's designs, design criteria, 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 standards. These are listed in Table 7-1.

"The staff concludes that the design of these systems conforms to applicable regulations, guides, technical positions, and industry standards, and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

2. Standard Review Plan Appendix 7-A, "Branch Technical Positions (EICSB)."

3. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

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SECTION 7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The areas reviewed in this section of the applicant's safety analysis report (SAR) include such control systems as the primary system pressure, temperature and water level controls, feedwater controls, and main turbine controls. The intent of the review is to assure that failures of these would not impair the protection system capability in any significant manner. Since the control systems of interest may vary from plant to plant depending on individual designs, the applicant should identify all such systems and provide analyses to support their classification as non-safety-related control systems.

The EICSB will review the following aspects of the non-safety-related control systems: the circuit-to-circuit failure modes of a single non-safety control system and their effect on the protection system, and gross failure modes of non-safety control systems and their functional effect on the protection system.

The APCSB and RSB provide assistance in verifying that all control systems have been identified and that the input signal parameters for the control systems are correct. The RSB determines that the control systems identified in this section are not required for safety and that no credit is taken in the plant accident analyses for the control systems identified as non-safety in this section.

The QAB verifies that the quality assurance program implemented for control system components, where necessary, is adequate.

II. ACCEPTANCE CRITERIA

The control systems not required for safety are acceptable if failures of control system components or total systems would not significantly affect the ability of plant safety systems to function as required, or cause plant conditions more severe than those for which the plant safety systems are designed.

Table 7-1 of this plan lists ghose deneral cesign Criteria (GDC) of Appendix A to 10 CFR Part 50, and standards of the Institute of Electrical and Electronic Engineers (IEEE) that are used as references in arriving at this conclusion. GDC 13 and 24 and IEEE Std 279, Section 4.7, are of special importance among these references.

1. Conformance with GDC 13 for Instrumentation and Control Requirements. Instrumentation should be provided to monitor variables and systems over their anticipated ranges for normal operation and for anticipated operational occurrences as appropriate to minimize challenges to safety systems. Appropriate controls should be provided to maintain these variables and systems within prescribed operating ranges.

2. Conformance with GDC 24 for Separation of Control Systems from Protection Systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel which is common to control and protection systems shall not violate the reliability, redundancy, and independence requirements of the protection system. The interconnections between the protection and control system shall be limited so as to assure that safety is not significantly impaired.

3. Conformance to IEEE Std 279, Section 4.7 for Control and Protection System Interaction. The direct circuit-to-circuit and functional interactions between control and protection systems for single random or multiple failures in the control system shall not prevent the protection system channel from meeting the minimum performance requirements specified in the design bases.

III. REVIEW PROCEDURES

1. The objectives in the review are:
   a. To establish that control systems identified as being non-safety-related, which may include, depending on plant design, the primary system pressure, temperature, and feedwater controls, steam generator water level controls, and main turbine controls are, in fact, not required for plant safety.
   b. To verify that no credit is taken for the operability of these control systems in the plant accident analyses in Chapter 15 of the SAR.
   c. To assure that failures of these control systems would not impair the capability of the protection system in any significant manner or cause plant conditions more severe than those for which the plant safety systems are designed.
   d. To establish that control system designs meet applicable requirements of the general design criteria and industry standards with regard to independence between control and protection functions.

2. In the construction permit (CP) review the descriptive information including the design bases and preliminary analyses, are reviewed to determine that there is reasonable assurance that the final design will meet these objectives. The RSB and APCSB identify the systems whose control system designs are to be reviewed and verify that no credit is taken for their operability in the plant accident analyses. EICSB reviews the descriptive information provided for those systems

at the CP stage to assure that control and protective functions are adequately separated and to assess the effects of control system failures, or to verify that commitments are made that such failures will be included in the plant safety design bases.

3. At the operating license (OL) stage, the objectives in (1) above are verified during the review of control system schematics. At the OL stage, EICSB reviews electrical schematic drawings for these control systems as necessary to assure that adequate attention has been given to the separation of control and protective functions and to possible effects of failures of these systems. The review includes interactions between control systems and effects on plant operation and safety systems due to control system malfunctions or failures.

4. A typical review procedure for pressurized water reactor (PWR) primary and secondary control system functions follows:

   a. The primary system pressure is maintained within specified limits by the use of pressurizer heaters and spray valves. The primary pressure control system description and schematics are reviewed:
      (1) To confirm that the system will maintain the primary coolant pressures within prescribed limits for normal and transient operating conditions.
      (2) To determine the effects of loss of power to the pressurizer heaters and spray valves.
      (3) To determine the effects of loss of air to any pneumatically-operated valves in the spray system.

      Assistance as needed is obtained from the RSB in evaluating these items.
   b. To meet the requirements of GDC 24 and Section 4.7 of IEEE Std 279 on control system interactions with the protection system, loss of primary pressure control function is analyzed. Assistance is obtained from RSB in establishing the sequence of events that would follow. The evaluation should show that failure of the primary pressure control system would not significantly degrade the capability of the protection system. Also, the reviewer determines that where a random failure in the pressure control system results in a plant condition requiring protective action and can also prevent proper action of a protection channel designed to protect against the condition, the remaining redundant channels will provide the protective action even when degraded by another random failure.
   c. The system description and control schematics of the feedwater regulating system are reviewed for failure modes of the system components. Assistance is obtained from the RSB and APCSB in identifying the control function parameters. The system actions are established for loss of air to the feedwater control valves and malfunction in the feedwater heater bypass valves. The reviewer should verify that manual override of the automatic control is designed into the system.
   d. The reviewer evaluates the effects of multiple failures in control systems resulting from single events. Failures in the secondary system water level

(i.e., feedwater flow and steam generator water level) controls are analyzed along with failure in the primary coolant pressure control, where a single event can cause these multiple failures. With the assistance from the RSB and APCSB the reviewer determines that control function failures of both primary pressure and secondary water level controls would not prevent the minimum required number of reactor protection system channels from tripping the reactor.

5. The following aspects of main turbine control systems are reviewed:
   a. The reviewer verifies that the turbine overspeed protection system is designed with redundant speed sensing instrumentation and logic circuitry, so as to ensure that no single failure would prevent the overspeed trip system from operating. The overspeed trip system should have the capability to permit online testing of its instrumentation and logic circuitry when the turbine is in operation.
   b. The controls that provide for automatic turbine runback on receipt of appropriate signals from the reactor systems are reviewed for the following points:
      (1) The signals should be redundant, with independent power supplies.
      (2) Physical independence should be maintained between redundant initiating circuits.
      (3) Although redundancy is not practical in the final device, the signals should actuate different control devices.
      (4) The final actuating device should be of high reliability.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

IV. EVALUATION FINDINGS

At the CP stage, it should be established that the information and commitments documented in the preliminary safety analysis report (PSAR) provide reasonable assurance that the final designs of non-safety-related control systems will conform with the intent of this plan.

At the OL stage, sufficient design detail of these control systems is reviewed to determine adequate conformance. Exceptions to the acceptance basis given in Section II are identified, with a statement as to how these exceptions provide a conservative basis for engineering design of the affected control systems.

The reviewer verifies that sufficient information has been submitted and the review supports conclusions of the following type, to be included in the staff's evaluation report:

"The staff has reviewed the controls for systems not required for safety, to determine the affects of failures or malfunctions of these controls on the reactor protection system and other plant safety-related systems. We conclude that failures or malfunctions of these controls would not be expected to degrade the capabilities of plant safety systems in any significant degree, or to lead to plant conditions more severe than those for which the safety systems are designed."

V. REFERENCES

1. Standard Review Plan Table 7-1, "Acceptance Criteria for Controls."

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APPENDIX 7-A BRANCH TECHNICAL POSITIONS (EICSB)

The EICSB Branch Technical Positions (BTP's) represent guidelines intended to supplement the acceptance criteria established in Commission regulations and regulatory guides, and in applicable IEEE standards. The BTP's originate in technical problems or questions of interpretation that arise in the detailed reviews of plant designs. The staff must make a judgement in each such case, in order to complete its review of the particular application. Where the same technical problem or question of interpretation arises in several cases, the staff's judgement on the point at issue is formalized in a BTP. The BTP is primarily an instruction to staff reviewers that outlines an acceptable approach to the particular issue and ensures a uniform treatment of the issue by staff reviewers. The approaches taken in the BTP's, like the recommendations of regulatory guides, are not mandatory, but do provide defined, acceptable, and immediate solutions to some of the technical problems and questions of interpretation that arise in the review process. In some instances, regulatory guides may be developed from BTP's after a sufficient experience in their use has accumulated.

All EICSB BTP's applicable to Chapters 7 and 8 have been collected in this Appendix for convenience. They are listed below:

BTP EICSB	Branch Technical Positions of the Electrical, Instrumentation and Control Systems Branch
1.	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors.
2.	Diesel-Generator Reliability Qualification Testing.
3.	Isolation of Low Pressure Systems From the High Pressure Reactor Coolant System.
4.	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines.
5.	Scram Breaker Test Requirements - Technical Specifications.

6. Capacity Test Requirements of Station Batteries - Technical Specifications.

7. Shared Onsite Emergency Electric Power Systems for Multi-Unit Generating Stations.

8. Use of Diesel-Generator Sets for Peaking.

9. Definition and Use of "Channel Calibration" - Technical Specifications.

10. Electrical and Mechanical Equipment Seismic Qualification Program.

11. Stability of Offsite Power Systems.

12. Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service.

13. Design Criteria for Auxiliary Feedwater Systems.

14. Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors.

15. Reactor Coolant Pump Breaker Qualification.

16. Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors.

17. Diesel-Generator Protective Trip Circuit Bypasses.

18. Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves.

19. Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems.

20. Design of Instrumentation and Controls Provided to Accomplish Change-over from Injection to Recirculation Mode.

21. Guidance for Application of Regulatory Guide 1.47.

22. Guidance for Application of Regulatory Guide 1.22.

23. Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown.

24. Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times.

25. Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole.

26. Requirements for Reactor Protection System Anticipatory Trips.

27. Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves.

BRANCH TECHNICAL POSITION EICSB 1

BACKFITTING OF THE PROTECTION AND EMERGENCY POWER SYSTEMS OF NUCLEAR REACTORS

A. BACKGROUND

The acceptance criteria used by the staff in the evaluation of protection and emergency power systems undergo improvement from time to time. With each change it is necessary to determine whether previously approved designs should be modified (backfitted) to meet the revised criteria. This determination is made on the basis of whether a significant incremental increase in safety of the plant would be obtained that would justify the various difficulties of the change.

The actions which raise the question of possible backfitting are:

1. Application for a full-term operating license for plants now operating with a provisional operating license.

2. Evaluation of a significant plant modification proposed by the staff or the licensee.

3. Application for a full-term operating license for plants now operating under DOD 91-B exemptions.

B. BRANCH TECHNICAL POSITION

For cases falling in the categories 1-3 in (A), above the following apply;

1. Instrumentation and electric equipment essential to safety which must function in an accident environment should be analyzed or tested to demonstrate this capability.

2. Protection circuits essential to safety should meet the single failure criterion of Section 4.2 of IEEE 279.

3. Where d-c power is required for safety, redundant d-c sources should be provided and the d-c circuits should meet the single failure criterion.

4. For reactor plants supplying electric power to electric utility grids, redundant sources of onsite a-c power should be provided and the a-c circuits should meet the single failure criterion. This aspect of the design of research and test reactors should be evaluated on an individual case basis.

C. REFERENCES

1. Note for P. A. Morris from E. G. Case, August 6, 1971.

BRANCH TECHNICAL POSITION EICSB 2

DIESEL-GENERATOR RELIABILITY QUALIFICATION TESTING

A. BACKGROUND

The increase in standby electrical generating capacity required for safety loads of the current large water-cooled power reactors has caused several applicants to propose standby power source design using diesel-generators or diesel-generator configurations not previously used. The staff concluded that qualification testing of these larger capacity machines or configurations would be required to demonstrate a capability and reliability at least equivalent to that of machines currently used for nuclear plant standby applications.

The proposals of nonstandard diesel-generator arrangements for Sequoyah, Fort St. Vrain, Hutchinson Island, and Fitzpatrick made it necessary to develop a consistent approach for determining acceptability. Regulatory Guides 1.6 and 1.9 were utilized as the bases.

B. BRANCH TECHNICAL POSITION

A start and load reliability test program should be required for all diesel-generator sets of a type or size not previously used as standby emergency power sources in nuclear power plant service. The objective of this program should be to establish a 0.99 reliability for starting and accepting design load in the desired time. An acceptable test program should include the following requirements:

1. At least two full-load and margin tests acceptable to the staff should be performed on each diesel-generator set to demonstrate the start and load capability of the units with some margin in excess of the design requirements. Proposed full-load and margin testing should be evaluated on an individual case basis to take account of the differences in unit design.

2. Prior to initial fuel loading, at least 300 valid start and load tests should be performed with no more than three failures allowed. At least 90% of these start tests shall be made from design cold ambient conditions (design hot standby conditions if standby temperature control system is provided) and 10% from design hot equilibrium temperature conditions. This would include all valid tests performed offsite. A valid start and load test shall be defined as a start from the specified temperature conditions with loading to at least 50% of continuous rating within the required time intervals, and continued operation until temperature equilibrium is attained.

3. A failure rate in excess of one per hundred should require further testing as well as review of the system design adequacy.

C. REFERENCES

1. Fort St. Vrain Safety Evaluation Report, May 1, 1971.

2. Zion 1 and 2 Safety Evaluation Report, March 10, 1972.

BRANCH TECHNICAL POSITION EICSB 3

ISOLATION OF LOW PRESSURE SYSTEMS FROM THE HIGH PRESSURE REACTOR COOLANT SYSTEM

A. BACKGROUND

During normal and emergency conditions, it is necessary to keep low pressure systems that are connected to the high pressure reactor coolant system properly isolated in order to avoid damage by overpressurization or the potential for loss of integrity of the low pressure system and possible radioactive releases. There have been a number of recommendations for accomplishing this aim. Until a more definitive guide is published, the criteria in in Part B, below, provide an adequate and acceptable design solution for this concern.

B. BRANCH TECHNICAL POSITION

The following measures should be incorporated in designs of the interfaces between low pressure systems and the high pressure reactor coolant system:

1. At least two valves in series should be provided to isolate any subsystem whenever the primary system pressure is above the pressure rating of the subsystem.

2. For system interfaces where both valves are motor-operated, the valves should have independent and diverse interlocks to prevent them from both being opened unless the primary system pressure is below the subsystem design pressure. Also, the valve operators should receive a signal to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

3. For those system interfaces where one check valve and one motor-operated valve are provided, the motor-operated valve should be interlocked to prevent the valve from opening whenever the primary pressure is above the subsystem design pressure, and to close automatically whenever the primary system pressure exceeds the subsystem design pressure.

4. Suitable valve position indication should be provided in the control room for the interface valves.

5. For those interfaces where the subsystem is required for ECCS operation, the above recommendations need not be implemented. System interfaces of this type should be evaluated on an individual case basis.

C. REFERENCES

1. Memorandum to E. G. Case from P. A. Morris, February 6, 1971.

2. Memorandum to P. A. Morris from D. Skovholt, February 19, 1971.

3. Note for E. G. Case from D. F. Knuth, April 13, 1972.

BRANCH TECHNICAL POSITON EICSB 4
REQUIREMENTS ON MOTOR-OPERATED VALVES IN THE ECCS ACCUMULATOR LINES

A. BACKGROUND

For many postulated loss-of-coolant accidents, the performance of the emergency core cooling system (ECCS) in pressurized water reactor plants depends upon proper functioning of the safety injection tanks (also referred to as "accumulators" or "flooding tanks" in some applications). In these plants, a motor-operated isolation valve (MOIV) and two check valves are provided in series between each safety injection tank and the reactor coolant (primary) system.

The MOIV's must be considered to be "operating bypasses" because, when closed, they prevent the safety injection tanks from performing the intended protective function. IEEE Std 279-1971 has a requirement for "operating bypasses" which states that the bypasses of a protective function will be removed automatically whenever permissive conditions are not met. This branch technical position provides specific guidance in meeting the intent of IEEE Std 279-1971 for safety injection tank MOIV's.

It should be noted that BTP EICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," also applies to these isolation valves and should be used in conjunction with this position.

B. BRANCH TECHNICAL POSITION

The following features should be incorporated in the design of MOIV systems for safety injection tanks to meet the intent of IEEE Std 279-1971:

1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the technical specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator.

2. Visual indication in the control room of the open or closed status of the valve.

3. An audible and visual alarm, independent of item (2), above, that is actuated by a sensor on the valve when the valve is not in the fully-open position.

4. Utilization of a safety injection signal to remove automatically (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the technical specifications).

C. REFERENCES

1. Memorandum to E. G. Case from P. A. Morris, February 10, 1971.

2. Arkansas 1, Unit 1, Safety Evaluation Report, January 23, 1973.

BRANCH TECHNICAL POSITION EICSB 5

SCRAM BREAKER TEST REQUIREMENTS - TECHNICAL SPECIFICATIONS

A. BACKGROUND

There have been some inconsistencies in the description of scram circuit test procedures in FSARs and technical specifications requirements. Some FSARs for plants with Westinghouse reactors describe the scram circuit test procedures and include a position for testing the scram breakers, but there are no provisions for doing so in the proposed technical specifications. It is the purpose of this branch technical position to establish a uniform practice in this matter.

B. BRANCH TECHNICAL POSITION

The requirement that control rod drive trip breakers be tested monthly should be included in all plant Technical Specifications issued. For a model, refer to the Oconne technical specifications page 4.1-4, Table 4.1-1, item 2.

C. REFERENCE

1. Memorandum to PWR Branch Chiefs from R. C. DeYoung, January 28, 1972.

BRANCH TECHNICAL POSITION EICSB 6

CAPACITY TEST REQUIREMENTS OF STATION BATTERIES - TECHNICAL SPECIFICATIONS

A. BACKGROUND

The capacity test requirements for station batteries are addressed in IEEE Std-450-1972 and IEEE Std-308 the 1971 and 1974 editions. The purpose of this branch technical position is to provide guidance for meeting the recommendations of these standards.

B. BRANCH TECHNICAL POSITION

All technical specifications shall include a requirement for periodic surveillance testing of onsite Class IE batteries. The test should meet the intent of Section 5.3.6 of IEEE Std 308-1971 to determine battery capacity including as a minimum the following requirements:

(1) An acceptance test of battery capacity shall be performed according to Section 4.1 of IEEE Std 450-1972.

(2) The performance discharge test listed in Table 2 of IEEE Std 308-1971 shall be performed according to Sections 4.2 and 5.4 of IEEE Std 450-1972.

(3) A battery service test, described in Section 5.6 of IEEE Std 450-1972, shall be performed during each refueling operation or at some other outage with intervals between tests not to exceed 18 months in order to satisfy Section 6.4 of IEEE Std 308-1971.

(4) A detailed description of the battery service test shall be included in Section 8.3 of the Safety Analysis Report.

C. REFERENCE

1. Memorandum to R. H. Vollmer from J. G. Keppler, March 20, 1972.

2. Memorandum to R. Carlson from V. D. Thomas, January 18, 1972.

BRANCH TECHNICAL POSITION EICSB 7

SHARED EMERGENCY ELECTRIC POWER SYSTEMS FOR MULTI-UNIT GENERATING STATIONS

A. BACKGROUND

The detailed operating license reviews of multi-unit stations using shared onsite power systems revealed that in almost every case sharing resulted in reduction in the number of and capacity of the onsite power sources to below that normally provided for the same number of units located at separate sites. This reduced capacity introduced a number of interactions that are potential safety problems. These interactions concern (1) the interconnection of ESF control circuits of each unit such that failures and maintenance or testing operations in one unit affect the availability of ESF in other units, (2) coordination between unit operators required in order to cope with an accident in one unit and safe shutdown of the remaining unit(s), and (3) system overload conditions as a consequence of a real accident in a unit coincident with a false or spurious accident signal in another unit. The purpose of this branch technical position is to provide guidance in assuring proper compliance with the requirements of General Design Criterion 5.

B. BRANCH TECHNICAL POSITION

1. For multi-unit generating stations now under design and construction and for which construction permit applications were made before May 1, 1973, the design of shared onsite emergency power systems should:

   a. Assure that a single failure, including a false or spurious accident signal, does not reduce the capability to supply automatically minimum engineered safety feature (ESF) loads in any unit and safely shut down the remaining units.

   b. Provide onsite power capacity sufficient to energize seismic Category I equipment to attain a safe and orderly cold shutdown of all units, assuming a single failure and loss of offsite power.

   c. Limit the interactions between unit engineered safety feature electrical circuits such that any allowable combination of maintenance and test operations in the units will not affect the capability to supply power automatically to minimum ESF loads in any unit.

   d. Minimize the coordination required between unit operators in order to accomplish (a), (b), and (c) above. Although each design will be evaluated on an individual basis in this regard, all shared onsite power systems should meet the following:

(1) Coordination between the unit operators should not be necessary in order to provide for (a) and (b), above.

(2) Complete information regarding the status of the shared system should be provided for each operator.

e. Conform with IEEE Std 308-1971 and Regulatory Guides 1.6 and 1.9.

2. The onsite emergency electrical power systems of multi-unit generating stations for which construction permit applications are made after May 1, 1973, should conform to the following criterion:

"Each unit shall have separate and independent onsite emergency electrical power systems, both a-c and d-c, capable of supplying minimum ESF loads and the loads required for achieving and maintaining a safe and orderly cold shutdown of the unit, assuming a single failure and loss of offsite power."

C. REFERENCES

1. Memorandum to V. Moore from V. Stello, August 24, 1973.

2. Memorandum to L. Rogers from J. F. O'Leary, August 25, 1972.

3. Memorandum to J. M. Hendrie from T. A. Ippolito, November 19, 1973.

4. Memorandum to G. A. Arlotto from V. Stello, December 10, 1973.

BRANCH TECHNICAL POSITION EICSB 8
USE OF DIESEL-GENERATOR SETS FOR PEAKING

A. BACKGROUND

General Design Criterion 17 requires that provisions be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with loss of the main generator, loss of power from the grid, or loss of standby power supplies. Additionally, IEEE Std 308 requires that the preferred (offsite) and standby power supplies shall not have a common failure mode. Common failure mode is defined as "a mechanism by which a single design basis event can cause redundant equipment to be inoperable." Although IEEE Std 308 does not preclude the use of emergency diesels for non-safety purposes, the staff concludes that the potential for common failure modes should preclude interconnection of onsite and offsite power sources except for short periods for load testing.

Review of the use of emergency diesel-generator sets for peaking service leads to the conclusion that the required frequent interconnection of the preferred and standby power supplies increases the probability of their common failure.

B. BRANCH TECHNICAL POSITION

General Design Criterion 17 and Section 5.2.1(5) of IEEE Std 308-1971 should be interpreted as prohibiting the use of plant emergency power diesel-generator sets for purposes other than that of supplying standby power when needed. In particular, emergency power diesel-generator sets should not be used for peaking service.

C. REFERENCES

1. Note to D. F. Knuth and V. A. Moore from J. M. Hendrie, January 23, 1973.
2. Memorandum to J. M. Hendrie and D. F. Knuth from V. A. Moore, January 4, 1973.

BRANCH TECHNICAL POSITION EICSB 9

DEFINITION AND USE OF "CHANNEL CALIBRATION" - TECHNICAL SPECIFICATIONS

A. BACKGROUND

In several PWR technical specifications, the term "channel calibration" was used to describe a "daily adjustment" for amplifier gain of the nuclear instrumentation power range channels. This adjustment was performed to maintain agreement between the indicated reactor nuclear power level and the reactor thermal power calculation. This adjustment is not considered by the staff to be a channel calibration. A calibration procedure performed on a monthly basis requires the following:

a. Performance of a functional test using a simulated signal to verify bistable action (protective trips including rod block trips and permissive interlocks) on a monthly basis.

b. Calibration of the upper and lower chambers of each flux channel for axial offset utilizing the in-core detectors on a calendar quarter basis.

c. Performance of a functional test using a simulated signal to verify positive and negative rate bistable action on a monthly basis.

Performance of a total system response time test is required during each refueling outage.

B. BRANCH TECHNICAL POSITION

The "daily adjustment," which does not fulfill the intent or requirements of a calibration procedure, should remain as a daily requirement but be deleted from the "channel calibration" category in the technical specifications.

C. REFERENCES

1. Memorandum to R. L. Tedesco from V. Stello, April 19, 1973.
2. Memorandum to R. C. DeYoung from R. L. Tedesco, April 27, 1973.

ELECTRICAL AND MECHANICAL EQUIPMENT SEISMIC QUALIFICATION PROGRAM

A. BACKGROUND

Subsequent to the publication of IEEE Std 344-1971, the staff determined that compliance with the standard was not in itself sufficient to assure an acceptable seismic qualification program for electrical and mechanical equipment. As a result, a supplement to IEEE Std 344-1971 was developed by the staff.

B. BRANCH TECHNICAL POSITION

1. For plants for which construction permit applications were docketed before October 27, 1972, and for which operating license reviews are not completed, information should be provided describing in detail the methods used for qualifying equipment under IEEE Std 344-1971, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

2. For plants for which construction permit applications are docketed after October 27, 1972, the following supplementary requirements to IEEE Std 344-1971 should be met:

   a. Seismic Test for Equipment Operability

      (1) A test and analysis program is required to confirm the functional operability of all seismic Category I electrical and mechanical equipment during and after a earthquake of magnitude up to and including the SSE. Analysis without testing may be acceptable only if structural integrity alone can assure the intended function. When a complete seismic testing is impracticable, a combination of tests and analyses may be acceptable.

      (2) The characteristics of the required input motion (i.e., the support motion in the seismic event) should be specified by one of the following:

         (a) Response spectrum.
         (b) Power spectral density function.
         (c) Time history.

      Such characteristics, as derived from the structures or systems seismic analyses, should be representative of the 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 input motion for the testing should be characterized in the same manner as the required input motion, and conservatism in amplitude and frequency content should be demonstrated. The frequency spectrum used should cover the range from 1 through 33 Hz. Any exceptions require justification.

(5) Seismic excitation generally has a broad frequency content. Random vibration input motion should be used. However, single frequency inputs, such as sine beats, may be applicable provided one of the following conditions are met:
   (a) The characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects).
   (b) The anticipated response of the equipment is adequately represented by one mode.
   (c) The test input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelop the corresponding seismic event response spectra of the individual modes.

(6) The input motion should be applied to the vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along 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.

(7) The fixture design should meet the following requirements:
   (a) Simulate the actual service mounting.
   (b) Cause no dynamic coupling to the test item.

(8) The in situ application of vibratory devices to superimpose seismic vibratory loadings on a complex active device for operability testing is acceptable when it can be shown that a meaningful test can be made in this fashion.

(9) 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 electrical and mechanical equipment to ensure their structural capability to withstand seismic excitation.

(2) The analytical results must include the following:

(a) The required input motions to the mounted equipment should be obtained and characterized in the manner as stated in Section 2.a.(2), above.

(b) The combined stresses of the support structures whould be within the limits of ASME Boiler and Pressure Vessel Code Section III, Subsection NF, "Component Support Structures," or other comparable stress limits.

(3) Supports should be tested with equipment installed. If the equipment is inoperative during the support test, the response at the equipment mounting locations should be monitored and characterized in the manner as stated in Section 2.a.(2). In such case, equipment should be tested separately and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response.

(4) The requirements of Sections 2.a.(2), (4), (5), (6), and (7), above, are applicable when tests are conducted on the equipment supports.

C. REFERENCES

1. Note to Electrical, Instrumentation and Control Systems Branch from T. A. Ippolito, January 7, 1974.

BRANCH TECHNICAL POSITION EICSB 11
STABILITY OF OFFSITE POWER SYSTEMS

A. BACKGROUND

The staff has traditionally required each applicant to perform stability studies for the electrical transmission grid which would be used to provide the offsite power sources to the plant. The basic requirement is that loss of the largest operating unit on the grid will not result in loss of grid stability and availability of offsite power to the plant under consideration. In some cases, such as plants on the island of Puerto Rico, the plant is connected to an isolated power system of limited generating capacity. These kinds of isolated power systems are inherently less stable than equivalent systems with supporting grid interties. It is also obvious that limited systems are more vulnerable to natural disasters such as tornadoes or hurricanes.

B. BRANCH TECHNICAL POSITION

1. The staff has concluded, from a review of appropriate reliability data, that power systems with supporting grid interties meet the grid availability criterion with some margin. This conclusion is applicable to the review of most plants located on the U. S. mainland.

2. There is also strong indication that an isolated system large enough to justify inclusion of a nuclear unit will also meet this criterion. However, as a conservative approach, the staff will examine the available generating capacity of a system, including interties if available, to withstand outage of the largest unit. If the available capacity is judged marginal to provide adequate stability of the grid, additional measures should be taken. These may include provisions for additional capability and margin for the onsite power system beyond the normal requirements, or other measures as may be appropriate in a particular case. The additional measures to be taken should be determined on an individual case basis.

BRANCH TECHNICAL POSITION EICSB 12
PROTECTION SYSTEM TRIP POINT CHANGES FOR OPERATION WITH REACTOR COOLANT PUMPS OUT OF SERVICE

A. BACKGROUND

For the past several years, including a time prior to the development of IEEE Std 279, the staff has required automatic adjustment to more restrictive settings of trips affecting reactor safety by means of circuits satisfying the single failure criterion. The basis for this requirement is that the function can be accomplished more reliably by automatic circuity than by a human operator. This design practice, which has also been adopted independently by the national laboratories and by much of industry, served as the basis for paragraph 4.15, "Multiple Set Points," of IEEE Std 279.

More recently, all applicants have stated that their protection systems were designed to meet IEEE Std 279. Paragraph 4.15 of IEEE Std 279 specified that where a mode of reactor operation requires a more restrictive set point, the means for insuring the more restrictive set point shall be positive and must meet the other requirements of IEEE Std 279. A number of designs have been proposed and accepted which reliably and simply satisfy this requirement. During the review of some applications, however, certain design deficiencies have been found. The purpose of this position is to provide additional guidance on the application of Section 4.15 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

1. If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihood to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically.

2. Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually.

C. REFERENCES

1. Report to the ACRS on the protection system trip point changes for operation with the reactor coolant pumps out of service, July 28, 1970.

2. Memorandum to R. C. DeYoung from V. Stello, September 14, 1973 (RESAR).

3. Millstone-3 Safety Evaluation Report, September 24, 1973.

4. Beaver Valley-2 Safety Evaluation Report, October 10, 1973.

BRANCH TECHNICAL POSITION EICSB 13
DESIGN CRITERIA FOR AUXILIARY FEEDWATER SYSTEMS

A. BACKGROUND

The function of the auxiliary feedwater system in pressurized water reactors is to provide an emergency source of feedwater supply to the steam generators. It is required to ensure safe shutdown in the event of a main turbine trip with loss of offsite power. The system is also started on a safety injection signal. Feedwater is pumped to each steam generator through normally open control valves. It was found that in some plant designs the auxiliary feedwater system did not meet the single failure criterion. It is the purpose of this branch technical position to provide guidance and to establish uniform requirements for acceptable designs of auxiliary feedwater systems.

B. BRANCH TECHNICAL POSITION

The auxiliary feedwater system should be capable of satisfying the system functional requirements after a postulated break in the auxiliary feedwater piping inside containment together with a single electrical failure. The basis for the position is that an auxiliary feedwater piping break would result in tripping the unit and, in turn, might cause loss of offsite power. Standard staff assumptions for analyzing postulated accidents include the assumption of loss of offsite power if the affected unit generator is tripped by the accident. Such a circumstance would leave the plant without adequate means for removal of afterheat even though the reactor coolant pressure boundary was intact, an unacceptable result. Plant heat removal systems must, in any postulated piping break, be capable of removing afterheat to the ultimate heat sink assuming a single electrical (active) failure anywhere in the auxiliary feedwater system or in the onsite power system.

C. REFERENCES

1. Note from T. A. Ippolito to EICSB, December 12, 1973.

BRANCH TECHNICAL POSITION EICSB 14

SPURIOUS WITHDRAWALS OF SINGLE CONTROL RODS IN PRESSURIZED WATER REACTORS

A. BACKGROUND

Recent operating experience with PWR's and subsequent reviews of PWR designs with regard to the requirements of General Design Criteria 20 and 25 have shown that single failures can cause inadvertent single rod withdrawals. The intent of this branch technical position is to provide specific guidance toward an acceptable interpretation and application of GDC 20 and 25.

B. BRANCH TECHNICAL POSITION

Applicants have to demonstrate compliance with the requirements of GDC 20 and 25. For this purpose, it has to be shown by analysis that the consequences of uncontrolled or erroneous withdrawal of a single control rod under any possible conditions of reactor operation does not result in exceeding specified acceptable fuel design limits. If the results of this analysis show that the limits may be exceeded, the applicant must provide the results of failure modes and effects analyses to show that a single failure occurring in the control system, or an operator error, will not cause the uncontrolled or erroneous withdrawal of a single control rod. If the results of these analyses show that it is possible for uncontrolled or erroneous withdrawal of single control rods to occur, and the specified fuel design limits could be exceeded as a result, then the protection system must be designed to detect and terminate the resulting transient before the fuel design limits are exceeded.

C. REFERENCES

1. Surry 3 and 4 Safety Evaluation Report, March 26, 1974.
2. Byron & Braidwood, First Set of Questions-Addendum, memorandum to R. C. DeYoung from V. Stello, December 12, 1973.

BRANCH TECHNICAL POSITION EICSB 15
REACTOR COOLANT PUMP BREAKER QUALIFICATION

A. BACKGROUND

An assumption usually made in accident analyses is that for complete loss of forced reactor coolant flow (resulting from a failure of the main coolant pump power supply that is presaged by an underfrequency condition), a reactor trip is initiated along with disengagement of the reactor coolant pumps from the power grid to assure that the pumps' kinetic energy is available for flow coastdown. Therefore, unless the pump breakers are Class IE and are housed in a seismic Category I structure, the required disengagement of the pump motors from the power grid when it experiences the underfrequency condition might not occur. It is the intent of this branch technical position to provide guidance in meeting this concern.

B. BRANCH TECHNICAL POSITION

1. If credit is taken for reactor coolant pump coastdown in the accident analyses, the pump breakers must be qualified in accordance with the requirements of IEEE Std 279-1971 and IEEE Std 308-1971. Further, they must be located in a seismic Category I structure.

2. Any reactor pump system trip sensors associated with these breakers should meet the requirements of IEEE Std 279-1971, regardless of whether or not credit is taken for pump coastdown. If credit is not taken for pump coastdown, the building or structure housing these breakers does not have to be seismic Category I. It has been tentatively established that unless the applicant can demonstrate by analysis that an underfrequency rate of 15 Hz/sec. will not prevent the pumps from performing their coastdown function, the tripping of the reactor coolant pump breakers will be considered a required safety action.

C. REFERENCES

1. Vogtle Safety Evaluation Report, December 18, 1973.

BRANCH TECHNICAL POSITION EICSB 16

CONTROL ELEMENT ASSEMBLY (CEA) INTERLOCKS IN COMBUSTION ENGINEERING REACTORS

A. BACKGROUND

Certain control element assembly interlocks provided in Combustion Engineering designs have not been treated as safety-related. It has been determined by the staff that, unless it can be shown by analysis that these interlocks are not required to assure fuel integrity, they should be treated as required for safety.

B. BRANCH TECHNICAL POSITION

The following interlocks in CE designs are considered safety-related, and unless it can be substantiated otherwise by supporting analyses, they should be designed to meet the requirements of IEEE Std 279. The interlocks in question are intended to prevent the following actions:

1. Insertion of shutdown CEA's before the regulating CEA's are inserted.

2. Simultaneous withdrawal of more than two groups of CEA's.

3. Withdrawal of a CEA group or groups out of proper sequence.

C. REFERENCES

1. Memorandum to P. A. Morris from E. G. Case, May 5, 1970.

BRANCH TECHNICAL POSITION EICSB 17

DIESEL-GENERATOR PROTECTIVE TRIP CIRCUIT BYPASSES

A. BACKGROUND

Where protective trips are provided to protect the standby diesel-generators from possible damage or degradation, these protective trips could interfere with the successful functioning of the diesel-generators when they are most needed, i.e., during an accident condition. In nuclear power plant applications, the criterion should be to provide standby power when needed to mitigate the effects of an accident condition, rather than to protect the diesel-generators from possible damage or degradation.

B. BRANCH TECHNICAL POSITION

1. The design of standby diesel-generator systems should retain only the engine overspeed and the generator differential trips and bypass all other trips under an accident condition. All those trips that are bypassed for an accident condition may be retained for the diesel-generator routine tests. This concept will reduce the probability of spurious trips during accident conditions and will also reduce the exposure of the equipment to damage from malfunctions during routine tests.

2. The design should include capability for testing the status and operability of the bypass circuits and should alarm abnormal values of all the bypassed parameters in the control room.

3. If other trips, in addition to the engine overspeed and generator differential, are retained for accident conditions, an acceptable design should provide two or more independent measurements of each of these trip parameters. Trip logic should be such that diesel-generator trip would require specific coincident logic.

4. The bypass circuitry for the diesel-generator protective trips should be designed to meet the requirements of IEEE Std 279-1971.

C. REFERENCES

1. Memorandum to R. C. DeYoung from D. F. Knuth, March 3, 1972.

2. St. Lucie Units 1 and 2 (Operating License and Construction Permit).

3. SWESSAR-P1 - Stone and Webster Corporation Standard Plant Design.

BRANCH TECHNICAL POSITION EICSB 18

APPLICATION OF THE SINGLE FAILURE CRITERION TO MANUALLY-CONTROLLED ELECTRICALLY-OPERATED VALVES

A. <u>BACKGROUND</u>

Where a single failure in an electrical system can result in loss of capability to perform a safety function, the effect on plant safety must be evaluated. This is necessary regardless of whether the loss of safety function is caused by a component failing to perform a requisite mechanical motion, or by a component performing an undesirable mechanical motion.

This position establishes the acceptability of disconnecting power to electrical components of a fluid system as one means of designing against a single failure that might cause an undesirable component action. These provisions are based on the assumption that the component is then equivalent to a similar component that is not designed for electrical operation, e.g., a valve that can be opened or closed only by direct manual operation of the valve. They are also based on the assumption that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The validity of these assumptions should be verified when applying this position.

B. <u>BRANCH TECHNICAL POSITION</u>

1. Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems of valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

2. Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant technical specifications should include a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

3. Electrically-operated valves that are classified as "active" valves, i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless: (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least ten minutes following occurrence of the event requiring such operation, and (c) it is demonstrated

that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant technical specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

4. When the single failure criterion is satisfied by removal of electrical power from valves described in (2) and (3), above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

5. The phrase "electrically-operated valves" includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

C. REFERENCES

1. Memorandum to R. C. DeYoung and V. A. Moore from V. Stello, October 1, 1973.

BRANCH TECHNICAL POSITION EICSB 19

ACCEPTABILITY OF DESIGN CRITERIA FOR HYDROGEN MIXING AND DRYWELL VACUUM RELIEF SYSTEMS

A. BACKGROUND

Certain design problems arise from the containment design concept which utilizes a drywell and suppression pool for heat removal after a loss-of-coolant accident (LOCA). Two such problems are (1) the hydrogen concentration in the drywell may, in a relatively short time, exceed the limits described in BTP CSB 6-2 (a safety-related problem), and (2) eventual cooling of the drywell will cause steam to condense, resulting in a partial vacuum which can draw water from the suppression pool and partially flood the drywell (a problem related to equipment deterioration and repair costs, not safety).

A hydrogen mixing system is proposed to mix the atmosphere in the larger containment volume outside the drywell with that in the drywell, thereby reducing the overall hydrogen concentration to an acceptable level. In some designs, the hydrogen mixing system bypasses the suppression pool, resulting in an additional load on the containment heat removal system, and in the possibility of overpressurizing the containment. (There are times during a LOCA when bypassing the suppression pool would quickly overpressurize the containment.)

Some designs propose to avoid flooding of the drywell by means of a vacuum relief system utilizing the valves of the hydrogen mixing system.

In view of the stresses to which the reactor operator might be subject during and following a LOCA, it has been concluded that automatic as well as manual initiation at the system level should be provided in BWR 6/Mark III plants.

B. BRANCH TECHNICAL POSITION

1. The design of the hydrogen mixing system should provide for both manual and automatic initiation and should conform to all criteria for protection systems, including the provisions of IEEE Std 279-1971 and Regulatory Guides 1.22 and 1.62. Automatic initiation should come from the sensors which sense that the hydrogen concentration in the drywell has exceeded the limits described in BTP CSB 6-2.

2. The design should provide interlocks in both the automatic and manual circuits that will preclude the opening of valves which bypass the suppression pool before blowdown is complete.

3. If the hydrogen mixing system bypasses the suppression pool, the containment heat removal system should be automatically initiated whenever the hydrogen mixing system is initiated.

4. The containment heat removal system should be automatically initiated upon indication of high pressure in the containment.

5. In conformance with paragraph 4.8 of IEEE Std 279-1971, all signal inputs to the hydrogen mixing system and to those portions of the vacuum relief system which are common to the hydrogen mixing system, should be direct measures, to the extent practical, of the desired variable. Exceptions should be identified and justified.

C. REFERENCES

1. Draft Memorandum to J. M. Hendrie from T. A. Ippolito, October 12, 1973.
2. Branch Technical Position CSB 6-2, "Guidelines for the Evaluation of the Bypass Leakage in Dual Containment Plants," attached to Standard Review Plan 6.2.5.

BRANCH TECHNICAL POSITION EICSB 20

DESIGN OF INSTRUMENTATION AND CONTROLS PROVIDED TO ACCOMPLISH CHANGEOVER FROM INJECTION TO RECIRCULATION MODE

A. BACKGROUND

Designs are reviewed with regard to the automatic and manual initiation of protective actions, as set forth in paragraph 4.17 of IEEE Std 279-1971. For some recent designs, the staff concluded that the proposed design of the circuits used to change over to the recirculation mode of operation following a loss-of-coolant accident did not conform to IEEE Std 279-1971, and the complexity of the proposed changeover procedure raised questions as to whether the operator could be expected to perform correctly the required actions within the time and based on the information available to him.

B. BRANCH TECHNICAL POSITION

1. A design that provides manual initiation at the system level of the transfer to the recirculation mode, while not ideal, is sufficient and satisfies the intent of IEEE Std 279-1971 provided that adequate instrumentation and information display are available to the operator so that he can make the correct decision at the correct time. Furthermore, it should be shown that, in case of operator error, there are sufficient time and sufficient information available so that the operator can correct the error, and the consequences of such an error are acceptable.

2. Automatic transfer to the recirculation mode is preferable to manual transfer, for the reasons cited above, and should be provided for standard plant designs submitted for review on a generic basis under the Commission's standardization policy.

C. REFERENCES

1. Memorandum to R. C. DeYoung from V. Stello, October 10, 1973 (Beaver Valley-2 Safety Evaluation Report - EICSB).

BRANCH TECHNICAL POSITION EICSB 21
GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.47

A. BACKGROUND

The recommendations of Regulatory Guide 1.47 need further detailing as to methods of providing an acceptable design for the bypass and inoperable status indicators for engineered safety feature (ESF) systems. The purpose of this branch technical position is to provide supplemental guidance for implementation of the recommendations of Regulatory Guide 1.47.

B. BRANCH TECHNICAL POSITION

The design criteria for bypass and inoperable status indication systems for ESF should reflect the importance of providing accurate information for the operator and of reducing the possibility for the indicating equipment to affect adversely the monitored safety systems. In developing the design criteria, the following should be considered:

1. The bypass indicators should be arranged to enable the operator to determine the status of each safety system and determine whether continued reactor operation is permissible.

2. When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit.

3. Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design.

4. Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based solely on the bypass indications.

5. The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems.

6. The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified.

C. REFERENCES

1. Memorandum to J. M. Hendrie from V. A. Moore, February 27, 1973.

BRANCH TECHNICAL POSITION EICSB 22

GUIDANCE FOR APPLICATION OF REGULATORY GUIDE 1.22

A. BACKGROUND

A recent application listed eight functions that are not tested while the reactor is operating at power. The applicant claimed that the periodic testing complied with Regulatory Guide 1.22. Regulatory Guide 1.22 does make provisions for actuated equipment that is not tested during reactor operation but it does not have provisions for excluding any portion of the protection system from the requirements of paragraphs 4.9 and 4.10 of IEEE Std 279-1971.

B. BRANCH TECHNICAL POSITION

All portions of the protection systems should be designed in accordance with IEEE Std 279-1971, as required by 10 CFR §50.55a(h). All actuated equipment that is not tested during reactor operation should be identified and a discussion of how each conforms to the provisions of paragraph D.4 of Regulatory Guide 1.22 should be submitted.

C. REFERENCES

1. Memorandum to R. C. DeYoung from V. Stello, September 24, 1973, (Millstone 3, Second Round of Questions).

BRANCH TECHNICAL POSITION EICSB 23

QUALIFICATION OF SAFETY-RELATED DISPLAY INSTRUMENTATION FOR POST-ACCIDENT CONDITION MONITORING AND SAFE SHUTDOWN

A. BACKGROUND

Instrumentation systems for post-accident monitoring and safe shutdown must survive the accident to be effective when needed. Environmental qualification should be in accordance with the provisions of IEEE Std 323-1974 and IEEE Std 344-1971. The recorders of these instrumentation systems are not required to function with accuracy during the safe shutdown earthquake; they must function with accuracy after the ground motion subsides without requireing any maintenance.

B. BRANCH TECHNICAL POSITION

The safety-related display instrumentation for post-accident monitoring and safe shutdown should be:

1. Redundant, with indicators in the control room for both channels and with at least one channel recorded.

2. Energized from the onsite emergency power supplies.

3. Designed in accordance with the requirements of IEEE Std 279-1971.

4. Qualified in accordance with the requirements of IEEE Std 323-1974 and IEEE Std 344-1971 as supplemented by BTP EICSB 10 with the exception that the recorders are not required to function within their required accuracy during the safe shutdown earthquake, but must function within their required accuracy immediately after the ground motion subsides without requiring any maintenance.

C. REFERENCES

1. Memorandum to V. A. Moore from V. Stello, October 12, 1973 (GESSAR).

BRANCH TECHNICAL POSITION EICSB 24

TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SENSOR RESPONSE TIMES

A. BACKGROUND

The accident analyses in safety analysis reports assume certain response times for the reactor protection systems. Periodic verification of the protection system response times should be made to assure that they are within the design specifications assumed in the accident analyses.

B. BRANCH TECHNICAL POSITION

1. Periodic tests for verification of system response times of reactor trip systems and engineered safety feature actuation systems should include the response times of the sensors whenever practical.

2. In some cases, indirect means of verifying sensor response times may be used. Details of such indirect means of verifying sensor response times should be included in applications and will be reviewed by the staff on an individual case basis until some uniformity of practice develops and generic guidance can be provided.

3. Exceptions to the above should be specifically identified and justified.

C. REFERENCES

Memorandum to V. A. Moore from V. Stello, October 12, 1973, (GESSAR Second Round of Questions, No. 2 and No. 9).

BRANCH TECHNICAL POSITION EICSB 25

GUIDANCE FOR THE INTERPRETATION OF GENERAL DESIGN CRITERION 37 FOR TESTING THE OPERABILITY OF THE EMERGENCY CORE COOLING SYSTEM AS A WHOLE

A. BACKGROUND

General Design Criterion 37 requires, in part, that the emergency core cooling system be designed to permit testing the operability of the system as a whole under conditions as close to design as practical. It is stated in one recent application that the safety injection and residual heat removal pumps are made inoperable during the system tests.

B. BRANCH TECHNICAL POSITION

In order to comply with the requirements of GDC 37, all ECCS pumps should be included in the system test.

C. REFERENCES

1. Memorandum to R. C. DeYoung from V. Stello, September 14, 1973 (RESAR).

BRANCH TECHNICAL POSITION EICSB 26
REQUIREMENTS FOR REACTOR PROTECTION SYSTEM ANTICIPATORY TRIPS

A. BACKGROUND

Several reactor designs have incorporated a number of anticipatory or "back-up" trips for which no credit was taken in the accident analyses. These trips, as a rule, were not designed to the requirements of IEEE Std 279 and therefore introduced non-safety grade equipment into the reactor protection system. It was determined by the staff that this was not an acceptable practice, because of possible degradation of the reactor protection system.

B. BRANCH TECHNICAL POSITION

All reactor trips incorporated in the reactor protection system should be designed to meet the requirements of IEEE Std 279, without exception. This position applies to the entire trip function from the sensor to the final actuated device.

C. REFERENCES

1. Shearon Harris Safety Evaluation Report, September 15, 1972.
2. Memorandum to V. A. Moore from V. Stello, October 12, 1973 (GESSAR).

BRANCH TECHNICAL POSITION EICSB 27

DESIGN CRITERIA FOR THERMAL OVERLOAD PROTECTION FOR MOTORS OF MOTOR-OPERATED VALVES

A. BACKGROUND

The National Electrical Code (NEC) recommends an overload setting of 115% to 125% of motor full-load current for most continuous duty motors.

According to the NEC, a short-time (intermittent) duty motor, such as a valve operator motor, shall be considered as protected against overcurrent by the branch circuit device, provided the overcurrent protection does not exceed the specified values in the code. The maximum rating of motor branch circuit protective fusing recommended by the NEC is 300% of motor full-load current.

The accuracy obtainable with a thermal overload relay trip generally varies from -5% to 0% of its trip set point. Since the primary concern in the application of overload relays is to protect the motor windings against excess heating, this negative tolerance in the relay trip characteristics is considered in the safe direction, as it will trip sooner to protect the motor. This feature of thermal overload relays could interfere with the successful functioning of a safety-related system. In nuclear power plant safety system applications, the criterion should be to drive the valve to its proper position to mitigate the effects of an accident condition, rather than to be concerned with degradation or failure of the motor due to excess heating.

B. BRANCH TECHNICAL POSITION

1. Thermal overload protection, if provided for safety-related system motor-operated valves, should have the trip set point set at a value high enough to prevent spurious trips due to design inaccuracies, trip set point drift, or variation in the ambient temperature at the installed location. The trip set point chosen should be consistent with that of any branch circuit protective device used. Periodic tests should be performed on each of the thermal overload devices to verify the accuracy and reliability of the overload trip set point.

2. Thermal overload protection may be bypassed under accident conditions. The bypass circuitry should be designed to IEEE Std 279-1971 criteria, as appropriate for the rest of the safety-related system.

C. REFERENCES

1. Memorandum to J. M. Hendrie from T. A. Ippolito, April 11, 1974.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

APPENDIX 7-B GENERAL AGENDA, STATION SITE VISITS

An important part of the review at the operating license stage is a site visit. It is preferable to have the site visit sometime before the completion of the drawing review. The purpose of the site visit is to supplement the review of the design based on the drawings and to evaluate the actual implementation of the design as installed at the site. The Regional Office of Regulatory Operations having jurisdiction over the plant under consideration should be notified ahead of time of the visit so that the regional inspectors can become familiar on a first-hand basis with findings that may require followup action. Since proper implementation of design is the ultimate goal of the technical review process, the importance of a site visit is self-evident. The following is a typical general agenda that may be used as a guide for developing a specific agenda for the plant under review.

1. Preliminary Discussions
   a. Unresolved items.
   b. Plant layout for touring.
   c. Special interest areas.

2. Control Room
   a. General layout.
   b. Nuclear and reactor protection instrument arrangement and layout.
   c. Rod position indication.
   d. Protection system initiation and bypass switch arrangements.
   e. Diesel control board.
   f. Cabling in control room (separation, loading, etc.).
   g. Radiation monitoring.
   h. Engineered safety feature initiation and bypass switch arrangements and status panels.

3. Cable Runs and Cable Spreading Area
   a. General layout.
   b. Degree of separation.
   c. Diverse wiring.
   d. Tray or wireway density (percentage fill).

e. Fire detection and protection.

f. Penetrations and cable terminations.

4. Switchgear Rooms

   a. General layout.

   b. Physical and electrical separation of redundant units.

   c. Potential for damage due to fire, missiles, etc.

   d. Cable installation.

   e. Fire detection and protection.

5. Battery Installations

   a. General layout.

   b. Physical and electrical separation.

   c. Potential for damage due to fire, missiles, etc.

   d. Fire detection and protection and security.

   e. Ventilation independence.

   f. Monitoring instrumentation.

6. Diesel Generators

   a. General layout.

   b. Physical and electrical separation of redundant units.

   c. Fuel supply system.

   d. Fire detection and protection.

   e. Qualification tests - interlocks and control panel.

   f. Auxiliary systems - starting air, combustion air, ventilation.

7. Instrument Piping

   a. Physical separation and single failure.

   b. Potential for damage due to fire, flooding, etc.

   c. Test features.

8. Transformers (Switchyard)

   a. Physical and electrical separation.

   b. Potential for damage due to fire, flooding, missiles, etc.

   c. Fire detection and protection.

9. Quality Control

   a. Onsite receipt, storage, installation, and protection procedures of installed instrumentation, equipment, and cables.

10. Reactor Building and Turbine Building

   a. Protection system instrument arrangement and layout.

   b. Potential for instrument damage due to fire, missiles, etc.

   c. Separation of piping and wiring to redundant instruments.

   d. Provisions for testing protection instruments.

11. <u>Shared Systems for Multi-Unit Sites</u>
    a. Equipment location and potential for damage.
    b. Control room control and assignment to accident unit.
    c. Availability upon completion of first unit.

12. <u>Steam Lines - Main, HPCI, RCIC</u>
    a. BWR temperature and radiation monitoring systems.
    b. Isolation valves.

13. <u>Recirculation Water System (Condenser)</u>
    a. Break detection and flood protection features.

14. <u>Shutdown Outside Control Room</u>
    a. Location for potential damage.
    b. Feedwater system, etc.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

TABLE 7-1

ACCEPTANCE CRITERIA FOR CONTROLS

Table 7-1 contains the acceptance criteria for the review plans of Chapter 7. These acceptance criteria include the applicable general design criteria, IEEE standards, regulatory guides, and branch technical positions (BTP) of the Electrical, Instrumentation and Control Systems Branch (EICSB). The table was prepared by EICSB for use by its members in reviewing Chapter 7 and for use by secondary review branch reviewers.

The applicability of these criteria to specific sections of Chapter 7 is indicated by an X in the matrix listing of criteria and SAR sections. There is a corresponding table (8-1) at the end of Chapter 8 covering the acceptance criteria of safety-related power supplies. The BTP listed in Tables 7-1 and 8-1 are contained in Appendix 7-A to the Chapter 7 review plans.

ACCEPTANCE CRITERIA FOR CONTROLS - TABLE 7-1

CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
1. 10 CFR Part 50									
a. 10 CFR §50.34	Contents of Application: Technical Information	X	X	X	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	X	X		
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50									
a. GDC 1	Quality Standards and Records	X	X	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems, and Components	X	X	X	X	X	X		
f. GDC 10	Reactor Design	X	X	X	X	X	X		
g. GDC 12	Suppression of Reactor Power Oscillations	X	X			X		X	
h. GDC 13	Instrumentation and Control	X	X	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design	X	X			X		X	
j. GDC 19	Control Room	X	X	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X	X	X		
l. GDC 21	Protection System Reliability and Testability	X	X	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X	X	X		

TABLE 7-1 (CONTINUED)

CRITERIA		TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
n.	GDC 23	Protection System Failure Modes	X	X	X	X	X	X		
o.	GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	X	X	
p.	GDC 25	Protection System Requirements for Reactivity Control Malfunctions	X	X			X			
q.	GDC 26	Reactivity Control System Redundancy and Capability	X	X		X	X		X	
r.	GDC 27	Combined Reactivity Control Systems Capability	X	X		X	X		X	
s.	GDC 28	Reactivity Limits	X	X			X	X	X	
t.	GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	X	X	
u.	GDC 33	Reactor Coolant Makeup	X			X	X	X		
v.	GDC 34	Residual Heat Removal	X		X	X	X	X		
w.	GDC 35	Emergency Core Cooling	X	X	X		X	X		
x.	GDC 37	Testing of Emergency Core Cooling System	X	X	X		X	X		
y.	GDC 38	Containment Heat Removal	X		X		X	X		
z.	GDC 40	Testing of Containment Heat Removal System	X		X		X	X		
aa.	GDC 41	Containment Atmosphere Cleanup	X		X		X	X		
bb.	GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X		X	X		
cc.	GDC 44	Cooling Water	X		X		X	X		
dd.	GDC 46	Testing of Cooling Water System	X		X		X	X		
ee.	GDC 50	Containment Design Basis	X				X	X		

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TABLE 7-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
ff. GDC 54	Piping Systems Penetrating Containment	X		X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X		X	X		
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:									
a. IEEE Std 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See 10 CFR §50.55a(h) and Reg. Guide 1.62
b. IEEE Std 308-1971	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X			X	X	X		See Reg. Guide 1.32.
c. IEEE Std 317-1972	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.63.
d. IEEE Std 336-1971 (ANSI N45.2.4-1972)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338-1971	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X	X	X		
f. IEEE Std 344-1971 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X	X	X		
g. IEEE Std 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Sytems	X	X	X	X	X	X	X	See Reg. Guide 1.53.
h. IEEE Std 384-1974 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X	X	X		X			

TABLE 7-1 (CONTINUED)

CRITERIA		TITLE	APPLICABILITY (SAR Section) 7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
4.	Regulatory Guides (RG)									
a.	RG 1.6	Independence Between Reduncant Standby (Onsite) Power Sources and Between Their Distribution Systems	X			X		X		
b.	RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident	X		X		X			
c.	RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X	X	X		
d.	RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	X	X		
e.	RG 1.29	Seismic Design Classification	X	X	X	X	X	X		
f.	RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	X	X	
g.	RG 1.32	Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X			X	X	X		
h.	RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	X	X		Use in conjunction with Position 3, RG 1.17
i.	RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X	X	X	X	X	X		
j.	RG 1.62	Manual Initiation of Protection Actions	X	X	X	X		X		
k.	RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	X	X	X	X	X	X	X	
l.	RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X						

TABLE 7-1 (CONTINUED)

	CRITERIA	TITLE	APPLICABILITY (SAR Section)							REMARKS
			7.1	7.2	7.3	7.4	7.5	7.6	7.7	
m.	RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2.	X	X	X	X	X	X	X	
n.	RG 1.75	Physical Independence of Electric Systems	X	X	X		X			
o.	RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X					X		
p.	RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X	X	X	X	X	X		
q.	RG 1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	X		X					
5.	Branch Technical Positions (BTP) EICSB									
a.	BTP EICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X	X	X		X		
b.	BTP EICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System	X			X		X		
c.	BTP EICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	X			X		X		
d.	BTP EICSB 5	Scram Breaker Test Requirements - Technical Specifications	X	X						
e.	BTP EICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications	X	X		X	X	X		
f.	BTP EICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X		X	X	X		
g.	BTP EICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X	X	X					

TABLE 7-1 (CONTINUED)

	CRITERIA	TITLE	7.1	7.2	7.3	7.4	7.5	7.6	7.7	REMARKS
			APPLICABILITY (SAR Section)							
h.	BTP EICSB 13	Design Criteria for Auxiliary Feedwater Systems	X		X					
i.	BTP EICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	X	X						
j.	BTP EICSB 15	Reactor Coolant Pump Breaker Qualification	X	X						
k.	BTP EICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors	X	X						
l.	BTP EICSB 18	Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves	X		X	X		X		
m.	BTP EICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X		X			X		
n.	BTP EICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recriculation Mode	X		X	X		X		
o.	BTP EICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	X	X		
p.	BTP EICSB 22	Guidance for Application of Reg. Guide 1.22	X	X	X	X	X	X		
q.	BTP EICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown	X				X			
r.	BTP EICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X	X	X		X		
s.	BTP EICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X		X	X				
t.	BTP EICSB 26	Requirements for Reactor Protection System Anticipatory Trips	X	X						
u.	BTP EICSB 27	Design Criteria for Thermal Overland Protection for Motors of Motor-Operated Valves	X		X	X		X		

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STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 8.1 INTRODUCTION

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)

I. AREAS OF REVIEW

Section 8.1 of the applicant's safety analysis report (SAR) is reviewed to determine the adequacy of the information presented with reference to the information requirements of the corresponding section of the Standard Format, Revision 2 (Item 4.1 of Ref. 1).

The review is also to include evaluation of the proposed technical specifications (SAR Chapter 16) to assure that they are adequate with regard to limiting safety system settings, limiting conditions for operation, and periodic surveillance testing.

The secondary review branches (APCSB, CSB, and RSB) review the listing of safety loads for completeness, i.e., to verify that all safety loads within their respective areas of primary review responsibility have been identified. If loads other than those identified are deemed to be safety-related, this information will be transmitted to EICSB.

II. ACCEPTANCE CRITERIA

The description of the power grid and offsite power system is acceptable when it can be concluded that the interrelationships between the nuclear unit, the utility grid, and the interconnecting grids are clearly defined. The identification of safety loads is acceptable when it can be concluded that all systems and devices that require electric power (a-c or d-c) to perform safety functions are identified.

Table 8-1, "Acceptance Criteria for Electric Power," lists the criteria currently applied by the staff to safety-related electric power systems. Implementation of these criteria will provide assurance that safety-related electric power systems will perform design safety functions as required. The applicant's list of design criteria for safety-related electric power systems is acceptable if it includes the items in Table 8-1, and if the SAR contains a statement to the effect that these criteria will be implemented (at the construction permit stage) or are implemented (at the operating license stage) in the design of the electrical power systems.

The fundamental bases for acceptance of the proposed technical specifications are that the limiting conditions for operation (LCO's) are such that sufficient equipment will be available for operation as required to meet the single failure criterion; that equipment outages that are permissible for a short period of time still leave available sufficient equipment to provide the protective function assuming no failures; and that the provisions of the technical specifications are compatible with the safety analyses. The operating procedures and restrictions which should be implemented if the available electric power sources are less than the LCO are discussed in Regulatory Guide 1.93.

III. REVIEW PROCEDURES

The objective of the review of Section 8.1 of the SAR is to determine if the information requirements defined in the corresponding section of the Standard Format, Revision 2, have been met.

The information presented should include: a brief description of the utility grid and its interconnections to other grids and to the nuclear unit (referred to as the preferred power system); a brief general description of the onsite power system (referred to as the standby power system); identification of the safety loads (i.e., the systems and devices that require electric power to perform safety functions); identification of the function performed by each load (e.g., emergency core cooling, containment cooling); the type of electric power (a-c or d-c) required by each load; and the design bases, criteria, standards, regulatory guides, and technical positions that will be implemented in the design of the safety-related electric power systems, including a discussion and a positive statement with regard to conformance of the design to each of these crtieria.

The review is performed as follows:

1. EICSB will establish that the utility grid is adequately described, and that the interconnections between the nuclear unit, the utility grid, and other grids are clearly defined. The descriptions should state whether facilities are existing or planned; if planned, the respective completion dates should be provided. The descriptions should not conflict with the more detailed information in subsequent sections of Chapter 8, and may reference these sections.

2. EICSB confirms that the description of the onsite power system (standby power system) is not in conflict with the more detailed information on this system in subsequent sections of Chapter 8.

3. EICSB will establish that all the devices and systems that require electric power to perform safety functions are identified, and that this identification does not conflict with the more detailed infromation provided in other sections of the SAR, particularly in Chapters 7 and 8. The definitions of safety-related systems in Standard Review Plan (SRP) 7.1 should be used as an aid in assessing the completeness of the identification of safety loads. Care should be exercised to assure that those loads required to maintain the plant within the envelope of operating conditions postulated in the accident analysis are identified as safety loads. Requests for evaluation should be made to the secondary review branches when there are novel designs or significant differences of opinion with regard to designations of safety loads.

4. The secondary review branches (APCSB, CSB, RSB) will confirm the identification of all safety loads within their respective areas of primary review responsibility. If loads other than those identified are deemed to be safety-related, this information should be transmitted to EICSB.

5. EICSB will confirm that the criteria identified as being applicable to the design of safety-related electric power systems include those items listed in Table 8-1. This will assure that the identification requirements of General Design Criteria (GDC) 1 of Appendix A of 10 CFR Part 50 are met. GDC 1 also require 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." Therefore, the SAR should include a discussion regarding the applicability of the criteria listed and a statement to the effect that the criteria will be implemented (CP) or are implemented (OL) in the design of safety-related electrical power systems.

6. The proposed plant technical specifications (Chapter 16) are reviewed by EICSB and the secondary review branches to:
   a. Confirm the suitability of the limiting safety system settings and the limiting conditions for 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.

For a construction permit (CP) review, it is only necessary to confirm that the applicant has identified those variables, conditions, or other items which have been determined to be probable subjects of the technical specifications. (See 10 CFR § 50.34 (a)(5).) The applicant's justification for the selection of those items is evaluated, with special attention to any that may significantly influence the final design. The specific provisions of the proposed technical specifications are not approved during the CP review. However, any specific provisions which are known to be unacceptable or which may influence acceptance of the preliminary design of the plant should be brought to the applicant's attention and, if appropriate, included in that portion of the staff's evaluation findings pertaining to the design of the affected systems.

For an operating license (OL) review, the proposed technical specifications are reviewed and evaluated in depth in accordance with the requirements of 10 CFR § 50.36. For the EICSB areas of review, a check is made that the limiting conditions for operation (LCO) correspond to the surveillance requirements; i.e., for each system or component that is the subject of a LCO, there must be corresponding surveillance requirements. Each system or component that performs a function for which credit is taken in the accident analyses should be the subject of an LCO. The limiting safety system settings should agree with the values assumed in the accident analyses, including appropriate allowances for instrument error, drift, etc. If the acceptance of the design of a particular system is based upon required plant conditions or particular operating procedures, such requirements should be included in the final technical specifications and, if appropriate, noted in that portion of the staff's evaluation findings pertaining to the design of the affected system. Operating procedures and restrictions acceptable to the Regulatory staff which should be implemented if the available electric power sources have less than the LCO are presented in Regulatory Guide 1.93.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information is presented in the SAR and that his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has identified safety-related electric power systems, safety loads, and applicable power system criteria, and has documented his intent to design and construct these systems in accordance with the criteria. It is concluded that design and construction of safety-related electric power systems in accordance with the criteria provide assurance that these systems will perform as designed."

V. REFERENCES

1. Standard Review Plan Table 8-1, "Acceptance Criteria for Electric Power."

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SECTION 8.2 OFFSITE POWER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The descriptive information, analyses, and referenced documents, including electrical single line diagrams, electrical schematics, logic diagrams, tables, and physical arrangement drawings for the offsite power systems, presented in the applicant's safety analysis report (SAR), are reviewed. The intent of the review is to determine that this system satisfies applicable acceptance criteria and will perform its design functions during plant normal operation, anticipated operational occurrences, and in accident conditions. The information provided at the construction permit (CP) stage should show that the design will be in conformance with the acceptance criteria and should support a statement to this effect to be included in the staff's construction permit safety evaluation report. At the operating license (OL) stage, review of the final design information and a site visit should establish that the design criteria have been correctly implemented, that the design meets the requirements of the safety analyses and conforms to the acceptance criteria, and should support a statement to this effect to be included in the staff's operating license safety evaluation report.

The offsite power system is referred to in industry standards and regulatory guides as the "preferred power system." It includes two or more identified power sources capable of operating independently of the onsite or standby power sources and encompasses the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, the main generator, and disconnect switches, provided to supply electric power to safety-related and other equipment.

The EICSB will pursue the following phases in review of the preferred power system.

1. The preferred power system arrangement is reviewed to determine that the required minimum of two separate circuits from the transmission network to the standby-power distribution system is provided. In determining the adequacy of this system, the independence of the two (or more) circuits is examined to see that both electrical and physical separation exists so as to minimize the chance of simultaneous failure. This includes a review of the assignment of power sources from the grid, location of rights-of-way, transmission lines and towers, transformers, switchyard interconnections (breakers and bus arrangements), switchyard control systems and power supplies, location of switchgear (in plant), interconnections between switchgear, cable routings, main generator disconnect, and the disconnect control system and power supply.

2. The independence of the preferred power system with respect to the standby power system is evaluated. The scope of review extends to the safety-related distribution system buses that are capable of being powered by standby power sources. It does not include the supply breakers of the safety-related distribution system buses. This evaluation will include a review of the electrical protective relaying and breaker control circuits and power supplies to assure that loss of one preferred system circuit will not cause or result in loss of the redundant counterpart, nor any standby power system sources.

3. Design information and analyses demonstrating the suitability of the power sources, transmission lines, breakers, and transformers used for supplying preferred power from a distant source are reviewed to assure that each path has sufficient capacity, capability, and reliability to perform its intended function. This will require examination of loads required to be powered for each plant operating condition; continuous and fault ratings of breakers, transformers, and transmission lines; loading, unloading, and transfer effects on equipment; and power capacity available from each source.

4. The instrumentation required for monitoring and indicating the status of the preferred power system is reviewed to assure that any change in the preferred power system which would prevent it from performing its intended function will be immediately identified by the control room operator. Also, all instrumentation for initiating safety actions associated with the preferred power system is reviewed.

5. Preoperational and initial startup tests and programs and periodic testing capabilities are reviewed.

6. The EICSB will also review the following:
   a. Environmental conditions such as those resulting from floods, hurricanes, high and low atmospheric temperatures, rain, and snow are considered in the review of the preferred power system to determine any effects on function.
   b. Quality group classifications of equipment of the preferred power system are reviewed.

c. The equipment and functions of the preferred power systems that are used as a basis for assumptions in the accident analyses are reviewed to assure that they conform to the requirements of those assumptions.

7. Other areas of review associated with this system are covered elsewhere as follows:
   a. Environmental design and qualification testing of electrical equipment are addressed in Standard Review Plan (SRP) 3.11.
   b. Technical specification requirements imposed upon the operation of the preferred power system are discussed in Chapter 16 of the applicant's safety analysis report (SAR). The review of technical specifications for the preferred power systems is covered in SRP 8.1.
   c. The APCSB will evaluate the adequacy of those auxiliary systems required for the proper operation of the preferred power system in connection with the review of SAR Chapters 9 and 10. These include such systems as heating and ventilation systems for switchgear in the circuits from the preferred power sources to the standby power distribution system buses and main generator auxiliaryy systems such as the cooling water system, hydrogen cooling system, electrohydraulic system, air supply system, and fire detection system.
   d. The APCSB will examine the physical arrangements of components and structures of the preferred power system to assure that the paths from the preferred power sources to the standby power distribution system buses will not experience simultaneous failure under operating or postulated accident environmental conditions.
   e. The RSB and APCSB will be consulted as required to assure proper identification of the electrical equipment and systems required as a function of time for each mode of reactor operation and accident condition.

II. ACCEPTANCE CRITERIA

In general, the preferred power system is acceptable when it can be concluded that two separate paths from the transmission network to the standby power distribution system are provided; adequate physical and electrical separation exists; and the system has the capacity, capability, and reliability to supply power to all safety loads and other required equipment.

Table 8-1 lists general design criteria (GDC), standards of the Institute of Electrical and Electronic Engineers (IEEE), regulatory guides, and staff technical positions utilized as the bases for arriving at this conclusion. In addition, the references include documents used by the reviewer as aids in ascertaining that the criteria have been met. Section III of this plan discusses the application of these documents to the review.

Details of the application of the acceptance criteria to the areas of review described in Section I of this plan are as follows:

1. System Design Requirements.

   a. GDC 33, 34, 35, 41, and 44 set forth requirements for the safety systems that must be supplied by the preferred power system. Also, these criteria state that safety system redundancy shall be such that, for preferred power system operation (assuming standby power is not available), the system safety function can be accomplished assuming a single failure. The acceptability of the preferred power system design in this regard is based on its capability to supply the redundant safety components and systems required by these GDC.

   b. GDC 17 requires two physically independent circuits from the offsite grid.

   c. The preferred power system must be independent of the standby power system. The basis for acceptance is that no single event, including a single protective relay, interlock, or switchgear failure, in the event of loss of standby power, will prevent the separation of the preferred power system from the standby power system or prevent the preferred power system from accomplishing its intended functions. The design must satisfy the requirements of GDC 17 in this regard. In addition, the preferred and standby power supplies should not have common failure modes, as required by Section 5.2-1(5) of IEEE Std 308. To assure that the preferred power system satisfies the requirements of GDC 17, as supplemented by GDC 34, 35, 38, 41 and 44, an acceptable design must be capable of restoring the preferred power supply after the loss of either circuit in a time period such that the plant can be safely shutdown, taking into account the effects of a single failure in the safety-related distribution system.

2. Testing, Quality Assurance, and System Operability Surveillance.

   a. To assure that the requirements of GDC 1 are met in the preferred power system, the quality assurance program must satisfy the requirements of IEEE Std. 336, as augmented by Regulatory Guide 1.30.

   b. Preoperational and initial startup test programs should be in accordance with Regulatory Guide 1.68, as augmented by Regulatory Guide 1.41. To assure that the periodic onsite testing capabilities satisfy the requirements of GDC 18 and 21, an acceptable testing program must satisfy Regulatory Guide 1.22.

   c. With regard to the surveillance of system operability status, an acceptable design must satisfy the positions of Regulatory Guide 1.47, as augmented by Branch Technical Position EICSB 21.

3. Secondary Review Branch Areas.

   For those areas of review identified in Section I of this plan as being the responsibility of other branches, the acceptance criteria are included in the applicable standard review plans. Some areas of review require close coordination between primary and secondary review branches in determining that a certain aspect of the design conforms with the criteria.

III. REVIEW PROCEDURES

The general objectives in the review of the preferred power system are to determine that this system satisfies the acceptance criteria and can reliably and adequately perform the functions that are assumed and used as a bases in the accident analyses for normal and abnormal plant conditions. In the CP review, the descriptive information, including the design bases and their relation to the acceptance criteria, preliminary analyses, electrical single line diagrams, and preliminary physical arrangement and layout drawings are examined to determine that the final design will meet this objective if properly implemented. During the OL review, this objective is verified by examination of final electrical schematics, physical arrangement and layout drawings, and equipment ratings identified in the SAR and confirmed during a visit to the site (SRP Appendix 7-B). To assure that the applicable criteria of Table 8-1 are satisfied, the review of the proposed design is performed as follows:

1. An understanding of the design bases, normal and abnormal operation modes, accident analyses, and plant equipment is required to evaluate the design and acceptability of the preferred power system. This information is gained by reading the SAR and in discussions with the applicant.

2. To assure that the requirements of GDC 17 are satisfied, the following review steps should be taken (as applicable for a CP or OL review):
   a. The electrical schematics should be examined to assure that at least two separate circuits from the transmission network to the standby power distribution system buses are provided (a switchyard may be common to these paths).
   b. The routing of transmission lines should be examined on the station layout drawings and verified during the site visit to assure that at least two independent circuits from the offsite grid to the safety-related distribution buses are physically separate and independent. Preferably these lines should enter the station on separate rights-of-way, ideally on opposite sides of the switchyard, should leave the switchyard on opposite sides, and should terminate at transformers located on opposite sides of the reactor or turbine building. No other line should cross these two circuits. As physical separation becomes less than the ideal, attention should be directed towards assuring that no single event such as a tower falling or a line breaking can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded.
   c. As the switchyard may be common to both circuits from the offsite grid to the safety-related distribution buses, the electrical schematics of the switch-yard breaker control system and power supply and the breaker arrangement itself should be examined for the possibility of simultaneous failure of both circuits from single events such as a breaker not operating during fault conditions, loss of a control circuit power supply, etc.
   d. The design is examined to determine that one of the two circuits can immediately provide power to safety-related equipment following a loss-of-coolant accident. GDC 17 does not require this circuit in itself to be single failure-proof for this accident. However, it is required that each circuit be available in sufficient time to prevent fuel design limits and design conditions of the reactor collant pressure boundary from being exceeded. Therefore, the

switchyard control system design and implementation should be such that any incoming line, switchyard bus, or any path to the safety-related distribution bus can be isolated. This is generally achieved by separated and redundant breaker tripping and closing devices, with each circuit independent of its redundant counterpart including control circuit power supplies. Designs that do not provide redundant control circuits must be justified by an analysis which shows the period of time that the station can remain in a safe condition assuming no a-c power is available. The time established in this analysis must be greater than the time required to re-establish a-c power from the offsite grid to the safety-related distribution bus for each single failure event. These designs sometimes depend on manual operation of the switchyard breakers, which involves an operator going to the yard and manually actuating valves controlling high pressure air stored in accumulators to open the breakers. It has been found in past reviews that several designs were such that the breakers could not be manually released by this action or by other means. Other items to be evaluated concern the consequences of shorting of switchyard buses, battery failures, status of breaker air accumulators, breaker failures, routing of control circuits and power supplies, shorting of transmission lines, and the design of a back-feed path through the main generator transformer if provided in the design.

e. Each of the circuits from the offsite grid to the safety-related distribution buses should have the capacity and capability to supply the loads assigned to the bus or buses it is connected to during normal or abnormal operating conditions, accident conditions, or plant shutdown conditions. Therefore, the loads to be supplied during these conditions should be determined from information provided by the RSB as to the equipment required to be operable for each condition. The capacity and electrical characteristics of transformers, breakers, buses, transmission lines, and the offsite grid power source for each path should be evaluated to assure that there is adequate capability to supply the maximum connected load during all plant conditions. The design should be examined to assure that during transfer from one power source to another the design limits of equipment are not exceeded.

f. The results of the grid stability analysis must show that loss of the largest single supply to the grid does not result in the complete loss of preferred power. The analysis should consider the loss, through a single event, of the largest capacity being supplied to the grid or removal of the largest load from the grid. This could be the total output of the station, the largest station on the grid, or possibly several large stations if these use a common transmission tower, transformer, or a breaker in a remote switchyard or substation. The station layout and the grid system layout drawings are reviewed to determine that all events were included in the analysis.

The applicant should include in the grid stability analysis the consideration of failure modes that could result in frequency variations exceeding the maximum rate of change determined in the accident analysis for loss of reactor coolant flow.

g. During the review of the electrical schematics, it should be determined that loss of standby power will not result in loss of preferred power, loss of one preferred power circuit will not result in loss of the other circuit, and loss of the main generator will not result in loss of either preferred power circuit.

3. To assure that the requirements of GDC 18 and 21, and Regulatory Guide 1.22 are satisfied, the electrical schematics should be examined to determine that the design includes provisions for testing the transfer of power to the safety-related distribution system from the main generator supply to the preferred power system, or to any other supply. It should also be established that the circuitry required to perform these transfer functions has the capability of being tested during plant operation.

4. To assure that the requirements of GDC 33, 34, 35, 38, 41 and 44 are satisfied, the electrical schematics of the systems required for reactor coolant makeup, residual heat removal, emergency core cooling, containment heat removal, containment atmosphere cleanup, and cooling water should be examined to assure that the circuits from the preferred power system can supply these systems assuming a single failure in these systems. Each of the circuits should be physically separate and independent of the other. If the minimum design required by GDC 17 is provided, the immediately available preferred circuit must be made available to the redundant portions of these systems.

5. To assure that the requirements of GDC 1 are satisfied, it should be determined that the design criteria and quality group classifications for all equipment conform to current codes and standards. The QAB will determine the adequacy of the quality assurance program.

6. To assure that the requirements (excluding seismic) of GDC 2 are satisfied, the QAB will provide information on the maximum probable flood, wave runup, hurricanes, high and low atmospheric temperatures, and rain and snow conditions. This information will be considered during the review to assure that the design minimizes the effects of these conditions. Items such as switchyard and transformer location could be affected by the maximum probable flood, wave runup, or hurricane conditions. Transmission lines and the ability to restore a preferred circuit could be affected by hurricanes, high or low temperatures, or rain and snow conditions.

7. To assure that the requirements of GDC 3 are satisfied, it should be determined that the equipment of the preferred power system is designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions. The review of the design criteria for the equipment should ascertain this. The APCSB will review the fire detection and fire fighting systems in the preferred power system areas to assure that adverse effects of fire are minimized. They will also examine ruptures of the fire fighting system to assure that they do not degrade the safety capability of structures, systems, and components to a condition where essential functions are lost.

8. To assure that the requirements of GDC 4 are satisfied, the APCSB will review the location of structures, systems, and components of the preferred power system to determine the protection provided against dynamic effects, including effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the station. This information will be used to determine the possibility of simultaneous loss of both paths of preferred power.

9. To assure that the requirements of GDC 5 are satisfied, the structures, systems, and components of the preferred power systems will be examined to identify any that are shared between units of a multi-unit station. These will be reviewed to ascertain that they are capable of performing all required safety functions in the event of an accident in one unit, with a simultaneous orderly shutdown and cooldown of the remaining units. Review of the design criteria should establish that the capacity and capability of incoming lines, power sources, and transformers for each required circuit have margin to achieve this. Spurious or false accident signals should not overload these circuits. SRP 8.3 further discusses spurious or false accident signal considerations.

10. To assure that the requirements of GDC 13 are satisfied, the preferred power system instrumentation provided to monitor variables and systems over anticipated ranges for normal operation, anticipated abnormal occurrences, and accident conditions should be identified during the electrical schematic and system description review. It should be ascertained that these instruments present status information that can be used to determine the condition of the preferred power system at all times. Review of the electrical schematics should determine that controls (automatic and manual) are provided to maintain these variables and systems within prescribed operating ranges. It should also be determined during the review of the electrical schematics that single failures of these controls and instruments will not violate the requirements of GDC 17.

11. The review of the electrical schematics of the automatic load dispatch system should ascertain that the reactor protection system is designed to prevent any load dispatch system actions that could interfere with safety actions during periods when safety actions are required. The results of analyses of this system should be reviewed to assure that no failure mode of the load dispatch system will cause an incident at the generating station or interfere with any protective action required.

In certain instances, it will be the reviewer's judgement that, for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such a non-uniform placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

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 offsite power system includes two or more identified power sources from the grid, transmission lines (overhead and underground), transmission line towers, transformers, switchyards and switchyard component control systems, switchyard battery systems, the main generator, and disconnect switches used to supply electric power to safety-related and other equipment. The review of the offsite power system for the ____________ plant covered single line diagrams (CP and OL), station layout drawings (CP and OL) and schematic diagrams (OL), and descriptive information. The review included the applicant's proposed design criteria and design bases for the offsite power system and his analyses of the adequacy of those criteria and bases. The review also included the applicant's analyses of the manner in which the design of the offsite power system conforms to the proposed design criteria.

"The basis for acceptance in the staff review has been conformance of the applicant's designs, design criteria, and design bases for the offsite power system to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, staff technical positions, and industry standards. These are listed in Table 8-1. (Table 8-1 should be included in the safety evaluation report, either at this point in 8.2 or in section 8.1.)

"The staff concludes that the design of the offsite power system conforms to applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 8-1, "Acceptance Criteria for Electric Power."

2. Standard Review Plan Appendix 7-B, "General Agenda, Station Site Visits."

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SECTION 8.3.1 A-C POWER SYSTEMS (ONSITE)

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The descriptive information, including functional logic diagrams, functional piping and instrument diagrams, electrical single line diagrams, physical arrangement drawings, and electrical schematics, for the a-c onsite power system, presented in the applicant's safety analysis report (SAR), are reviewed. The intent of the review is to determine that the a-c onsite power system satisfies applicable acceptance criteria and will perform its intended functions during all plant operating and accident conditions.

The a-c onsite power system is referred to in industry standards and regulatory guides as the "standby power system." It includes those power sources, distribution systems, and vital supporting systems provided to supply power to safety-related equipment and capable of operating independently of the offsite power system (referred to as the preferred power system). Diesel generator sets have been widely used as the power source for the standby power supplies and will be covered in this review plan. Other power sources such as nearby hydroelectric, nuclear, or fossil units including gas turbine-generator sets will not be addressed herein. These power sources will continue to be evaluated on an individual case basis until staff technical positions applicable to them are developed. In addition, those interface areas between the standby and preferred power systems at the station distribution system level are within the scope of review of this plan insofar as they relate to the independence of the standby power system.

The EICSB will pursue the following phases in the review of the standby power system during both the construction permit (CP) and operating license (OL) stages of the licensing process:

1. System Redundancy Requirements

The standby power system is reviewed to determine that the required redundancy of safety-related components and systems is maintained in the standby power system with regard to both power sources and associated distribution systems. This will include an examination of the standby power network configuration including the power supply feeders, switchgear arrangement, loads supplied from each bus, and power connections to the instrumentation and control devices of the power system.

2. Conformance with the Single Failure Criterion

In establishing the adequacy of this system to meet the single failure criterion, both electrical and physical separation of redundant power sources and associated distribution systems are examined to assess the independence between redundant portions of the system. This will include a review of interconnections between redundant buses, buses and loads, and buses and power supplies; physical arrangement of redundant switchgear and power supplies; and criteria and bases governing the installation of electrical cables for redundant power systems. Should the proposed design provide for sharing of the standby power system between units at ths same site, the adequacy of such a design to meet the single failure criterion is reviewed.

3. Standby and Preferred Power Systems Independence

In evaluating the independence of the standby power system with respect to the preferred power system, the scope of review extends to the station distribution load centers which are powered from the unit auxiliary transformers and the startup transformers (considered for the purposes of this plan as the offsite or preferred power sources). It includes the supply breakers connecting the "low" side of these transformers to the distribution buses. This evaluation includes a review of the electrical protective relaying circuits and power supplies to assure that in the event of a loss of preferred power, the independence of the standby power system is established through prompt opening of isolation-feeder breakers. Also, the capability of the preferred power system circuits to deliver power to the safety-related buses is reviewed to assure that no single failure will result in loss of the minimum required redundancy of the preferred power circuits to the safety-related buses.

4. Standby Power Supplies

Design information and analyses demonstrating the suitability of the diesel generators as standby power supplies are reviewed to assure that the diesel generators have sufficient capacity, capability, and reliability to perform their intended function. This will include an examination of the characteristics of each load and the length of time each load is required, the combined load demand connected to each diesel generator during the "worst" operating condition, automatic and manual loading and unloading of each diesel generator, voltage and frequency recovery characteristics of the diesel generators, continuous and short-term ratings for the diesel generators, acceptance criteria with regard to the number of successful diesel generator tests and allowable failures to demonstrate acceptability, and starting and load shedding circuits. In addition, where the proposed design provides for the connection of non-safety loads to the diesel generators or sharing of diesel generators between nuclear units at the same site, particular review emphasis is given to the possibility of marginal capacity and degradation of reliability that may result from such design provisions.

5. Identification of Cables, Cable Trays, and Terminal Equipment
The means proposed for identifying the standby power system cables, cable trays, and terminal equipment as safety-related equipment in the plant are reviewed. Also, the identification scheme used to distinguish between redundant cables, cable trays, and terminal equipment of the power system is reviewed.

6. Vital Supporting Systems
The instrumentation, control circuits, and power connections of vital supporting systems are reviewed to determine that they are designed to the same criteria as those for the Class IE loads and power systems that they support. This will include an examination of the vital supporting system component redundancy; power feed assignment to instrumentation, controls, and loads; initiating circuits; load characteristics; equipment identification scheme; and design criteria and bases for the installation of redundant cables.

7. System Testing and Surveillance
Preoperational and initial start-up test programs and periodic onsite testing capabilities are reviewed. The means proposed for automatically monitoring the status of system operability are reviewed.

8. Other Review Areas
Other areas of review associated with this system that are covered elsewhere are as follows:

   a. Environmental design and qualification testing of electrical equipment are addressed in Standard Review Plan (SRP) 3.11.
   b. Onsite d-c control power feeds to the standby power system are addressed in SRP 8.3.2.
   c. Technical specification requirements imposed upon the operation of the standby power system are discussed in Chapter 16 of the SAR. Assistance and consultation are provided in accordance with the review procedures in SRP 8.1.
   d. The APCSB, under the 9.5 standard review plans, will identify and evaluate the adequacy of those auxiliary systems that are vital to the proper operation of the standby power system and its connected Class IE loads. These include such systems as the heating and ventilation systems for switchgear and diesel generator rooms and all diesel generator auxiliary systems such as the cooling water system, combustion air supply system, starting system, fuel oil storage and transfer system, and fire detection and protection system. In particular, it will determine that the piping, ducting, and valving arrangement of redundant vital auxiliary supporting systems meet the single failure criterion. In addition, the APCSB will examine the physical arrangement of components and structures for Class IE systems and their supporting auxiliary systems, and determine that single events and accidents will not disable redundant features.

e. The CSB, under the 6.2 standard review plans, will identify those containment ventilation systems provided to maintain a controlled environment for safety-related instrumentation and electrical equipment located inside the containment.

f. The MEB, under SRP 3.10, will review the criteria for seismic qualification and the test and analysis procedures and methods to assure the operability of Category I instrumentation and electrical equipment, including cable trays, switchgear, control room boards, and instrument racks and panels, in the event of a seismic occurrence.

g. The QAB, under SRP 17.1 and 17.2, will verify the adequacy of the quality assurance program for the installation, inspection, and testing of Class IE instrumentation and electrical equipment and will coordinate the requirements for the technical specifications.

h. The RSB, under the 5.4, 6.3, and 15.0 standard review plans, will identify the engineered safety feature (ESF) and safe shutdown loads and systems and will verify that the minimum time intervals for the connection of ESF loads to the standy power system during accident conditions are satisfactory.

II. ACCEPTANCE CRITERIA

In general, the standby power system is acceptable when it can be concluded that this system has the required redundancy, meets the single failure criterion, and has the capacity, capability, and reliability to supply power to all required safety loads. Table 8-1 lists general design criteria (GDC), standards of the Institute of Electrical and Electronic Engineers (IEEE), regulatory guides, and branch technical positions utilized as the bases for arriving at this conclusion. Also, Table 8-1 includes those evaluation guides used by the reviewer as aids in ascertaining that the criteria have been met. Section III of this plan discusses the application of these evaluation guides to the review. The application of the acceptance criteria to the areas of review described in Section I of this plan is as follows:

1. System Redundancy Requirements

GDC 33, 34, 35, 38, 41, and 44 set forth requirements with regard to the safety systems that must be supplied by the standby power system. Also, these criteria state that safety system redundancy should be such that for standby power system operation (assuming preferred power is not available), the system safety function can be accomplished assuming a single failure. The acceptability of the standby power system with regard to redundancy is based on conformance to the same degree of redundancy required of safety-related components and systems by these GDC.

2. Conformance with the Single Failure Criterion

As required by GDC 17, the standby power system must be capable of performing its safety function assuming a single failure. To meet this requirement, electrical independence between redundant portions of this system must be maintained. An acceptable design in this regard is one that conforms to IEEE Std 308 and follows the recommendations of Regulatory Guide 1.6. Should the proposed design provide for sharing of the standby power system between units at the same site, the governing criteria stated in IEEE Std 308 are not explicit enough to be used as the basis for acceptance. Therefore, the acceptability of such a design to meet the single failure criterion is based on the design satisfying the recommendations of Regulatory Guide 1.81. This Guide sets forth acceptable bases for implementing the requirements of GDC 5, "Sharing of Structures, Systems, and Components." To assure that physical independence of redundant equipment, including cables and cable trays, is maintained in accordance with meeting the requirements of GDC 2, 3, and 4, an acceptable design arrangement must satisfy the requirements set forth in IEEE Std 384, as augmented by Regulatory Guide 1.75.

3. Standby and Preferred Power Systems Independence
The basis for acceptance is that no single failure including single protective relay, interlock, or switchgear failure, causing the loss of preferred power, will prevent the separation of the preferred power system from the standby power system or limit the standby power system in accomplishing its intended function. To assure the independence of the standby power system in the event of a failure in the preferred power system, an acceptable design must satisfy the requirements of GDC 17. In addition, the preferred and standby power supplies should not have common failure modes, as required by Section 5.2.1 (5) of IEEE Std 308. In assuring that the design of the preferred power circuits to the safety-related buses is consistent with satisfying the power availability requirements of GDC 17, as supplemented by GDC 34, 35, 38, 41 and 44, an acceptable design must be capable of withstanding the effects of a single failure without a reduction of the capability of the preferred power circuits to less than the minimum required for safety.

4. Standby Power Supplies
   a. The capacity, capability, and reliability of the standby power supply diesel generator sets are acceptable if the basis for selection of the diesel generator sets follows the recommendations of Regulatory Guide 1.9.
   b. If the proposed design provides for sharing of the standby power system between units at the same site, the acceptance criteria utilized in determining that such a design complies with the requirements of GDC 5 are given in Regulatory Guide 1.81. This guide sets forth two principal positions. Position 2 is being applied to reviews for all operating license and construction permit applications docketed prior to June 1, 1973. In essence, Position 2 permits sharing if the standby power system has sufficient capacity and capability to supply the minimum ESF loads in any unit and also the equipment needed to safely shut down the remaining units. The capacity and capability are acceptable if system safety functions can be accomplished in the event of an accident in one unit, assuming a single failure or a spurious or false accident signal from another unit and loss of preferred power. Position 3 is being applied to construction permit applications docketed after June 1, 1973. It prohibits the sharing of standby power systems between nuclear units.
   c. Should the proposed design provide for the connection and disconnection of non-class IE loads to and from the Class IE standby power supplies, it should conform to IEEE Std 384, as augmented by Regulatory Guide 1.75, with respect to the role

isolation devices play in this regard. The design must be such as to assure that the interconnections and the added non-class IE loads will not result in any degradation of the Class IE system.

d. Diesel generator qualification testing programs are acceptable if they satisfy Position 5 of Regulatory Guides 1.6 and 1.9, as augmented by Branch Technical Position EICSB 2.

e. Regarding the design of thermal overload protection for motors of motor-operated safety-related valves, the acceptability of the design is based on Branch Technical Position EICSB 27.

5. Identification of Cables, Cable Trays, and Terminal Equipment

The method used for identifying standby power system cables, cable trays, and terminal equipment as safety-related equipment in the plant, and the identification scheme used to distinguish between redundant cables, cable trays, and terminal equipment are acceptable if in accordance with IEEE Std 384 as supplemented by Regulatory Guide 1.75.

6. Vital Supporting Systems

The instrumentation, controls, and electrical equipment for those supporting systems identified as vital to the proper functioning of Class IE systems are acceptable if the design conforms to the same criteria as for the Class IE systems supported.

7. System Testing and Surveillance

To assure that the preoperational and initial startup test programs for the standby power system meet the requirements of GDC 1, they must be in accordance with Regulatory Guide 1.68, as augmented by Regulatory Guide 1.41. To assure that the periodic onsite testing capabilities satisfy the requirements of GDC 18 and 21, an acceptable testing program should include the positions of Regulatory Guide 1.22. With regard to surveillance of the standby power system operability status, an acceptable design should satisfy the positions of Regulatory Guide 1.47, as augmented by Branch Technical Position EICSB 21.

8. Fire Stops and Seals

The basis for acceptance of fire stops and seals is the use of noncombustible and heat resistant materials as described in GDC 3 at all penetrations of walls and floors and at specified intervals of longer cable runs. In addition, it should be acceptably demonstrated that the means provided for fire detection and extinguishment will prevent a fire in one system from propagating to another redundant system within the time frame constraints of the fire stops themselves.

9. Other Review Areas

For those areas of review identified as being the responsibility of other branches, the acceptance criteria and their application to the areas of review are included in the appropriate standard review plans. However, there are some acceptance criteria that are commonly used by both primary and secondary review branches as the basis for determining that a design is acceptable. For the standby power system, these criteria and their application to the areas of review are as follows:

a. Seismic Design Requirements

In determining the adequacy of the seismic design of Category I instrumentation and electrical equipment, both the MEB and EICSB will perform reviews in this regard to ascertain that the proposed design satisfies such standards as IEEE Std 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," as supplemented by Branch Technical Position EICSB 10, "Electrical and Mechanical Equipment Seismic Qualification Program."

b. Quality Assurance

To assure that the requirements of GDC 1 are met in the standby power system, the quality assurance program for the Class IE instrumentation and electrical equipment must satisfy the requirements of such standards as IEEE Std 336, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations," as augmented by Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment." Both the QAB and EICSB will perform reviews in this regard to ascertain that the proposed quality assurance program is consistent with the acceptance criteria.

III. REVIEW PROCEDURES

The main objectives in the review of the standby power system are to determine that this system has the required redundancy, meets the single failure criterion, and has the capacity, capability, and reliability to supply power to all required safety loads. In the CP review, the descriptive information, including the design bases and their relation to the acceptance criteria, preliminary analyses, electrical single line diagrams, functional logic diagrams, preliminary functional piping and instrumentation diagrams (P&IDs), and preliminary physical arrangement drawings are examined to determine that there is reasonable assurance that the final design will meet these objectives. At the OL stage, these objectives are verified during the review of final electrical schematics, functional P&IDs, and physical arrangement drawings and are confirmed during a visit to the site. To assure that these objectives have been met in accordance with the requirements of the criteria, the review is performed as detailed below.

In addition to the review procedures of the EICSB, this section identifies those aspects of the review that will be accomplished by the secondary review branches.

1. System Redundancy Requirements

Based on the information provided by the RSB with regard to the required redundancy of safety-related components and systems (GDC 33, 34, 35, 38, 41, and 44), the descriptive information including electrical single line diagrams (CP and OL stage), functional P&IDs (CP and OL stage), and electrical schematics (OL stage) is reviewed to verify that this redundancy is reflected in the standby power system with regard to both power sources and associated distribution systems. Also, it is verified that redundant safety loads are distributed between redundant distribution systems, and that the instrumentation and control devices for the Class IE loads and power system are supplied from the related redundant distribution systems.

2. Conformance with the Single Failure Criterion

In evaluating the adequacy of this system in meeting the single failure criterion (GDC 17), both electrical and physical separation of redundant power sources and distribution systems, including their connected loads, are reviewed to assess the independence between redundant portions of the system.

To assure electrical independence, the design criteria, analyses, description, and implementation as depicted on functional logic diagrams, electrical single line diagrams, and electrical schematics are reviewed to determine that the design meets the requirements set forth in IEEE Std 308 and satisfies the positions of Regulatory Guide 1.6. Additional guidance in evaluating this aspect of the design is derived from IEEE Std 379, "Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems," as augmented by Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." Since IEEE Std 308 does not set forth specific criteria governing the design of the circuits that initiate and control standby power, the reviewer utilizes IEEE Std 279 as an evaluation guide to ascertain that the designs of these circuits satisfy the same single failure requirements as protection systems. Other aspects of the design where special review attention is given to ascertain that the electrical independence has not been compromised are as follows:

a. Should the proposed design provide for sharing of the standby power system between units at the same site, the criteria of IEEE Std 308 governing the sharing of this system between units are not specific enough to be used as the basis for assessing the adequacy of the design in meeting the requirements of GDC 5 and satisfying the single failure criterion. Therefore, the acceptability of such a design is determined by reviewing the proposed system design criteria and electrical schematics and analyses substantiating the adequacy of the design to withstand the consequences of electrical faults and failures in one unit with the respect to the others. Generally, the reviewer is guided by the requirements set forth in Position 2 of Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," for CP applications docketed before June 1, 1973 and for OL applications. Position 3 of this Regulatory Guide prohibits the sharing of standby power systems between nuclear units for construction permit applications docketed after June 1, 1973. Further details of the review with regard to Position 2 on sharing of the standby power system between units are covered in Item 4, below.
b. The interconnections between redundant load centers through bus tie breakers and multi-feeder breakers used to connect extra redundant loads to either of the redundant distribution systems are examined to assure that no single failure in the interconnections will cause the paralleling of the standby power supplies. To assure this, the control circuits of the bus tie breakers or multi-feeder breakers must preclude automatic transferring of load centers or loads from the designated supply to the redundant counterpart upon loss of the designated supply (Position 4 of Regulatory Guide 1.6). Regarding the interconnections through

bus tie breakers, an acceptable design will provide for two tie breakers connected in series and physically separated from each other in accordance with the acceptance criteria for separation of Class IE systems which is discussed below. Further, the interconnection of redundant load centers must be accomplished only manually. With respect to the interconnections through the multi-feeder breakers supplying power to extra redundant loads, the review relates to the utilization of the extra redundant unit as one of the required operating units (if the substituted for normal unit is inoperable). If this is the selected mode of operation prior to an accident concurrent with the loss of offsite power, it is verified by reviewing the breaker arrangement and associated control circuits that no single failure in the feeder breaker which is not connected to the extra redundant unit could cause the closing of this breaker resulting in the paralleling of the power supplies. To assure against compromising the independence of the redundant power systems under this situation, an acceptable design for connecting extra redundant loads to either distribution system will provide for at least dual means for connecting and isolating each load from each redundant bus. Such a design must also meet the acceptance criteria for electrical and physical separation of Class IE systems. In addition, the provisions of the design to automatically break all the interconnections (e.g., open tie and multi-feeder breakers) between redundant load centers immediately following an accident condition concurrent with the loss of offsite power are reviewed to ascertain that the independence of the redundant portions of this system is established given a single failure.

c. To assure physical independence, the criteria governing the physical separation of redundant equipment, including cables and cable trays, and their implementation as depicted on preliminary (CP stage) or final (OL stage) physical arrangement drawings are reviewed to determine that the design arrangements satisfy the requirements set forth in IEEE Std 384 as augmented by Regulatory Guide 1.75. This standard and regulatory guide set forth acceptance criteria for the separation of circuits and electrical equipment contained in or associated with the Class IE power system. In essence, the review objective is to determine that the design provides for redundant portions of this system to be located in physically separated seismic Category I structures (GDC 2). It is verified that each structure has independent heating and ventilation (H&V) systems (including supply and exhaust pipes or ducts) to assure against single events and accidents from disabling redundant features (GDC 3, 4). The APCSB has primary responsibility in the review of the design arrangement of the Class IE systems and their vital supporting systems, except for the cable design which is the responsibility of the EICSB. Within the scope of review of this area, the APCSB will also verify the adequacy of physical barriers such as doors separating redundant portions of this system to assure that events such as fire and flooding in one structure will not be propagated to other redundant equipment structures (GDC 3, 4). To determine that the independence of the redundant cable installation is consistent with satisfying the requirements set forth in IEEE Std 384 as supplemented by Regulatory Guide 1.75, the proposed design criteria governing the separation of Class IE cables and raceways are reviewed including such criteria as those for cable derating; cable tray filling; cable routing in containment, penetration areas, cable spreading rooms, control rooms and other congested areas; sharing of cable trays with non-safety-related cables or with cables of the same system or other systems; prohibiting cable splices in conduits and trays; control wiring and components associated with Class IE electric systems in control boards, panels, and relay racks; and fire barriers and separation between redundant trays. With regard to determining the adequacy of the physical independence of redundant cables through penetration areas, the reviewer utilizes, in addition to IEEE Std 384 and Regulatory Guide 1.75, IEEE Std 317 as augmented by Regulatory Guide 1.63 as evaluation guides to ascertain that the electric penetration assemblies are designed in accordance with the requirements for Class IE equipment.

3. Standby and Preferred Power Systems Independence

In ascertaining the independence of the standby power system with respect to the preferred power system, the electrical ties between these two systems as well as the physical arrangement of the interface equipment are reviewed to assure that no single failure will prevent the separation of the redundant portions of the standby power system from the preferred power system when required. The scope of review extends to the supply breakers connecting the low side of the unit auxiliary transformers and start-up transformers (referred to as the offsite or preferred power supplies) to the station non-Class IE distribution buses through which power is made available to the Class IE buses. The number of electrical circuits from the preferred power supplies to the safety buses are to be consistent with satisfying the requirements of redundancy and independence of GDC 34, 35, 38, 41, and 44. That is, for preferred power system operation (assuming standby power is not available), the system safety function can be accomplished assuming a single failure.

To determine that the physical independence of the preferred power circuits to the Class IE buses is consistent with satisfying the requirements of GDC 17 and Section 5.2.1(5) of IEEE Std 308, the physical arrangement drawings are examined to verify that each circuit is physically separate and independent from its redundant counterparts. In addition, the final feeder-isolation breaker in each circuit through which preferred power is supplied to the safety buses must be designed and physically separated in accordance with the requirements for Class IE systems. Following the loss of preferred power, the safety buses are powered solely from the standby power supplies. Under this situation, the design of the feeder-isolation breaker in each preferred power circuit must preclude the automatic connection of preferred power to the respective safety bus upon the loss of standby power. In this regard, an acceptable design will include the capability for restoring preferred power to the respective safety bus by manual actuation only.

In assessing the adequacy of the electrical ties between the standby and preferred power systems, and the capability of the preferred power circuits to deliver power to the safety-related buses, both primary and secondary backup protective relaying schemes and their coordination, relay settings, and assigned control power supplies are reviewed to assure that in the event of an electrical fault, occurring between the preferred power transformer supply breakers and the safety buses, no single failure will result in reducing the number of preferred power cirucits to less than the minimum required for safety, or prevent the separation of the affected circuit from the respective redundant portion of the standby power system. In addition, it is verified that no single protective relay or interlock failure will prevent separation of the required redundant portions of the standby power system from the preferred power system upon loss of the latter.

In reviewing the mode of operation where both power systems are being operated in parallel (such is the case during full load testing of standby power supply diesel generator sets), the interlock scheme including electrical protective relay coordination and settings are closely examined to verify that the independence of the required redundant portions of the standby power system is established upon a failure in the preferred power system. The event of concern under this mode of operation is an accident concurrent with a loss of offsite power and a single failure preventing the opening of the feeder-isolation breaker through which the paralleling of the power systems was being accomplished. Because the signal to start the diesel generator sets is normally derived from undervoltage relays and under this situation the voltage is maintained above the trip relay settings by the diesel generator under test, the remaining redundant diesel generators will not be commanded to start running. Consequently, the added capacity resulting from the connection of non-Class IE loads to the diesel generator under test will cause the tripping of this diesel due to overload. The end result could be the total loss of power to the safety buses. However, this power interruption could be of momentary duration if the remaining redundant diesel generators are commanded automatically to start by undervoltage relay action immediately after total power is lost. The diesel generator under test will be inoperable due to the self-locking feature preventing restarting after an overload trip condition. The reviewer ascertains that the time delay introduced in making power available to the safety buses as a result of this event is within the response time limits assumed in the accident analyses. Included is verification that subsequent failures such as those resulting from improper electrical relaying coordination and self-locking features will not impair the automatic starting of the remaining redundant diesel generators required to meet minimum safety requirements. If the time delay introduced in making power available to the safety buses is not tolerable, it must be demonstrated that either the probability of occurrence of this event is low when compared to the frequency and duration of testing each diesel, or the design must provide diverse automatic signals, other than undervoltage, to assure the availability of standby power to the safety buses.

As an outcome of reviewing the parallel operation of the preferred and standby power systems, the use of the standby power supply diesel generator sets to supply power to the electrical system during peak load demand periods was found by the staff to be unacceptable. The basis for this conclusion is that the required frequent interconnections of the preferred and standby power supplies do not minimize the probability of their coincident loss (GDC 17) nor can the design be made immune to common failure modes (Section 5.2.1(5) of IEEE Std 308). Further details amplifying the basis for this conclusion are included in Branch Technical Position EICSB 8 which sets forth the basis for prohibiting the use of diesel generator sets for purposes other than emergency standby power supplies.

4. <u>Standby Power Supplies</u>

In assuring that the requirements of GDC 17 and IEEE Std 308 have been met with regard to the standby power supply diesel generator sets having sufficient capacity, capability, and reliability to supply the required distribution system loads, the design bases, design criteria, analyses, description, and implementation as depicted on electrical drawings and functional P&IDs are reviewed to verify that the bases for selection of the diesel generator sets satisfy the positions of Regulatory Guide 1.9. Supplemental guidance for evaluating the suitability of the diesel generators as standby power supplies is obtained from IEEE Std 387, "Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations." Specifically, the reviewer first becomes familiar with the purpose and operation of each safety system, including system component arrangement as depicted on functional P&IDs, expected system performance as established in the accident analyses, modes of system operation and their interactions during normal and accident conditions, and interactions between systems. Following this, it is verified that the tabulation of all safety-related loads to be connected to each diesel generator is consistent with the information establishing the safety-related systems and loads and their required redundancy. The characteristics of each load (such as motor horsepower, volt-amp rating, inrush current, starting volt-amps and torque), the length of time each load is required, and the basis used to establish the power required for each safety load (such as motor nameplate rating, pump run-out condition, or estimated load under expected flow and pressure) are utilized to verify the calculations establishing the combined load demand to be connected to each diesel during the "worst" operating condition. In applying this combined load demand to the selection of each diesel generator capacity, an acceptable design must satisfy Positions 1 and 2 of Regulatory Guide 1.9.

To assure that each diesel generator is capable of starting and accelerating to rated speed all the connected loads in the required sequence and within the minimum time intervals established by the accident analyses, the reviewer examines for each diesel generator the loading profile curves, voltage and frequency recovering characteristic curves, and the response time of the excitation system to load variations. This examination must verify that the capability of each diesel generator to respond to voltage and frequency variations satisfies Position 4 of Regulatory Guide 1.9. In addition, the adequacy of the circuit design for starting and disconnecting and connecting safety loads from and to each diesel generator is checked. This includes a review of the starting initiating circuits; manual and automatic sequential loading and unloading circuits; interrupting capacity of switchgear, load centers, control centers, and distribution panels; grounding requirements; and electrical protective relaying circuits including their coordination, relay settings, and assigned control power supplies for each load and each diesel generator. In reviewing the criteria governing the design of the thermal overload protection for motors of motor-operated safety-related valves, the reviewer is guided by Branch Technical Position EICSB 27.

Regarding the review of the electrical protective trip circuits of the diesel generator sets, Branch Technical Position EICSB 17 is utilized as an evaluation guide. Although this guide sets forth specific recommendations for a particular plant, it can be used to ascertain that the design of these circuits are consistent with minimizing the likelihood of false diesel generator trips during emergency conditions. The capability of the automatic sequential loading circuits to reset during a sustained low voltage condition on the diesel generators is reviewed to assure that upon restoration of normal voltage, the Class IE loads can be connected in the prescribed sequence. Otherwise, the reconnection of all the loads at the same time could result in an overload condition causing the trip of the respective diesel generator. In assuring that those Class IE loads being powered through latched-type breakers are capable of being reconnected to their respective buses after restoration of power, the design must provide for resetting the breaker anticyle feature when there is an undervoltage condition. The normal function of this feature is to prevent immediate reclosure of a breaker following a trip.

Where the proposed design provides for the sharing of diesel generators between units at the same site, and connection and disconnection of non-Class IE loads to and from the Class IE distribution buses, particular attention is given in the review to assure that the implementation of such design provisions does not compromise the capacity, capability, or reliability of the standby power supplies.

GDC 5 prohibits sharing unless it can be shown that the diesel generators are capable of performing all required safety functions in the event of an accident in one unit and an orderly shutdown and cooldown of the remaining units. In assuring that the proposed design for sharing diesel generators between units meets the requirements of GDC 5 and 17 as supplemented by GDC 34, 35, 38, 41, and 44 and satisfies the positions of Regulatory Guide 1.9, the reviewer is guided by Regulatory Guide 1.81. This guide sets forth two principal positions. Position 3 applies to those construction permit applications docketed after June 1, 1973, and prohibits the sharing of standby power systems between units. Conformance of the design with Position 3 is verified by reviewing the descriptive information including electrical drawings to assure that the standby power system of each unit is electrically independent with respect to the standby power system of other units.

Position 2 establishes acceptable bases under which sharing of standby power systems between units is permitted. Conformance with Position 2 as regards the adequacy of diesel generator capacity and capability under the sharing mode of operation is verified by following the procedure discussed above for tabulating and summing all loads. In particular, the load tabulation and calculations establishing the diesel generator capacity are examined to assure that the selected capacity is sufficient to power the minimum ESF loads in any unit and safely shut down the remaining units, in the event of an accident in one unit and a single failure or spurious or false accident signal from another unit and loss of preferred power to all the units. In addition, the physical arrangement of instrumentation and control devices on control room panels and consoles in one unit with respect to the other units is examined to assure that the design minimizes the coordination needed between unit operators to accomplish sharing of the standby power systems.

In the absence of specific criteria in IEEE Std 308 governing the connection and disconnection of non-Class IE loads to and from the Class IE distribution buses, the review of the interconnections will consider isolation devices as defined in IEEE Std 384 and augmented by Regulatory Guide 1.75 to determine the adequacy of the design. In assuring that the interconnections between non-Class IE loads and Class IE buses will not result in the degradation of the Class IE system, the isolation device through which standby power is supplied to the non-Class IE load, including control circuits and connections to the Class IE bus, must be designed to meet Class IE requirements. Should the standby power supplies not have been sized to accommodate the added non-Class IE loads during emergency conditions, the design must provide for the automatic disconnection of those non-Class IE loads upon the detection of the emergency condition. This action must be accomplished whether or not the load was already connected to the power supply. Further, the design must also prevent the automatic or manual connection of these loads during the transient stabilization period subsequent to this event.

The description of the qualification test program (CP stage) and the results of such tests (OL stage) for demonstrating the suitability of the diesel generators as standby power supplies are judged to be acceptable if they satisfy the acceptance criteria stated in Section II.4 of this SRP. In the event that diesel generators have not been selected for a particular plant, a commitment from the applicant to obtain diesel generators of a design that have been previously qualified for use in nuclear power plant applications, or to perform qualification tests on diesel generators of a new design in accordance with the acceptance criteria is considered acceptable at the CP stage of review.

The APCSB will review the adequacy of the non-electrical aspects of the design for those auxiliary systems that have been identified as essential to the operation of Class IE loads and power supplies. This will include verification that there is seismic Cagegory I onsite fuel oil storage capacity for operation at full rated load of one redundant diesel generator for at least seven days.

5. Identification of Cables, Cable Trays, and Terminal Equipment

The identification scheme used for Class IE cables, raceways, and terminal equipment in the plant and Class IE internal wiring in the control boards is reviewed to see that it is consistent with IEEE Std 384 as supplemented by Regulatory Guide 1.75. This includes the criteria for differentiating between safety-related cables, cable trays, and terminal equipment of different channels or divisions, non-safety-related cable which is run in safety trays, non-safety-related cable which is not associated physically with any safety division, and safety-related cables, raceways, and terminal equipment of one unit with respect to the other units at a multi-unit site.

6. <u>Vital Supporting Systems</u>

The APCSB and EICSB will review those auxiliary systems identified as being vital to the operation of Class IE loads and systems. The EICSB reviews the instrumentation, control, and electrical aspects of the vital supporting systems to assure that their design conforms to the same criteria as those for the Class IE systems that they support. Hence, the review procedure to be followed for ascertaining the adequacy of the vital supporting systems is the same as that discussed herein for Class IE systems. In essence, the reviewer first becomes familiar with the purpose and operation of each vital supporting system, including its component arrangement as depicted on functional P&IDs. Subsequently, the design criteria, analyses, and description and implementation of the instrumentation, control, and electrical equipment as depicted on electrical drawings, are reviewed to verify that the design is consistent with satisfying the acceptance criteria for Class IE systems. In addition, it is verified that the vital supporting system loads have been accounted for in the calculations for sizing the Class IE power supplies. Further, the power feed assignments for the vital supporting system redundant instrumentation, control devices, and loads are examined to verify that they are powered from the same redundant distribution system as the Class IE system that they support.

The APCSB reviews the non-electrical aspects of the vital supporting systems to verify that the design, capacities, and physical independence of these systems are adequate for their intended functions. Included is a review of the heating and ventilation (H&V) systems identified as necessary to Class IE systems, such as the H&V systems for the electrical switchgear and diesel generator rooms. The APCSB will verify the adequacy of the H&V system design to maintain the temperature and level of humidity in the room required for proper operation of the safety equipment during both normal and accident conditions. It will also verify that redundant H&V systems, as well as other redundant vital supporting systems such as the ones associated with the diesel generator units (i.e., cooling water system, combustion air supply system, starting system, fuel oil storage and transfer system, and fire detection and protection system) are located in the same enclosure as the redundant unit they serve, or are separated in accordance with the same criteria as those for the Class IE systems they support. Other aspects of the review by the APCSB are to determine that the diesel generator combustion air quality is such that it will not impair the starting and continuous running reliability of the unit and whether or not it is necessary to maintain the cooling water and lubricating oil warm while the diesel engine is on standby to enhance the starting reliability of the unit.

7. <u>System Testing and Surveillance</u>

The proposed preoperational and initial startup test programs for the standby power system including its vital supporting systems are reviewed to verify that the proposed programs are consistent with Regulatory Guides 1.68 and 1.41. In assuring that the proposed periodic onsite testing capabilities of Class IE systems satisfy the requirements of GDC 18 and 21, the descriptive information (CP and OL stages) functional logic diagrams (CP and OL stages), and electrical schematics (OL stage) are reviewed to verify that the design has the built-in capability to permit integral testing of Class IE systems on a periodic basis when the reactor is in operation. The reviewer is guided by the positions in Regulatory Guide 1.22 in determining an acceptable periodic testing program for actuation devices (e.g., breakers) and actuated equipment. Since IEEE Std 308 does not include requirements for periodic testing of the circuits that initiate and control standby power, the reviewer utilizes IEEE Std 279 and IEEE Std 338 as evaluation guides to ascertain that the testing of these circuits, including electrical protective relays, permissives, bypasses, and control devices, is in accordance with the basic requirements for protection systems.

The descriptive information (CP and OL stages) and the design implementation as depicted on electrical drawings (OL stage) of the means proposed for automatically indicating at the system level a bypassed or deliberately inoperative status of a redundant portion of a Class IE system are reviewed to ascertain that the design is consistent with Regulatory Guide 1.47 and Branch Technical Position EICSB 21. This position establishes the basis to be considered in arriving at an acceptable design for the inoperable status indication system.

8. <u>Fire Stops and Seals</u>

In assuring that the requirements of GDC 3 have been met with regard to the fire stops and seals, the list of materials, their characteristics with regard to flammability and fire retardancy, and their fire underwriters rating should be reviewed. All cable and cable tray penetrations through walls and floors as well as any other types of cable ways or conduits should have fire stops installed. A review of the design criteria for fire stops should reveal the maximum physical vertical and horizontal distances between stops on longer cable runs and the testing that demonstrates the fire stops and seals will perform their intended function. Fire barriers are generally rated for a given temperature and a given time interval. The reviewer should determine if the rating of the fire stops is sufficient to allow extinguishment of the fire before it can affect a redundant cabling system. This will require coordination with Auxiliary Power and Conversion Systems Branch, in conjunction with SRP Section 9.5.1.

9. <u>Other Review Areas</u>

For those areas of review identified as being the responsibility of other branches, the review procedures are included in the appropriate standard review plans. However, there are some areas that are commonly reviewed by both primary and secondary review branches. For the standby power system, the review procedures for these areas are as follows:

a. <u>Seismic Design Requirements</u>

The MEB has primary responsibility in assuring that the seismic design of Category I instrumentation and electrical equipment satisfies the MEB acceptance criteria, which include IEEE Std 344. The EICSB supplements the MEB by reviewing the description of the seismic qualification test program (CP stage) and the results of such tests and analyses (OL stage) for demonstrating the capability of Class IE instrumentation, control devices, and associated circuits to withstand the effects of a seismic event. The adequacy of the seismic design for major electrical apparatus (such as the switchgear, motors, and diesel generator sets) and their supports will be determined by the MEB. The EICSB utilizes IEEE Std 344 as supplemented by Branch Technical Position EICSB 10 as the basis for acceptable seismic designs.

b. Quality Assurance

In assuring that the quality of Class IE equipment is commensurate with present codes and standards (GDC 1), the QAB will review the proposed quality assurance program to ascertain that it is consistent with satisfying the QAB acceptance criteria. The EICSB is guided by the requirements set forth in IEEE Std 336, as augmented by Regulatory Guide 1.30, to ascertain that the proposed quality assurance program for Class IE instrumentation and electrical equipment is acceptable.

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 standby power system includes the onsite power sources, distribution systems, vital auxiliary supporting systems, instrumentation, and controls utilized to supply power to safety-related components and systems. The scope of review included the descriptive information (CP and OL), functional logic diagrams (CP and OL), functional piping and instrument diagrams (CP and OL), electrical single line diagrams (CP and OL), preliminary (CP) and final (OL) physical arrangement drawings, and electrical schematics (OL) for the standby power system and for those auxiliary systems that are vital to the proper operation of the Class IE standby power system and its connected Class IE loads. The review has included the applicant's design bases and their relation to the proposed design criteria for the standby power system and for the vital supporting systems and the applicant's analyses of the adequacy of those criteria and bases. The review also has included the applicant's proposed means for identifying safety-related cables, cable trays, and terminal equipment in the plant; the preoperational and initial startup test programs and periodic onsite testing capabilities; the qualification test programs (CP) and the results (OL) demonstrating the suitability of the diesel generators as standby power supplies; the seismic qualification test program (CP) and the results and analyses (OL); and the quality assurance programs for the standby power system."

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the standby power system and vital 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. These are listed in Table 8-1.

"On the basis of our review, we have concluded that the standby power system conforms to applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 8-1, "Acceptance Criteria for Electric Power."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 8.3.2 D-C POWER SYSTEMS (ONSITE)

REVIEW RESPONSIBILITIES

Primary - Electrical, Instrumentation and Control Systems Branch (EICSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Mechanical Engineering Branch (MEB)
Reactor Systems Branch (RSB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The d-c power systems include those d-c power sources and their distribution systems and vital supporting systems provided to supply motive or control power to safety-related equipment. Batteries and battery chargers are used as the power sources for the d-c power system, and inverters are used to convert d-c from the d-c distribution system to a-c instrumentation power as required. Information on the d-c power system presented in the applicant's safety analysis report (SAR) is reviewed by the staff to determine that the d-c power system required for safe operation during all operating and accident conditions meets the requirements of General Design Criteria (GDC) 17 and 18 and are consistent with Regulatory Guide 1.32, applicable industry standards, and staff positions as listed in Table 8-1. For construction permit (CP) applications, the descriptive information presented for the d-c power system should include commitments to meet the acceptance criteria listed below or adequate justification for exceptions taken, preliminary single line diagrams illustrating the redundancy of d-c power supplies, preliminary load assignments, and preliminary physical arrangement drawings illustrating the independence of redundant batteries and distribution circuits. For operating license (OL) applications, the descriptive information presented should include final single line diagrams, electrical schematics, final physical arrangement drawings, and complete load distribution diagrams, as are needed to determine that the d-c power system has sufficient capacity and capability to meet its functional requirements and otherwise satisfies the mandatory design criteria.

The EICSB will pursue the following phases in the review of the d-c power system:

1. The system is reviewed to determine that the required redundancy of components and subsystems is provided. This will require an examination of the d-c power system configuration including power supply feeders, load center arrangements, loads supplied from each bus, and power connections to the instrumentation and control devices of the system.

2. In determining the adequacy of this system to meet the single failure criterion, the electrical and physical separation of redundant power sources and associated distribution systems are examined to assess the independence between redundant portions of the system. This will include a review of the interconnections between redundant buses, buses and loads, and buses and power supplies; the physical arrangement of redundant load centers and power supplies; proposed sharing of the d-c power system between units at the same site; and the design criteria and bases governing the installation of electrical cable for redundant portions of the systems.

3. Design information and analyses demonstrating the suitability of batteries and battery chargers as d-c power supplies are reviewed to assure that they have sufficient capacity, capability, and reliability to perform their intended functions. This will require an examination of the characteristics of each load; the length of time each load is required; the combined load demand connected to each battery or battery charger during the "worst" operating condition; the voltage recovering characteristics of the battery and battery chargers; and the continuous and short term ratings for the battery and battery chargers.

   In addition, where the proposed design provides for the connection of non-safety-related loads to the d-c power system and sharing of batteries and battery chargers between units at the same site, particular review emphasis is given to assuring against marginal capacity and degradation of reliability that may result from implementing such design provisions.

4. The means proposed for identifying the d-c power system cables and cable trays as safety-related equipment in the plant are reviewed. Also, the identification scheme used to distinguish between redundant cables and cable trays of the power system is reviewed.

5. The instrumentation, control circuits, and power connections of vital supporting systems are reviewed to determine that they are designed to the same criteria as those for the Class IE loads and power systems that they support. This will include an examination of the vital supporting system component redundancy, power feed assignment to instrumentation, control of loads, initiating circuits, load characteristics, equipment identification scheme, and design criteria and bases for the installation of redundant cables.

6. Preoperational and initial start-up test programs and periodic onsite testing capabilities are reviewed. The means proposed for automatically monitoring the status of system operability are reviewed.

7. Other areas of review associated with these systems which are covered elsewhere are as follows:
   a. Environmental design and qualification testing of electrical equipment are addressed in Standard Review Plan (SRP) 3.11.
   b. Technical specification requirements imposed upon the operation of the d-c power system are discussed in Chapter 16 of the SAR. Assistance and consultation on technical specifications for the d-c power system are provided in accordance with the procedures stated in SRP 8.1.

The APCSB will evaluate the adequacy of those auxiliary systems that are vital to the proper operation of the d-c power system. These include such systems as the heating and ventilation systems for load center, battery, and battery charger and inverter rooms, and fire detection and protection systems. In particular, the APCSB will determine that the piping, ducting, and valving arrangments of redundant vital auxiliary supporting systems meet the single failure criterion. In addition, the APCSB will examine the physical arrangment of components and structures associated with the d-c power system and its supporting auxiliary aystems and determine that single events and accidents will not disable redundant features.

The CSB will identify those containment ventilation systems provided for maintaining a controlled environment for safety-related instrumentation and electrical equipment located inside the containment.

The MEB reviews the criteria for seismic qualification analyses, and the test and analysis procedures and methods to assure the operability of instrumentation and electrical equipment in the event of a seismic occurrence.

The RSB will identify any differences or changes in the safety-related loads and systems from those stated in the SAR that are needed to assure sufficient capacity.

The QAB will verify the adequacy of the quality assurance program for this system.

II. ACCEPTANCE CRITERIA

The d-c power system is acceptable when it can be concluded that this system has the required redundancy, meets the single failure criterion, and has the capacity, capability, and reliability to supply d-c power to all safety-related loads required by the accident analyses. Table 8-1 lists the criteria that are utilized as the bases for arriving at this conclusion. In addition, the references include those evaluation guides used by the reviewer as aids in ascertaining that the criteria have been met. Section III of this plan discusses the application of these evaluation guides to the review. The application of most of the acceptance criteria to the areas of review described in Section I of this plan is detailed below. The applicability of other criteria listed in Table 8-1 but not specifically addressed above is considered to be self-evident, and their application in the review process is considered self-explanatory.

1. <u>System Redundancy Requirements</u>

GDC 22, 33, 34, 35, 38, 41, and 44 set forth requirements with regard to safety-related systems that must be supplied by the onsite (a-c and d-c) power systems. Also, these criteria state that safety-related system redundancy shall be such that for onsite power system operation (assuming preferred power is not available) the system safety function can be accomplished assuming a single failure. The acceptability of the onsite d-c power system with regard to redundancy is based on conformance to the same degree of redundancy required of safety-related components and systems by these GDC.

2. <u>Conformance with the Single Failure Criterion</u>

As required by GDC 17, the d-c power system must be capable of performing its safety function assuming a single failure. To meet this requirement, electrical independence between redundant portions of this system must be maintained. An acceptable design in this regard must meet the requirements of IEEE Std 308 and satisfy the positions of Regulatory Guide 1.6. Should the proposed design provide for sharing of the d-c power system between units at the same site, the governing criteria stated in IEEE Std 308 are not explicit enough to be used as the basis for acceptance. Therefore, the acceptability of such a design to meet the single failure criterion is based on the design satisfying the recommendations of Regulatory Guide 1.81. This position sets forth acceptable bases for implementing the requirements of GDC 5, "Sharing of Structures, Systems, and Components." To assure that physical independence of redundant equipment, including cables and cable trays, is maintained in accordance with the requirements of GDC 2, 3, and 4, an acceptable design arrangement should satisfy the positions of Regulatory Guide 1.75.

3. <u>Power Supplies</u>

a. The capacity, capability, and reliability of the d-c power supplies is acceptable if the basis for selection of the batteries and battery chargers satisfies the requirements of IEEE Std 308.

b. The Regulatory position in Regulatory Guide 1.81 states that the sharing of d-c power systems between generating units will not be permitted.

c. Should the proposed design provide for the connection and disconnection of non-safety-related loads to and from the standby d-c power supplies, it should conform to Regulatory Guide 1.75 with respect to the role isolation devices play in this regard. The design must be such as to assure that the interconnections and the added non-safety-related loads will not result in any degradation of the safety-related system.

d. Regarding the design of thermal overload protection for motors of motor-operated safety-related valves, the acceptability of the design is based on Branch Technical Position EICSB 27.

4. <u>Identification of Cables and Cable Trays</u>

The method used for identifying d-c power system cables and cable trays as safety-related equipment in the plant, and the identification scheme used to distinguish between redundant cables and cable trays are acceptable if in accordance with Regulatory Guide 1.75.

5. Vital Supporting Systems
The instrumentation, controls, and electrical equipment for those supporting systems identified as vital to the proper functioning of the safety-related systems are acceptable if the design conforms to the same criteria as for the safety-related systems supported.

6. System Testing and Surveillance
To assure that the preoperational and initial start-up test programs for the d-c power system meet the requirements of GDC 1, they must be in accordance with Regulatory Guides 1.68 and 1.41. To assure that the periodic onsite testing capabilities satisfy the requirements of GDC 18 and 21, an acceptable testing program should include the battery capacity tests described in Section 5 of IEEE Std 450 and the positions of Regulatory Guide 1.22. With regard to surveillance of the d-c power system operability status, an acceptable design should satisfy the positions of Regulatory Guide 1.47, as augmented by Branch Technical Position EICSB 21.

7. Other Review Areas
For those areas of review identified as being the responsibility of other branches, the acceptance criteria and their application to the areas of review are included in the appropriate standard review plans. However, there are some acceptance criteria that are commonly used by both primary and secondary review branches as the basis for determining that a design is acceptable. For the d-c power system, these criteria and their application to the areas of review are as follows:

a. Seismic Design Requirements
In determining the adequacy of the seismic design of Category I instrumentation and electrical equipment, both the MEB and EICSB will perform reviews in this regard to ascertain that the proposed design satisfies such standards as IEEE Std 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations," as supplemented by Branch Technical Position EICSB 10, "Electrical and Mechanical Equipment Seismic Qualification Program."

b. Quality Assurance
To assure that the requirements of GDC 1 are met in the d-c power system, the quality assurance program for the safety-related instrumentation and electrical equipment must satisfy the requirements of IEEE Std 336, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations," as augmented by Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment." Both the QAB and EICSB will perform reviews in this regard to ascertain that the proposed quality assurance program is consistent with the acceptance criteria.

III. REVIEW PROCEDURES

The main objectives in the review of the d-c power system are to determine that this system has the required redundancy, meets the single failure criterion, and has the capacity, capability, and reliability to supply d-c power to all required safety-related loads. In the CP review, the descriptive information, including the design bases and their relation to the acceptance criteria, preliminary analyses, electrical single line diagrams, functional logic diagrams, preliminary functional piping and instrumentation diagrams (P&IDs), and preliminary physical arrangement drawings are examined to determine that there is reasonable assurance that the final design will meet these objectives. At the OL stage, these objectives are verified during the review of final electrical schematics, functional P&IDs, and physical arrangement drawings and are confirmed during a visit to the site. To assure that these objectives have been met in accordance with the requirements of the criteria, the review is performed as detailed below.

In certain instances, it will be the reviewer's judgement that for a specific case under review, emphasis should be placed on specific aspects of the design, while other aspects of the design need not receive the same emphasis and in-depth review. Typical reasons for such placement of emphasis are the introduction of new design features or the utilization in the design of design features previously reviewed and found acceptable.

In addition to the review procedures of the EICSB, this section identifies those aspects of the review that will be accomplished by the secondary review branches.

1. System Redundancy Requirements

   Based on the information provided by the RSB with regard to the required redundancy of safety-related components and systems (GDC 33, 34, 35, 38, 41, and 44), the descriptive information including electrical single line diagrams (CP and OL stages), functional P&IDs (CP and OL stages), and electrical schematics (OL stage) is reviewed to verify that this redundancy is reflected in the d-c power system with regard to both power sources and associated distribution systems. Also, it is verified that redundant safety-related loads are distributed between redundant distribution systems, and that the instrumentation and control devices for the safety-related loads and power system are supplied from the related redundant distribution systems.

2. Conformance with the Single Failure Criterion

   In evaluating the adequacy of this system to meet the single failure criterion (GDC 17), both electrical and physical separation of redundant power sources and distribution systems, including their connected loads, are reviewed to assess the independence between redundant portions of the system.

   To assure electrical independence, the design criteria, analyses, description, and implementation as depicted on functional logic diagrams, electrical single line diagrams, and electrical schematics are reviewed to determine that the design meets the requirements set forth in IEEE Std 308 and satisfies the positions of Regulatory Guide 1.6. Additional guidance in evaluating this aspect of the design is derived from IEEE Std 379, "Guide for the Application of the Single-Failure Criterion to

Nuclear Power Generating Station Protection Systems," as augmented by Regulatory Guide 1.53. Since IEEE Std 308 does not set forth specific criteria governing the design of the circuits that initiate and control d-c power, the reviewer utilizes IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," as an evaluation guide to ascertain that the designs of these circuits satisfy the basic single failure requirements of protection systems. Other aspects of the design where special review attention is given to ascertain that the electrical independence has not been compromised are as follows:

The interconnections between redundant load centers through bus tie breakers and multi-feeder breakers used to connect extra redundant loads to either of the redundant distribution systems are examined to assure that no single failure in the interconnections will cause the paralleling of the d-c power supplies. To assure this, the control circuits of the bus tie breakers or multi-feeder breakers must preclude automatic transferring of load centers or loads from the designated supply to the redundant counterpart upon loss of the designated supply (Position 4 of Regulatory Guide 1.6). Regarding the interconnections through bus tie breakers, an acceptable design will provide for two tie breakers connected in series and physically separated from each other in accordance with the acceptance criteria for separation of safety-related systems which is discussed below. Further, the interconnection of redundant load centers must be accomplished only manually.

To assure physical independence, the criteria governing the physical separation of redundant equipment including cables and cable trays, and their implementation as depicted on preliminary (CP stage) or final (OL stage) physical arrangement drawings are reviewed to determine that the design arrangement satisfies the positions of Regulatory Guide 1.75, "Physical Independence of Electric Systems." This guide sets forth acceptance criteria for the separation of circuits and electrical equipment contained in or associated with the safety-related power system. In essence, the review objective is to determine that the design provides for redundant portions of this system to be located in physically separated seismic Category I structures (GDC 2). It is verified that each structure has independent heating and ventilation (H&V) systems (including supply and exhaust pipes or ducts) to assure against single events and accidents from disabling redundant features (GDC 3, 4). The APSCB has primary responsibility in the review of the design arrangement of the Class IE systems and their vital supporting systems, except for the cable design which is the responsibility of the EICSB. The APCSB will also verify the adequacy of physical barriers such as doors separating redundant portions of this system to assure that events such as fire and flooding in one structure will not be propagated to other redundant equipment structures (GDC 3, 4). To determine that the independence of the redundant cable installation is consistent with the position set forth in Regulatory Guide 1.75, the proposed design criteria governing the separation of safety-related cables and raceways are reviewed including such criteria as those for cable derating; cable tray filling; cable routing in containment, penetration.

areas, cable spreading rooms, control rooms, and other congested areas; sharing of cable trays with non-safety-related cables or with cables of the same system or other systems; prohibiting cable splices in conduits and trays; fire detection and protection in the areas where cables are installed; spacing of power and control wiring and components associated with safety-related electric systems in control boards, panels, and relay racks; and fire barriers and separation between redundant trays. With regard to determining the adequacy of the physical independence of redundant cables through penetration areas, the reviewer utilizes, in addition to Regulatory Guide 1.75, IEEE Std 317 as augmented by Regulatory Guide 1.63 as evaluation guides to ascertain that the electric penetration assemblies are designed in accordance with the requirements for safety-related equipment.

3. D-C Power Supplies

In assuring that the requirements of GDC 17 and IEEE Std 308 have been met with regard to the d-c power system (batteries and battery chargers) having sufficient capacity, capability, and reliability to supply the required distribution system loads, the design bases, design cirteria, analyses, description, and implementation as depicted on electrical drawings and performance characteristic curves are reviewed. To establish that the capacity of the d-c supply is adequate to power the prescribed loads, the nameplate capacity claimed in the design bases is checked against the loads identified in electrical distribution diagrams. The capability of the system is reviewed by evaluating the performance characteristic curves that illustrate the response of the supplies to the most severe loading conditions at the plant. The performance characteristic curves would include voltage profile curves, discharge rate curves, and temperature effect curves. The reliability of the d-c supplies should be assured by periodic discharge tests of the batteries as described in IEEE Std 450, and amplified by Branch Technical Position EICSB 6.

The reviewer first becomes familiar with the purpose and the operation of each safety system, including system component arrangements as depicted on functional P&IDs, expected system performance as established in the accident analyses, modes of system operation and interactions during normal and accident conditions, and interactions between systems. Following this, it is verified that the tabulation of all safety-related loads to be connected to each d-c supply is consistent with the information provided by the RSB.

The characteristics of each load (such as motor horsepower and volt-amp ratings, inrush current, starting volt-amps and torque), the length of time each load is required, and the basis used to establish the power required for each safety-related load (such as motor name plate rating, pump run out condition, or estimated load under expected flow and pressure) are utilized to verify the calculations establishing the combined load demand to be connected to each d-c supply during the "worst" operating conditions. In reviewing the design of the thermal overload protection for motors of motor-operated safety-related valves, the reviewer is guided by Branch Technical Position EICSB 27.

Where the proposed design provides for the sharing of d-c supplies between units at the same site, and connection and disconnection of non-safety-related loads to and from the safety-related distribution buses, particular attention is given in the review to assure that the implementation of such design provisions does not compromise the capacity, capability, or reliability of these supplies.

In the absence of specific criteria in IEEE Std 308 governing the connection and disconnection of non-safety-related loads to and from the safety-related distribution buses, the review of the interconnections will consider isolation devices as defined in Regulatory Guide 1.75 and engineering judgement to determine the adequacy of the design. In assuring that the interconnections between non-safety-related loads and safety-related buses will not result in the degradation of the safety-related system, the isolation device through which d-c power is supplied to the non-safety-related load, including control circuits and connections to the safety-related bus, must be designed to meet safety Class IE requirements. Should the d-c power supplies not have been sized to accommodate the added non-safety-related loads during emergency conditions, the design must provide for the automatic disconnection of those non-safety-related loads upon detection of the emergency condition. This action must be accomplished whether or not the load was already connected to the power supply.

The description of the qualification test program (CP stage) and the results of such tests (OL stage) for demonstrating the suitability of the batteries and battery charger as d-c power supplies are judged to be acceptable if they satisfy the acceptance criteria listed in Section II.3 of this SRP or Table 8-1.

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 d-c power system includes the batteries, battery chargers, and distribution centers used to supply power to d-c operated safety-related equipment. The scope of review of the d-c power system included single line diagrams (CP and OL), schematic diagrams (OL), and descriptive information for the d-c power system and for those auxiliary supporting systems that are essential to the operation of the d-c power system. The review has included the applicant's proposed design criteria and his analyses of the adequacy of those criteria and bases. The review also has included the applicant's analyses of the manner in which the design of the d-c power system conforms to the proposed design criteria. The basis for acceptance in the staff review has been conformance of the applicant's design, design criteria, and design bases for the d-c power system to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry standards. These are listed in Table 8-1.

"The staff concludes that the design of the d-c power system conforms to applicable regulations, guides, technical positions, and industry standards and is acceptable."

V. REFERENCES

1. Standard Review Plan Table 8-1, "Acceptance Criteria for Electric Power."

11/24/75

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

TABLE 8-1
ACCEPTANCE CRITERIA FOR ELECTRIC POWER

Table 8-1 contains the acceptance criteria for the review plans of Chapter 8. These acceptance criteria include the applicable general design criteria, IEEE standards, regulatory guides, and branch technical positions (BTP) of the Electrical, Instrumentation and Control Systems Branch (EICSB). The table was prepared by EICSB for use by its members in reviewing Chapter 8 and for use by the secondary review branch reviewers.

The applicability of these criteria to specific sections of Chapter 8 in indicated by an X in the matrix listing of criteria and SAR sections. There is a corresponding similar table (7-1) at the end of Chapter 7 covering the acceptance criteria of safety-related instrumentation and controls. The BTP listed in Tables 7-1 and 8-1 are contained in Appendix 7-A to the Chapter 7 review plans.

ACCEPTANCE CRITERIA FOR ELECTRIC POWER - TABLE 8-1

CRITERIA	TITLE	APPLICABILITY (SAR Section) 8.1	8.2	8.3.1	8.3.2	REMARKS
1. 10 CFR Part 50						
a. 10 CFR §50.34	Contents of Applications: Technical Information	X	X	X	X	
b. 10 CFR §50.36	Technical Specifications	X	X	X	X	
c. 10 CFR §50.55a	Codes and Standards	X	X	X	X	
2. General Design Criteria (GDC), Appendix A to 10 CFR Part 50						
a. GDC-1	Quality Standards and Records	X	X	X	X	
b. GDC-2	Design Bases for Protection Against Natural Phenomena	X	X	X	X	
c. GDC-3	Fire Protection	X	X	X	X	
d. GDC-4	Environmental and Missile Design Bases	X	X	X	X	
e. GDC-5	Sharing of Structures, Systems, and Components	X	X	X	X	
f. GDC-13	Instrumentation and Control	X	X	X	X	
g. GDC-17	Electric Power Systems	X	X	X	X	
h. GDC-18	Inspection and Testing of Electrical Power Systems	X	X	X	X	
i. GDC-21	Protection System Reliability and Testability	X	X	X	X	
j. GDC-22	Protection System Independence	X			X	

TABLE 8-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 8.1	8.2	8.3.1	8.3.2	REMARKS
k. GDC-33	Reactor Coolant Makeup	X	X	X	X	
l. GDC-34	Residual Heat Removal	X	X	X	X	
m. GDC-35	Emergency Core Cooling	X	X	X	X	
n. GDC-38	Containment Heat Removal	X	X	X	X	
o. GDC-41	Containment Atmosphere Cleanup	X	X	X	X	
p. GDC-44	Cooling Water	X	X	X	X	
3. Institute of Electrical and Electronics Engineers (IEEE) Standards:						
a. IEEE Std 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X		X	X	See 10 CFR §50.55a(h) and Reg. Guide 1.62
b. IEEE Std 308-1971	Criteria for Class IE Electric Systems for Nuclear Power Generating Stations	X	X	X	X	See Reg. Guide 1.32.
c. IEEE Std 317-1972	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X		X	X	See Reg. Guide 1.63.
d. IEEE Std 336-1971 (ANSI N45.2.4-1972)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	See Reg. Guide 1.30.
e. IEEE Std 338-1971	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X		X	X	
f. IEEE Std 344-1971 (ANSI N41.7)	Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations	X		X	X	

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TABLE 8-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 8.1	8.2	8.3.1	8.3.2	REMARKS
g. IEEE Std 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems	X		X	X	See Reg. Guide 1.53.
h. IEEE Std 384-1974 (ANSI N41.14)	Criteria for Separation of Class IE Equipment and Circuits	X		X	X	
i. IEEE Std 387-1972 (ANSI N41.13)	Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Stations	X		X		
j. IEEE Std 450-1972	Recommended Practice for Maintenance, Testing and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries	X			X	
4. Regulatory Guides (RG)						
a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	X		X	X	
b. RG 1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	X		X		
c. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X	
d. RG 1.29	Seismic Design Classification	X		X	X	
e. RG 1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	X	X	X	X	
f. RG 1.32	Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"	X	X		X	

TABLE 8-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 8.1	8.2	8.3.1	8.3.2	REMARKS
g. RG 1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	X	X	X	X	
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X	Use in conjunction with Position 3, RG 1.17.
i. RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	X		X	X	
j. RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	X		X	X	
k. RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X	X	X	
l. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2	X	X	X	X	
m. RG 1.75	Physical Independence of Electric Systems	X		X	X	
n. RG 1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	X		X	X	Use in conjunction with BTP-7
o. RG 1.89	Qualification of Class IE Equipment for Nuclear Power Plants	X		X	X	
p. RG 1.93	Availability of Electric Power Sources	X	X	X	X	
5. Branch Technical Positions (BTP) EICSB						
a. BTP EICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X		X	X	
b. BTP EICSB 2	Diesel-Generator Reliability Qualification Testing	X		X		

TABLE 8-1 (CONTINUED)

CRITERIA	TITLE	APPLICABILITY (SAR Section) 8.1	8.2	8.3.1	8.3.2	REMARKS
c. BTP EICSB 6	Capacity Test Requirements of Station Batteries-Technical Specifications	X			X	
d. BTP EICSB 7	Shared Onsite Emergency Electrical Power Systems for Multi-Unit Generating Stations	X		X	X	
e. BTP EICSB 8	Use of Diesel-Generator Sets for Peaking	X		X		
f. BTP EICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X		X	X	
g. BTP EICSB 11	Stability of Offsite Power Systems	X	X			
h. BTP EICSB 17	Diesel Generator Protective Trip Circuit Bypasses	X		X		
i. BTP EICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X	
j. BTP EISCB 27	Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves	X		X	X	

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.1.1 NEW FUEL STORAGE

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Reactor Systems Branch (RSB)
Core Performance Branch (CPB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

Nuclear reactor plants include storage facilities for the dry storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling requirements. The safety function of the storage facility is to maintain the new fuel in a subcritical array during all credible storage conditions. The APCSB reviews the new fuel storage facility design including the fuel assembly storage racks and storage vault.

1. The facility design is reviewed with respect to the following:
   a. The quantity of fuel to be stored.
   b. The design and arrangement of the storage racks for maintaining a subcritical array during all storage conditions.
   c. The degree of subcriticality, and the supporting analysis and associated assumptions.
   d. The effects of external loads and forces on the new fuel storage racks and vault (e.g., safe shutdown earthquake, crane uplift forces).
   e. The effects of sharing in multi-unit complexes, and failures of other plant equipment close to the new fuel storage facility.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluations of the system. The secondary reviews are as follows: The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of facility structures to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The RSB determines that the assigned seismic and quality group classifications for facility components are acceptable. The MTEB verifies, upon request, the compatability of the materials of construction with service conditions. The CPB verifies, upon request, that the k_{eff} of loaded storage racks is acceptable. The RAB reviews the adequacy of the shielding design and the radiation monitoring system.

II. ACCEPTANCE CRITERIA

Acceptability of the new fuel storage facility design as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR, is based on specific general design criteria and regulatory guides, and on independent calculations and staff judgments with respect to facility functions and component selection. Listed below are specific criteria related to the storage facility.

1. The design of the new fuel storage facility is acceptable if the integrated design is in accordance with the following criteria:
   a. General Design Criterion 2, as related to the ability of structures housing the facility and the facility components to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.
   b. General Design Criterion 3, as related to protection against fire hazards.
   c. General Design Criterion 4, with respect to structures housing the facility and the facility components being capable of withstanding the effects of external missiles and internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, such that safety functions will not be precluded.
   d. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.
   e. General Design Criterion 61, as related to the facility design for fuel storage, including the following elements:
      (1) The capability for periodic testing of components important to safety.
      (2) Shielding for radiation protection.
      (3) Provisions for containment or confinement.
   f. General Design Criterion 62, as related to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations.
   g. General Design Criterion 63, as it relates to monitoring systems provided to detect excessive radiation levels.
   h. Regulatory Guide 1.29, as related the seismic design classification of facility components.
   i. Fuel storage capacity and criticality limits as discussed in III.1 and III.2 below.

An additional basis for determining the acceptability of the facility is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design meet the acceptance criteria given in Section II of this plan. For operating license (OL) applications, the review procedures and acceptence criteria 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 review procedures given are for a typical storage system. Any variance of the review, to adjust to a proposed unique design, is such as to assure that the facility design conforms to the criteria in Section II. The reviewer selects and emphasizes material from this review plan as may be appropriate for a particular case.

1. The quantity of new fuel to be stored onsite forms the basis for the design capacity of the vault and the number of storage racks provided. The SAR is reviewed to determine that the facility description section has stated the storage capacity provided by the design. The SAR's for recent light water reactor applications have stated that the storage space provided is consistent with the number of new fuel assemblies used during the refueling cycle. In general, storage capacity for at least one-third of a core is usually provided for each unit of a plant (e.g., 1/3 core for single unit design and 2/3 core for a dual unit design).

2. The information provided in the SAR pertaining to criticality safety of the new fuel storage facility is evaluated based in part on previously approved facilities or on independent calculations by CPB upon request. The facility design criteria, safety evaluation, system description, and the layout drawings for the storage vault and racks are reviewed to verify that:
   a. Criticality information (including the associated assumptions and input parameters) in the SAR must show that the spacing between fuel assemblies in the storage racks is sufficient to maintain the array, when fully loaded and flooded with nonborated water, in a subcritical condition, i.e., k_{eff} of less than about 0.95. Furthermore, the design of the new fuel storage racks will be such that the K_{eff} will not exceed 0.98 with fuel of the highest anticipated enrichment in place assuming optimum moderation. Credit may be taken for neutrons absorbing materials. An independent criticality analysis will not be performed when the design of the storage racks and physical characteristics of the fuel (e.g., enrichment, rod size, number of rods, spacing, and shims) is the same, or is demonstrated in the SAR to be less reactive than those of similar facilities which have been licensed.
   b. The design is such that a fuel assembly cannot be inserted anywhere in the racks other than in the design locations and provisions for drainage are made in the vault design.
   c. Failures of systems or structures not designed to seismic Category I standards and located in the vicinity of the new fuel storage facility will not cause a decrease in the degree of subcriticality provided. Reference to the SAR description section and the general arrangement and layout drawings will be necessary, as well as the tabulation of seismic design classifications for structures and systems. A statement in the SAR establishing the above condition as a design criterion is acceptable.
   d. Design calculations should show that the storage racks and the anchorages can withstand the maximum uplift forces available from the crane without an increase in k_{eff}. A statement in the SAR that excessive forces cannot be applied due to the

design of the crane handling system is acceptable if justification is presented. The evaluation procedures identified in Standard Review Plan 9.1.4 are used to validate this statement.

e. The vault and racks have been designed to preclude damage from dropped heavy objects.

f. Sharing of a storage facility in multi-unit plants does not result in any added potential for increasing the k_{eff} of the storage array.

3. The reviewer verifies that the safety function of the facility will be maintained, as required, if the facility is subjected to natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. In making this determination, the reviewer considers the following points:

   a. The facility design basis and criteria, and the component classification tables presented in the SAR are reviewed to verify that the new fuel storage facility, including the storage vault and racks, have been classified and will be designed to seismic Category I requirements.

   b. The essential portions of the new fuel racks and storage vault are reviewed to verify that protection from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles is provided. Flood protection and missile protection criteria are discussed in the standard review plans for Chapter 3 of the SAR. The reviewer utilizes the procedures of those review plans, as appropriate, to assure that the analyses presented are valid. A statement to the effect that the storage will be located in a seismic Category I structure that is designed to withstand the effects of tornado missiles and floods or that components of the system will be located in individual rooms that will withstand the effects of both flooding and missiles is an acceptable commitment at the CP stage.

4. The evaluations of the new fuel storage facility that are carried out by the secondary review branches are done according to the procedures and criteria in standard review plans for their areas of responsibility.

IV. EVALUATION FINDINGS

The reviewer verifies that the information provided and his review support, conclusions of the following type, to be included in the staff's safety evaluation report:

"The new fuel storage facility includes the fuel assembly storage racks, the concrete storage vault that contains the storage racks, and auxiliary components. The scope of review of the new fuel storage facility for the ________________plant, includes layout drawings, piping and instrumentation diagrams, and descriptive information for the facility and the supporting systems that are essential to the safe operation of the facility. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the new fuel storage facility regarding the provisions necessary to maintain a subcritical array during normal, abnormal, and accident conditions. (CP)] [The review has determined that the applicant's analysis of the design of the new fuel storage facility and supporting systems is in conformance with the proposed design criteria and design bases. (OL)]

"The basis for acceptance in the review has been conformance of the applicant's designs, design criteria, and design bases for the new fuel storage facility and its 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 new fuel storage facility 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 Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 3, "Fire Protection."

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 61, "Fuel Storage and Handling and Radioactivity Control."

6. 10 CFR Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

7. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel Waste and Storage."

8. Regulatory Guide 1.29, "Seismic Design Classification."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.1.2 SPENT FUEL STORAGE

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Reactor Systems Branch (RSB)
Core Performance Branch (CPB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions and to provide a safe means for the confinement and cask loading of the assemblies.

The APCSB reviews the spent fuel storage facility design including the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits. The cooling system is reviewed independently.

1. The facility and components are reviewed with respect to the following:
   a. The quantity of fuel to be stored.
   b. The design and arrangement of the storage racks for maintaining a subcritical array during all conditions.
   c. The degree of subcriticality provided along with the analysis and associated assumptions.
   d. The effects of external loads and forces on the spent fuel storage racks and pool (e.g., safe shutdown earthquake, crane uplift forces, missiles, and dropped objects).
   e. Design codes, materials compatibility, and shielding requirements;.
2. The provisions to preclude dropping the spent fuel shipping cask into the pool are reviewed separately in conjunction with the review of the cask loading pit area.

3. The APCSB review of the provisions for maintaining the pool level and cooling is discussed in conjunction with the spent fuel cooling system review.

4. The applicant's proposed technical specifications are reviewed at the operating license (OL) stage, as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the facility. The secondary reviews are as follows: the SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of structures housing the facility to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The RSB determines that the assigned seismic and quality group classifications for the system components are acceptable. The MTEB verifies, upon request, the compatability of the materials of construction with service conditions. The CPB verifies, upon request, that the k_{eff} of loaded storage racks is acceptable. The RAB reviews the adequacy of the shielding design and the radiation monitoring system.

II. ACCEPTANCE CRITERIA

Acceptability of the spent fuel storage facility design as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR is based on specific general design criteria and regulatory guides, and on independent calculations and staff judgments with respect to system functions and component selection. Listed below are specific criteria related to the storage facility.

1. The design of the spent fuel storage facility is acceptable if the integrated design is in accordance with the following criteria:

   a. General Design Criterion 2, as related to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR;

   b. General Design Criterion 3, as related to protection against fire hazards.

   c. General Design Criterion 4, as related to structures housing the facility and the facility itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, such that safety functions will not be precluded.

   d. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

e. General Design Criterion 61, as related to the facility design for fuel storage and handling of radioactive materials, including the following elements:
   (1) The capability for periodic testing of components important to safety.
   (2) Provisions for containment or confinement.
   (3) The capability to prevent reduction in fuel storage coolant inventory under accident conditions.

f. General Design Criterion 62, as related to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations.

g. General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

h. Regulatory Guide 1.13, as it relates to the fuel handling and storage facility design to prevent damage resulting from the SSE, to prevent loss of water from the fuel pool that could uncover the fuel, and to protect the fuel from mechanical damage.

i. Regulatory Guide 1.29, as related to the seismic design classification of facility components.

j. Fuel storage capacity and criticality limits as discussed in III.1 and III.2 below.

An additional basis for determining the acceptability of the spent fuel storage facility is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

III. <u>REVIEW PROCEDURES</u>

The procedures below are used during the construction permit (CP) application review to determine that the design criteria and bases and the preliminary design meet the acceptance criteria given in Section II of this plan. For the review of the operating license (OL) application, the review procedures and acceptance criteria will be utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design. The OL review includes verification that the content and intent of the technical specifications prepared by the applicant are in agreement with requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

The review procedures given below are for a typical storage system. Any variance of the review, to take account of a proposed unique design, will be such as to assure that the facility design conforms to the criteria in Section II. The reviewer selects and emphasizes material from this review plan, as may be appropriate for a particular case.

1. The quantity of spent fuel to be stored onsite forms the basis for the design capacity of the fuel pool and the number of storage racks provided. The SAR is reviewed to determine that the design basis and facility description section has stated the storage capacity provided by the design. The SARs for recent light water reactor applications have stated that the storage space provided is consistent with the maximum number of spent fuel assemblies unloaded from the core during the refueling cycle plus the fuel contained in a full core load (e.g., 1-1/3 core for a single unit plant and 1-2/3 core for a dual unit facility).

2. The information provided in the SAR pertaining to criticality safety of the spent fuel storage facility is evaluated, based in part on previously approved facilities or on independent calculations by CPB upon request. The facility design criteria, safety evaluation, system description and the layout drawings for the spent fuel pool and storage racks are reviewed to verify that:

    a. Criticality information (including the associated assumptions and input parameters) in the SAR must show that the center-to-center spacing between fuel assemblies in the storage racks is sufficient to maintain the array, when fully loaded and flooded with nonborated water, in a subcritical condition. A k_{eff} of less than about 0.95 for this condition is acceptable. An independent criticality analysis will not be performed when the design of the storage racks and physical characteristics of the fuel (e.g., enrichment, rod size, number of rods, spacing, and shims) is the same or is demonstrated in the SAR to be less reactive than those of similar facilities which have been licensed.

    b. The design of the storage racks is such that a fuel assembly cannot be inserted anywhere other than in a design location.

    c. Failures of systems or structures not designed to seismic Category I standards and located in the vicinity of the spent fuel storage facility will not cause a decrease in the degree of subcriticality provided. Reference to the SAR description section and the general arrangement and layout drawings will be necessary, as well as the tabulation of seismic design classifications for structures and systems. A statement in the SAR establishing the above condition as a design criterion is acceptable. (CP)

    d. Design calculations should show that the storage racks and the anchorages can withstand the maximum uplift forces available from the crane without an increase in k_{eff} or a decrease in pool water inventory. A statement in the SAR that excessive forces cannot be applied due to the design of the crane handling system is acceptable if justification is presented. The evaluation procedures identified in Standard Review Plan 9.1.4 are used to validate this statement.

    e. The spent fuel storage pool and racks are designed to preclude damage from dropped heavy objects.

f. Sharing of storage facilities in multi-unit plants will not increase the potential for the loss of pool water or decrease the degree of subcriticality provided.

3. The reviewer verifies that the safety function of the facility will be maintained, as required, if the facility is subjected to adverse natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. In making this determination, the reviewer considers the following points:

   a. The facility design basis and criteria and the component classification tables are reviewed to verify that the spent fuel storage facility including the storage pool and racks have been classified and designed to seismic Category I requirements. The APCSB will accept a statement that the facility will be designed and constructed as a seismic Category I system. (CP)

   b. The essential portions of the spent fuel storage system are reviewed to verify that protection from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles is provided. Flood protection and missile protection criteria are discussed in the standard review plans for Chapter 3 of the SAR. The reviewer utilizes the procedures of those review plans, as appropriate, to assure that the analyses presented are valid. APCSB will accept a statement to the effect that the facility is located in a seismic Category I structure that is tornado missile and flood protected or that components of the system will be located in individual rooms that will withstand the effects of both flooding and missiles.

4. The wet storage of spent fuel assemblies for safe handling also necessitates the underwater transfer of spent fuel to a loading area for shipment in spent fuel casks. The SAR is reviewed to verify that the design basis and facility description section has stated that a separate spent fuel shipping cask loading area (pit) has been provided adjacent to the spent fuel pool. The loading pit, by virtue of its proximity to the spent fuel pool, is subjected to the same adverse environmental phenomena. Accordingly, the reviewer verifies that the loading pit has been designed so that the safety function of the integrated system will be maintained during these environmental conditions. In addition, the reviewer verifies that the following are included in the design:

   a. An interconnecting canal between the fuel pool and the loading pit should be provided to permit the underwater transfer of fuel to the shipping cask, with provisions for isolating from the fuel pool. A statement in the SAR that these elements are included in the design is acceptable. The reviewer uses engineering judgment to assure himself that the means provided meet the intent stated.

   b. The SAR safety evaluations, results of design calculations, and the general arrangement and layout drawings should show that the spent fuel loading pit has been designed to withstand the loads from dropped heavy objects including the shipping cask, and that the loading area is not an integral part of the storage pool floor so that if a dropped object should breach the pit area, the drainage would not lower the fuel pool water to an unacceptable level. The review of cranes and other elements of the fuel handling system to assure that the design of these components minimizes the likelihood of dropping heavy loads is done under Standard Review Plan 9.1.4.

IV. EVALUATION FINDINGS

The reviewer verifies that the information provided and his review support conclusions of the following type, to be included in the staff's safety evaluation report:

"The spent fuel storage facility includes the spent fuel storage racks, the spent fuel storage pool that contains the storage racks, and the associated equipment storage pits. The scope of review of the spent fuel storage facility for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the facility and the auxiliary supporting systems that are essential to the operation of the facility. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the spent fuel storage facility and the provisions necessary to maintain a subcritical array during all normal, abnormal, and accident conditions. (CP)] [The review has determined that the applicant's analysis of the design of the spent fuel storage facility and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the spent fuel storage facility and necessary auxiliary 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 spent fuel storage facility 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 Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 3, "Fire Protection."

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 61, "Fuel Storage and Handling and Radioactivity Control."

6. 10 CFR Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

7. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage."

8. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

9. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

All nuclear reactor plants include a spent fuel pool for the wet storage of spent fuel assemblies. The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant depending upon the individual design. The safety function to be performed by the system in all cases remains the same; that is, the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. Other functions performed by the system, not related to safety, include water cleanup for the spent fuel pool, refueling canal, refueling water storage tank and other equipment storage pools; means for filling and draining the refueling canal and other storage pools; and surface skimming to provide clear water in the storage pool.

The APCSB review of the spent fuel pool cooling and cleanup system covers the system from inlet to and exit from the storage pool and pits, the seismic Category I water source and piping used for fuel pool makeup, the cleanup system filter-demineralizers and the regenerative process to the point of discharge to the radwaste system.

1. The capability of the spent fuel pool cooling and cleanup system to provide adequate cooling to the spent fuel during all operating conditions is reviewed including the following considerations:
   a. The quantity of fuel to be cooled, including the corresponding requirements for continuous cooling during normal, abnormal, and accident conditions.
   b. The ability of the system to maintain pool water levels.
   c. The ability to provide alternate cooling capability and the associated time required for operation.

d. Provisions to provide adequate make-up to the pool.
e. Provisions to preclude loss of function resulting from single active failures or failures of non-safety-related components or systems.
f. The means provided for the detection and isolation of system components that could develop leaks or failures.
g. The instrumentation provided for initiating appropriate safety actions.
h. The ability of the system to maintain uniform pool water temperature conditions and minimize corrosion products, fission products, and impurities in the water.

The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows: The SEB determines the acceptability of the design analyzes, 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 probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that the system is designed in accordance with applicable codes and standards. The RSB determines that the assigned seismic and quality group classifications for the 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 upon request, determines the adequacy of the design, installation, inspection, and testing of all essential electrical components required for proper operation.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the spent fuel pool cooling and cleanup system, as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR is based on specific general design criteria and regulatory guides, and on independent calculations and staff judgments with respect to system functions and component selection. Listed below are specific criteria related to the spent fuel pool cooling and cleanup systems.

1. The design of the spent fuel pool cooling and cleanup system is acceptable if the integrated design is in accordance with the following criteria:
   a. 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.
   b. General Design Criterion 4, with respect to structures housing the systems and the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

c. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

d. General Design Criterion 44, to include:

   (1) The capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions.
   (2) Suitable redundancy of components so that safety functions can be performed assuming a single active failure of a component coincident with the loss of all offsite power.
   (3) The capability to isolate components, systems, or piping, if required, so that the system safety function will not be compromised.

e. General Design Criterion 45, as related to the design provisions to permit periodic inspection of safety-related components and equipment.

f. General Design Criterion 46, as related to the design provisions to permit operational functional testing of safety-related systems or components to assure structural integrity and system leak tightness, operability, and adequate performance of active system components, and the capability of the integrated system to perform required functions during normal, shutdown, and accident situations.

g. General Design Criterion 61, as related to the system design for fuel storage and handling of radioactive materials, including the following elements:

   (1) The capability for periodic testing of components important to safety.
   (2) Provisions for containment.
   (3) Provisions for decay heat removal.

h. The capability to prevent reduction in fuel storage coolant inventory under accident conditions.

i. General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal, to detect excessive radiation levels, and to initiate appropriate safety actions.

j. Regulatory Guide 1.13, as it relates to the system design to prevent damage resulting from the SSE.

k. Regulatory Guide 1.26 as it relates to quality group classification of the system and its components.

l. Regulatory Guide 1.29, as related to the seismic design classification of system components.

m. Branch Technical Position APCSB 3-1, as it relates to breaks in high and moderate energy piping systems outside containment.

An additional basis for determining the acceptability of the spent fuel pool cooling and cleanup system is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

III. REVIEW PROCEDURES

The 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 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 and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The review procedures for OL applications include 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 as a result of the staff's review.

The review procedures given below are for a typical system. Any variance of the review, to take account of a proposed unique design, will be such as to assure that the system meets the criteria of Section II. In the review, the spent fuel pool cooling and cleanup system is evaluated with respect to its capability to perform the necessary safety functions during all conditions, including normal operation and refueling, abnormal storage conditions, and accident conditions.

1. The safety function of the system for refueling and normal operations is identified by reviewing the information provided in the SAR pertaining to the design bases and criteria and the safety evaluation section. The SAR section on the system functional performance requirements is also reviewed to determine that it describes the minimum system heat transfer and system flow requirements for normal plant operation, component operational degradation requirements (i.e., pump leakage, etc.) and describes the procedures that will be followed to detect and correct these conditions should degradation become excessive. The reviewer, using failure modes and effects analyses, determines that the system is capable of sustaining the loss of any active component and evaluates, on the basis of previously approved systems or independent calculations, that the minimum system requirements (cooling load and flow) are met for these failure conditions. The system piping and instrumentation diagrams (P&IDs), layout drawings, and component descriptions are then reviewed for the following points:
   a. Essential portions of the system are correctly identified and are isolable from the nonessential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion and indicate required classification changes. System drawings are also 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. For the typical system, the drawings and description are reviewed to verify that automatically operated isolation valves separate nonessential portions and components from the essential portions.
   b. Heat exchangers, pumps, valves and piping for the cooling portion of the system are designed to quality group and seismic Category I requirements in accordance with applicable criteria, as described in the system design bases and criteria, and the component classification tables. The APCSB will accept a statement that the system will be designed and constructed as a seismic category I system.

c. The stated quantity of fuel to be cooled by the spent fuel cooling system is consistent with the quantity of fuel stored, as stated in Section 9.1.2 of the SAR.

d. For the maximum heat load with normal cooling systems in operation the temperature of the pool should be kept at or below 140°F and the liquid level in the pool is maintained. The associated parameters for the decay heat load of the fuel assemblies, the temperature of the pool water, and the heatup time or rate of pool temperature rise for the stated storage conditions are reviewed on the basis of independent analyses or comparative analyses of pool conditions that have been previously found acceptable.

e. The spent fuel pool and cooling systems have been designed so that in the event of failure of inlets, outlets, piping, or drains, the pool level will not be inadvertently drained below a point approximately 10 feet above the top of the active fuel. Pipes or external lines extending into the pool that are equipped with siphon breakers, check valves, or other devices to prevent drainage are acceptable as a means of implementing this requirement.

f. A seismic Category I makeup system and an appropriate backup method to add coolant to the spent fuel pool are provided. The APCSB evaluates the component seismic classification table to assure that the primary makeup system is designed as a seismic Category I system. The secondary (backup) system need not be a permanently installed system, nor Category I, but must take water from a Category I source. Engineering judgment and comparison with plants of similar design are used to determine that the makeup capacities and the time required to make associated hookups are consistent with heatup times or expected leakage from structural damage.

g. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

2. The review verifies that the system has been designed so that system functions will be maintained, as required, in the event of adverse natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine the following:

   a. The failure of portions of the system, or of other systems not designed to seismic Category I standards systems and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the pool and cooling system, will not preclude essential functions. Reference to SAR Chapter 2, describing site features and the general arrangement and layout drawings, will be necessary as well as to the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR to the effect that the above conditions are met are acceptable. (CP)

   b. The essential portions of the spent fuel pool cooling system are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR.

The reviewer utilizes the procedures identified in these plans to assure that the analyses presented are valid. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate.

3. The system design information and drawings are analyzed to assure that the following features will be incorporated. A statement that these features will be included in the design by some appropriate means is a basis for acceptance. (CP)
   a. A leakage detection system is provided to detect component or system leakage. An adequate means for implementing this requirement is to provide sumps or drains with adequate capacity and appropriate alarms in the immediate area of the system.
   b. Components and headers of the system are designed to provide individual isolation capabilities to assure system function, control system leakage, and allow system maintenance.
   c. Design provisions are made to assure the capability to detect leakage of radioactivity or chemical contamination from one system to another and to preclude long-term corrosion, organic fouling, or the spreading of radioactivity. Radioactivity monitors and conductivity monitors located in the system discharge lines are acceptable means for implementing this requirement.

4. The essential portions of the system must be protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the system, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and the procedures for reviewing this information are given in the corresponding review plans.

5. The SAR descriptive information, P&IDs, layout drawings, and system analyses are reviewed to assure that essential portions of the system will function following design basis accidents, assuming a concurrent single active component failure. The reviewer evaluates failure mode and effects analyses presented in the SAR to assure function of required components, trace the availability of these components on system drawings, and check that minimum system flow, makeup, 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.

6. The spent fuel pool cleanup system and various auxiliary systems are designated as non-safety-related systems and are designed accordingly (non-seismic Category I). These systems are evaluated to assure that their failure cannot affect the functional performance of any safety-related system or component. The relationship and proximity between the non-safety system and safety-related systems or components are determined by reviewing the integrated structure and component layout diagrams. Independent analyses, engineering judgement, and comparisons with previously approved systems

are used to verify that where a non-safety-related system interconnects or interfaces with the cooling system, its failure by any event or malfunction will not preclude adequate functional performance of the cooling system.

7. The cleanup system is also reviewed to assure that it has been designed with the capability to maintain acceptable pool water conditions. The P&IDs and associated information provided in the SAR is reviewed to verify the following:
   a. A means has been provided for mixing to produce a uniform temperature throughout the pool.
   b. The cleanup components have the capacity and capability to remove corrosion products, fission products, and impurities so that water clarity and quality will enable safe operating conditions in the pool.
   c. The capability for processing the refueling canal coolant during refueling operations has been provided.
   d. Provisions to preclude the inadvertent transfer of spent filter and demineralized media to any place other than the radwaste facility have been provided.

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 spent fuel pool cooling and cleanup system includes all components and piping of the system from inlet to and exit from the storage pool and pits, the seismic Category I water source and piping used for fuel pool makeup, the cleanup system filter-deminerlizers and the regenerative process to the point of discharge to the radwaste system. The scope of review of the spent fuel pool cooling and cleanup system for the ____________________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the spent fuel pool cooling and cleanup system regarding the requirements for continuous cooling during normal, abnormal, and accident conditions. (CP)] [The review has determined that the applicant's analysis of the design of the spent fuel pool cooling and cleanup systems and supporting systems is in conformance with the 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 spent fuel pool cooling and cleanup systems and its 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 spent fuel pool cooling and cleanup system 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 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 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

7. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

8. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage."

9. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

10. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants."

11. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

12. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failure in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.1.4 FUEL HANDLING SYSTEM

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)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The APCSB reviews the fuel handling system (FHS) from the receiving of the new fuel through the shipping of the spent fuel from the plant site. The design layout, which shows the functional geometric layout of the handling equipment, including the areas of movement over and around the fixed locations of safety-related facilities during fuel handling, is reviewed to determine that the various handling operations can be performed safely. The main emphasis in the FHS review is on critical load handling in which inadvertent operations or equipment malfunctions, either separately or in combination, could cause a release of radioactivity or prevent safe shutdown of the reactor.

1. The APCSB reviews the transporting, hoisting, and rigging operations in the fuel handling system as to methods, selection of handling equipment, and safety devices.

2. The APCSB reviews the design of the FHS with respect to the following aspects of individual components and the integrated system:

   a. Performance and load handling requirements specified for equipment.

   b. Handling control features.

   c. The methods and equipment for transferring fuel assemblies from the reactor core to the storage location.

   d. The methods and equipment for transferring stored fuel to the spent fuel shipping cask.

   e. Design codes and standards used for the handling and transportation mechanisms.

The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this review plan.

Secondary reviews will be performed by other branches where necessary and as requested by APCSB to complete the overall evaluation of the FHS. The secondary reviews are as follows. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as a safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification testing and operability of components and confirm that the components, piping, and structures are designed in accordance with applicable codes and standards. The RSB will determine that the seismic and quality group classifications for the system components are acceptable. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all essential electrical components (sensing, control, and power). The RAB reviews the design of the fuel handling system and the spent fuel transfer process to determine whether occupational radiation exposures during spent fuel handling will be as low as practicable.

II. ACCEPTANCE CRITERIA

Acceptability of the FHS design, as described in the applicant's safety analysis report (SAR) including related sections of Chapters 2 and 3 of the SAR, is based on specific general design criteria, regulatory guides, and safety standards and engineering codes. An additional basis for determining the acceptability of the FHS will be the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience. Listed below are specific criteria as they relate to the FHS.

The FHS is acceptable if the integrated design of the structural, mechanical, and electrical elements, the manual and automatic operating controls, and the safety devices provide adequate system control for the specific procedures of handling operations, if the redundancy and diversity needed to protect against malfunctions or failures are provided, and if the design conforms to the following criteria:

1. General Design Criterion 2, as related to the ability of structures, equipment, and mechanisms to withstand the effects of natural phenomena such as earthquakes, tornadoes, floods, and hurricanes.

2. General Design Criterion 5, as related to the capability of shared equipment and components important to safety.

3. Regulatory Guide 1.29, as related to the seismic design classification of components.

4. ANSI standards for components, machinery, and subsystems.

5. Engineering society design standards, codes, or industry standard specifications applicable to the selection of components and subsystems.

6. Branch Technical Position APCSB 9-1, as related to overhead handling systems designed to preclude a load drop from a single failure.

7. For the case where a single failure-proof crane has not been provided, the proposed facility design will be acceptable if it can be determined that the consequences of a load drop would not affect the ability of the plant to be shut down or result in the release of significant amounts of radioactive materials.

III. REVIEW PROCEDURES

The fuel handling system provides for handling of fuel assemblies, spent fuel casks, and other critical loads. The general objective of the review is to confirm that the FHS design precludes system malfunctions or failures that would prevent safe shutdown of the reactor or cause a release of radioactivity. There are variations in the designs of proposed handling systems, hence there will be variations in system requirements and the type and number of critical loads to be handled. For the purpose of this review, the FHS is assumed to include one of two crane types:

1. Cranes whose critical loads, if dropped while being handled, can damage essential equipment or cause a release of radioactivity and are, therefore, designed (including associated rigging and connections to the load) to be "single failure-proof" so that the load could not fall in the event of a single failure.

2. Cranes whose critical loads, if dropped while being handled, cannot damage essential equipment or cause a release of radioactivity because of facility design provisions such as physical separation of essential equipment from load-handling pathways or load limitations.

The procedures listed here are used in the construction permit (CP) review to determine that the FHS design criteria and bases and the preliminary FHS 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 design criteria and bases have been appropriately implemented in the FHS final design.

The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. The system performance requirements for the FHS are reviewed to determine that they cover the handling system concept used in the design, and describe the component and subsystem functions within the integrated system. The performance requirements should also define any degradation considered for components and describe the procedures that are followed to detect and correct degraded conditions.

2. The performance specifications required as part of the design and described in the SAR are reviewed to determine that the design, material selection, manufacturing, installation, testing, and operating procedures are in accordance with state-of-the-art practice. The reviewer verifies that the consensus standards, engineering codes, and industrial or manufacturing association standards selected and used are adequate and appropriate for the FHS.

3. For cranes of "type 2", as defined above, the information presented in the SAR is reviewed to determine that the specific arrangement of the system and subsystems and the load handling paths to be used are described with respect to locations of essential equipment. For overhead cranes and other lifting devices with load limitations or that are separated from essential equipment, the reviewer covers the following points:

   a. The size, shape, and dimensions of the potentially most damaging load (the load which, if dropped by the crane, will cause the most damage), its weight and center of gravity, lifting points, stability, and handling speeds, are compared with the performance specifications to determine the compatibility of the design with load handling and movement requirements. The reviewer uses the requirements of codes and standards and, if required, performs an independent analysis to determine acceptability of the system.

   b. The instrumentation and control system, including the limit and safety devices provided for automatic and manual operation for both normal and emergency conditions, that are required to operate to maintain safety in the event of a failure of the system are reviewed. The results of failure modes and effects analyses are used by the reviewer to determine that the control system adequately limits loads or limits crane load movement, assuming a single failure, without affecting the function of essential equipment or causing the release of radioactivity.

   c. The description of operating and test procedures presented in the SAR is reviewed to determine that load proof-testing, design-rated load testing, nondestructive testing, preventative checks, and examinations of hookup are in accordance with the requirements of the safety standards set forth in ANSI standards.

4. For cranes that have been designed to be single failure-proof, i.e., cranes of "type 1," as defined above, the reviewer determines that the design conforms to Branch Technical Position APCSB 9-1.

5. The information presented in the SAR for the fuel handling equipment, including the equipment storage areas, is reviewed to determine that a seismic event cannot result in damage to spent fuel or essential equipment.

6. The fuel transfer carriage design is reviewed to determine the means of preventing damage to fuel assemblies due to movement of the carriage when the "upender" is in the vertical position.

7. The review for OL applications includes a determination that the content and intent of the technical specifications are in agreement with the requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

IV. EVALUATION FINDINGS

The reviewer verifies that the information provided and his review support conclusions of the following type, to be included in the staff's safety evaluation report:

"The fuel handling system includes all components and equipment used in moving fuel from the receiving of new fuel to the shipping of spent fuel from the plant site. The scope of review of the fuel handling system (FHS) for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to the safe operation of the FHS. [The review has included the applicant's proposed design criteria and design bases for the FHS, the adequacy of those criteria and bases, and the requirements for safe operation of the FHS during normal and abnormal conditions. (CP)] [The review has included the applicant's analysis of the manner in which the design of the FHS and supporting systems conforms to the 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 FHS 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 FHS 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 Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems and Components."

3. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

4. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

5. Branch Technical Position, APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," attached to this plan.

BRANCH TECHNICAL POSITION APCSB 9-1

OVERHEAD HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS

A. BACKGROUND

Overhead handling systems are used for handling heavy items at nuclear power plants. The handling of heavy loads such as a spent fuel cask raises the possibility of damage to the load and to safety-related equipment or structures under and adjacent to the path on which it is transported should the handling system suffer a breakdown or malfunction.

Two methods are used in nuclear power plants to prevent damage to safety features or release of radioactive material due to dropping of heavy loads, such as a spent fuel cask. One is protection by physical design of the facility to preclude damage to spent fuel and safety-related systems if a heavy load should be dropped. The other is to provide an overhead handling system that is designed so that a connected load would not fall in the event of a failure or malfunction.

An overhead handling system includes all the structural, mechanical, and electrical components that are needed to lift and transfer a load from one location to another. Primary load-bearing components, equipment, and subsystems such as the driving equipment, drum, rope reeving, control, and braking systems require special attention. Proper support of the rope drums ensures that they would be retained and prevented from failing or disengaging from the braking and control system in case of a shaft or bearing failure. If the hoisting system (raising and lowering) includes two mechanical holding brakes, each with better than full-load stopping capacity, that are automatically activated when electric power is off or when mechanically tripped by overspeed or overload devices, a critical load will be safely held or controlled in case of failure in the individual load-bearing parts of the hoisting machinery. Failure of the bridge or trolley travel to stop when power is shut off or an overspeed or overload condition due to malfunction or failure in the drive system can be prevented and controlled by appropriate safety and limit devices and brake systems.

Since the crane industry has not yet developed codes or standards that adequately cover the design, operation, and testing for a "single failure-proof" crane, the APCSB has developed a branch position to provide a consistent basis for reviewing equipment and components for such overhead handling systems. The position below delineates acceptable codes and standards and supplements them with specific recommendations on features that will prevent, control, or stop inadvertent operation or malfunction of the mechanical supporting and moving components of the handling system.

B. BRANCH TECHNICAL POSITION

Overhead handling systems intended to provide single failure-proof handling of loads should be designed so that no single failure or malfunction will result in dropping or loosing control of the heaviest (critical) loads to be handled. Such handling systems should be designed, fabricated, installed, inspected, tested, and operated in accordance with the following:

1. General Performance Specifications

a. Separate performance specifications should be prepared for a permanent crane which is to be used for construction prior to use for plant operation. The allowable design stress limits should be identical for both cases, and the sum total of simultaneously applied loads should not result in stress levels causing any permanent deformation other than that due to localized stress concentrations.

b. The operating environment, including maximum and minimum pressure, temperature, humidity, and rates of change of these parameters, should be specified to determine the venting and drainage required for box girder sections. The specifications should also state the corrosive and hazardous conditions that may occur during operation. Fracture toughness for the steel structural materials should be considered. Plate thickness, with a margin for the lowest operating temperatures, should determine the type of steel that can be used with or without toughness tests. The selection of steel materials will be reviewed on a case-by-case basis.

c. The crane should be classified as seismic Category I and should be capable of retaining the maximum design load during a safe shutdown earthquake, although the crane may not be operable after the seismic event. The bridge and trolley should be provided with means for preventing them from leaving their runways with or without the design load during operation or under seismic loadings. The design rated load plus operational and seismically-induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

d. All weld joints for load-bearing structures, including those susceptible to lamellar tearing, should be inspected by nondestructive examinations for soundness of the base metal and weld metal.

e. A fatigue analysis should be considered for critical load-bearing structures and components of the crane handling system. The cumulative fatigue usage factors should reflect effects of cyclic loadings from both the construction and operating periods.

f. Preheat and postheat treatment temperatures for all weldments should be specified in the weld procedures. For low-alloy steel, the recommendations of Regulatory Guide 1.50 should be followed.

2. Safety Features

a. The automatic and manual controls and devices required for normal crane operation should be designed such that a malfunction of these controls and devices, and possible subsequent effects during load handling, will not prevent the handling system from being maintained at a safe neutral holding position.

b. Auxiliary systems, dual components, or ancillary systems should be provided such that in case of subsystem or component failure the load will be retained and held in a safe position.

c. Means should be provided for devices which can be used in repairing, adjusting, or replacing failed components or subsystems when failure of an active component or subsystem has occurred and the load is supported and retained in the safe (temporary) position with the system immobile. As an alternative to repairing the crane in place, means may be provided for moving the handling system with load to a laydown area that has been designed for accepting the load and making the repairs.

3. Equipment Selection

a. Dual load attaching points should be provided on the load block or lifting device, designed so that each attaching point will be able to support a static load of 3W (W is weight of the design rated load), without permanent deformation other than that due to localized stress concentrations in areas for which additional material has been provided for wear.

b. Lifting devices such as lifting beams, yokes, laddle or trunnion type hooks, slings, toggles, or clevises should be of redundant design with dual or auxiliary devices or combinations thereof. Each device should be designed to support a static load of 3W without permanent deformation.

c. The vertical hoisting (raising and lowering) mechanism which uses rope and consists of upper sheaves (head block), lower sheaves (load block), and rope reeving system, should be designed with redundant means for hoisting. Maximum hoisting speed should be no greater than 5 fpm.

d. The head and load blocks should be designed to maintain a vertical load balance about the center of lift from the load block through the head block, and should have a dual reeving system. The load block should maintain alignment and a position of stability with either system and be able to support 3W and maintain load stability and vertical alignment from the center of the head block through all hoisting components to the center of gravity of the load.

e. The design of the rope reeving system should be dual, with each system providing separately the load balance on the head and load blocks through the configuration of ropes and rope equalizers. Selection of the hoisting rope or running rope should consider the size, construction, lay, and means or type of lubrication to maintain efficient working of the individual wire strands as the rope passes over sheaves during the hoisting operation. The effects of impact loadings, acceleration, and emergency stops should be included in selection of the rope and reeving system. The wire rope should be 6 x 37 Iron Wire Rope Core (IWRC) or comparable classification.

The stress in the lead line to the drum during hoisting at the maximum design speed with the design rated load should not exceed 20% of the manufacturer's rated strength of the rope. The static stress in rope (load is stationary) should not exceed 12-1/2% of the manufacturer's rated strength. Line speed during hoisting (raising or lowering) should not exceed 50 fpm.

f. The maximum fleet angle from drum to lead sheave in the load block should not exceed 3-1/2 degrees at any point during hoisting and there should be only one 180° reverse bend for each rope leaving the drum and reversing on the first or lead sheave on the load block, with no other reverse bends other than at the equalizer if a sheave-type equalizer is used. The fleet angles for rope between individual sheaves should not exceed 1-1/2 degrees. Equalizers may be beam or sheave type. For the recommended 6 x 37 IWRC classification wire rope, pitch diameter of the lead sheave should be 30 times rope diameter for the 180° reverse bend, 26 times rope diameter for running sheaves, and 13 times rope diameter for equalizers. The pitch diameter is measured from the center of the rope in the sheave groove through the sheave center. The dual reeving system may be a single rope from each end of a drum terminating at a beam-type load and rope stretch equalizer with each rope designed for total load, or a 2-rope system may be used from each drum or separate drums with a sheave or beam equalizer, or any other combination which provides two separate and complete reeving systems.

g. The vertical hoisting system components, which include the head block, rope reeving system, load block, and dual load attaching device, should each be designed to sustain a load of 2W (W is the weight of the design rated load). A 2W static load test should be performed for each reeving system and load attaching point at the manufacturer's plant. Each reeving system and each one of the load attaching devices should be assembled with approximately a 6 inch clearance between head and load blocks and should support 200% of the design rated load without degradation of the components or permanent deformation other than that due to localized stress concentrations. Measurements of the geometric configuration of the attaching points should be made before and after test followed by nondestructive examination, which should consist of combinations of magnetic particle, ultrasonic, radiographic, and dye penetrant examinations to verify the soundness of fabrication and assure the integrity of this portion of the hoisting system. The results of examinations should be documented and recorded for the hoisting system for each overhead crane.

h. Means should be provided to sense such items as electric current, temperature, overspeed, overloading, and overtravel. Controls should be provided to stop the hoisting movement within 3 inches maximum of vertical travel through a combination of electrical power controls and mechanical braking and torque control systems should one rope of the dual reeving system fail.

i. The control systems may be designed as combination electrical and mechanical systems and may include such items as contractors, relays, resistors, and thyristors in combination with mechanical devices and mechanical braking systems. The electric controls should be selected to provide a maximum breakdown torque limit of 175% of the required rating for a-c motors or d-c motors (series or shunt wound) used for the hoisting drive motors. Compound wound d-c motors should not be used. The control systems provided should consider hoisting (raising and lowering) of all loads, including the design rated load, and the effects of inertia of the rotating hoisting machinery such as motor armatures, shafts and couplings, gear reducers, and drums.

j. The mechanical and structural components of the hoisting system should have the required strength to resist failure should "two-blocking"[1/] or "load hangup"[2/] occur during hoisting. The designer should provide means to absorb or control the kinetic energy of rotating machinery in the event of two-blocking or load hangup. The location and type of mechanical brakes and controls should provide positive and reliable means to stop and hold the hoisting drums for these occurrences. The hoisting system should be able to withstand the maximum torque of the driving motor, if a malfunction occurs and power to the driving motor cannot be shut off at the time of load hangup or two-blocking.

k. The load hoisting drum on the trolley should be provided with structural and mechanical safety devices to prevent the drum from dropping, disengaging from its holding brake system, or rotating, should the drum or any portion of its shaft or bearings fail.

l. To preclude excessive breakdown torque, the horsepower rating (HP) of the electrical motor drive for hoisting should provide no more than 110% of the calculated HP requirement to hoist the design rated load at the maximum design hoist speed.

m. The minimum hoist braking system should include one power control braking system (not mechanical or drag brake-type) and two mechanical holding brakes. The holding brakes should be activated when power is off and should be automatically tripped by mechanical means on overspeed to the full holding position if a malfunction occurs in the electrical brake controls. Each holding brake should be designed to 125% - 150% of maximum developed torque at the point of application (location of the brake in the mechanical drive). The minimum design requirements for braking

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[1/] "Two-blocking" is an inadvertently continued hoist which brings the load and head block assemblies into physical contact, thereby preventing further movement of the load block and creating shock loads to rope and reeving system.

[2/] "Load hangup" occurs when the load block or load is stopped during hoisting by entanglement with fixed objects, thereby overloading the hoisting system.

systems that will be operable for emergency lowering after a single brake failure should be two holding brakes for stopping and controlling drum rotation. Provisions should be made for manual operation of the holding brakes. Emergency brakes or holding brakes which are to be used for manual lowering should be capable of operation with full load and at full travel and provide adequate heat dissipation. Design for manual brake operation during emergency lowering should include features to limit the lowering speed to less than 3.5 fpm.

n. The dynamic and static alignment of all hoisting machinery components including gearing, shafting, couplings, and bearings should be maintained throughout the range of loads to be lifted with all components positioned and anchored on the trolley machinery platform.

o. Increment drives for hoisting may be provided by stepless controls or inching motor drives. Plugging[3/] should not be permitted. Controls to prevent plugging should be included in the electrical circuits and the control system. Floating point[4/] in the electrical power system, when required for bridge or trolley movement, should be provided only for the lowest operating speeds.

p. To avoid the possibility of overtorque within the control system, the horsepower rating of the driving motor and gear reducer for trolley and bridge motion of an overhead bridge crane should not exceed 110% of the calculated requirement at maximum speed and with the design rated load. Incremental or fractional inch movements, when required, should be provided by such items as variable speed or inching motor drives. Control and holding brakes should each be rated at 100% of maximum drive torque at the point of application. If two mechanical brakes are provided, one for control and one for holding, they should be adjusted with one brake in each system for both the trolley and bridge leading the other and should be activated by release or shutoff of power. The brakes should also be mechanically tripped to the "on" or "holding" position in the event of a malfunction in the power supply or an overspeed condition. Provisions should be made for manual operation of the brakes. The holding brake should be designed so that it cannot be used as a foot-operated slowdown brake. Drag brakes should not be used. Opposite wheels on bridges or trolleys which support the bridge or trolley on the runways should be matched and have identical diameters. Trolley and bridge speeds should be limited. A maximum speed of 30 fpm for the trolley and 40 fpm for the bridge is recommended.

q. The complete operating control system and provisions for emergency controls for the overhead crane handling system should be located in the main cab on the

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[3/] Plugging is the momentary application of full line power to the drive motor for the purpose of promoting a limited movement.

[4/] The point in the lowest range of movement control at which power is on, brakes are off, and motors are not energized.

bridge. Additional cabs located on the trolley or lifting devices should have complete control systems similar to the bridge cab. Manual controls for hoisting and trolley movement may be provided on the trolley. Manual controls for the bridge may be located on the bridge. Remote controls or pendant controls for any of these motions should be the same as those provided in the bridge cab control panel. Provisions should be made in the design for devices for emergency control or operations. Limiting devices, mechanical and electrical, should be provided to indicate, control, and prevent overtravel and overspeed of hoist (raising or lowering) and for trolley and bridge travel movements. Buffers for bridge and trolley travel should be included.

r. Safety devices such as limit type switches provided for malfunction, inadvertent operation, or failure should be in addition to and separate from the control devices provided for operation.

s. The operating requirements for all travel movements (vertical and horizontal movements or rotation, singly or in combination) for permanent plant cranes should be clearly defined in the operating manual for hoisting and for trolley and bridge travel. The designer should establish the maximum working load (MWL). The MWL should not be less than 85% of the design rated load (DRL) capacity for the new crane at time of operation. The redundancy provided, design factors, selection of components, and balance of auxiliary-ancilliary and duel items in the design and manufacture should be taken into account in setting the maximum working load for the critical load handling crane system(s). The MWL should not exceed the DRL for overhead crane handling systems.

t. When the permanent plant crane is to be used for construction and the operating requirements for construction are not identical to those required for permanent plant service, the construction operating requirements should be defined separately. The crane should be designed structurally and mechanically for the construction loads, plant service loads, and the functional performance requirements for each. At the end of the construction period, the crane handling system should be adjusted for the performance requirements of permanent plant service. The conversion or adjustment may include the replacement of such items as motor drives, blocks, and reeving system. After construction use, the crane should be thoroughly inspected using nondestructive examinations and should be performance tested. If the load and performance requirements are different for construction and plant service periods, then the crane should be tested for both phases. The crane integrity should be verified by the designer and manufacturer and load testing to 125% of the design rated load required for the operating plant should be done before the crane is used as permanent plant equipment.

u. Installation instructions should be provided by the manufacturer. These should include a full explanation of the crane handling system, its controls, and the limitations for the system, and should cover the requirements for installation, testing, and preparations for operation.

4. <u>Mechanical Checks, Testing, and Preventive Maintenance</u>

a. A complete mechanical check of all crane systems as installed should be made to verify the method of installation and to prepare the crane for testing. During and after installation the proper assembly of electrical and structural components should be verified. The integrity of all control, operating, and safety systems is to be verified as to satisfaction of installation and design requirements.

The crane designer and crane manufacturer should provide a manual of information and procedures for use in checking, testing, and crane operation. The manual should also describe a preventive maintenance program based on the approved test results and information obtained during the testing; it should include such items as servicing, repair, and replacement requirements, visual examinations, inspections, checking, measurements, problem diagnosis, nondestructive examination, crane performance testing, and special instructions.

Information concerning proof testing on components and subsystems as required and performed at the manufacturer's plant to verify component or subsystem ability to perform should be available for the checking and testing performed at the place of installation of the crane system.

b. The crane system should be prepared for the static test of 125% of the design rated load. The tests should include all positions of hoisting, lowering, and trolley and bridge travel with the 125% rated load and other positions as recommended by the designer and manufacturer. After satisfactory completion of the 125% static test and adjustments required as a result of the test, the crane handling system should be given full performance tests with 100% of the design rated load for all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices. The crane handling system should demonstrate the ability to lower and move the design rated load by manual operation and with the use of emergency operating controls and devices which have been included in the handling system.

The complete hoisting machinery should be allowed to two-block during the hoisting test (load block limit and safety devices are bypassed). This test should be conducted without load and at slow speed, to provide assurance of the integrity of the design, equipment, controls, and overload protection devices. The test should demonstrate that the maximum torque that can be developed by the driving system, including the inertia of the rotating parts at the overtorque condition, will be absorbed or controlled prior to two-blocking.

The complete hoisting machinery should be tested for ability to sustain a load hangup condition by a test in which the load block attaching points are secured to a fixed anchor or excessive load. The drum should be capable of one full revolution before starting the hoisting test.

c. The preventive maintenance program recommended by the designer and manufacturer should also prescribe and establish the MWL for which the crane will be used. The maximum working load should be plainly marked on each side of the crane for each hoisting unit. It is recommended that critical load handling cranes should be continuously maintained at 95% of DRL capacity for the MWL capacity.

C. REFERENCES

1. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

2. "Table of Engineering, Manufacturing, and Operating Standards, Practices, and References," attached to this position.

TABLE OF
ENGINEERING, MANUFACTURING, AND OPERATING STANDARDS,
PRACTICES, AND REFERENCES

AISE	Association of Iron and Steel Engineers (Std. No. 6). General items for overhead cranes and specifically for drums, reeving systems, blocks, controls, and electrical, mechanical, and structural components.
AISC	American Institute of Steel Construction, "Manual of Steel Construction." Runway and bridge design loadings for impact, and structural supports.
ASME	American Society of Mechanical Engineers. References for testing, materials, and mechanical components.
ASTM	American Society for Testing Materials. Testing and selection of materials.
ANSI	American National Standards Institute (A10, B3, B6, B15, B29, B30 and N45 series). N series of ANSI standards for quality control. ANSI consensus standards for design, manufacturing, and safety.
IEEE	Institute of Electrical and Electronics Engineers. Electrical power and control systems.
AWS	American Welding Society (D1.1.72 - 73/74 revisions). Fabrication requirements and standards for crane structure and weldments.
EEI	Edison Electrical Institute. Electrical systems.
SAE	Society of Automotive Engineers, "Standards and Recommended Practices." Recommendations and practices for wire rope, shafting, lubrication, fasteners, materials selection, and load stability.
CMAA	Crane Manufacturers Association of America (CMAA 70). Guide for preparing functional and performance specifications and component selection.
NEMA	National Electrical Manufacturers Association. Electrical motor, control, and component selections.
WRTB	Wire Rope Technical Board and their manufacturing members. Selection of rope reeving system, and reeving efficiencies.
MHI	Materials Handling Institute and their member associations and association members such as American Gear Manufacturing Association for gears and gear reducers and Antifriction Bearing Manufacturers Association for bearings selection.
WRC	Welding Research Council, "Control of Steel Construction to avoid Brittle Fracture," and Bulletin #168, "Lamellar Tearing."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.2.1 STATION SERVICE WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The APCSB reviews the system from the service water pump intake to the points of cooling water discharge. The ultimate heat sink (reviewed under Standard Review Plan (9.2.5) is the intake source of water provided to the SWS for longterm cooling of station features required for plant shutdown and also any special equipment required to prevent or mitigate the consequences of postulated accidents and as such is an interface system to the SWS. The SWS pump performance characteristics will be compared to the high and low water levels of the ultimate heat sink to assure that pumping capability can be provided for extended periods of operation following postulated events.

1. The APCSB reviews the characteristics of the SWS components (pumps, heat exchangers, pipes, valves) with respect to their functional performance as affected by adverse environmental occurrences, by abnormal operational requirements, and accident conditions such as a loss-of-coolant accident (LOCA) and the loss of offsite power. Since the SWS normally has requirements that relate to cooling functions during normal plant operation as well as for safety functions, the review will include an evaluation of the capability of the system to perform these multiple functions.

2. The APCSB reviews the system to determine that a malfunction, a failure of a component, or the loss of a cooling source will not reduce the safety-related functional performance capabilities of the system. Specifically, the system is reviewed to verify that:

a. System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from missiles, pipe whip, and jet impingement that may result from piping cracks or breaks.
b. Design code requirements, as applicable to the assigned quality group and seismic category, are met.
c. Effects of failure of the non-seismic Category I equipment, structure, or components on safety-related portions of the SWS system are taken into account in the design. In addition, the review includes the consequences of postulated pipe breaks in high and moderate energy fluid systems.

3. The APCSB also reviews the design of the SWS with respect to:
   a. Functional capability during abnormally high water levels; i.e., adequate flood protection during the probable maximum flood.
   b. The capability for detection, control, and isolation of system leakage including the capability for detection and control of radioactive leakage into and out of the system and prevention of accidental releases to the environment.
   c. Measures to preclude long-term corrosion and organic fouling that would tend to degrade system performance.
   d. Provisions for system and component operational testing, including the instrumentation and control features that determine and verify that the system is operating in a correct mode (i.e., valve position, pressure and temperature indication).

4. The APCSB reviews the SWS capability to flood the reactor containment should this be required in a post-accident recovery situation.

5. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows: the RSB identifies essential components associated with the reactor coolant system and the emergency core cooling systems that are required for operation during normal operations and accident conditions. The RSB establishes accident cooling load functional requirements and minimum time intervals and determine that the seismic and quality group classifications for system components are acceptable. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification of components and confirm that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will evaluate the controls, instrumentation, and power sources with respect to capabilities, capacity, and reliability for supplying power during normal and emergency conditions to safety-related pumps, valves and other components.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the service water system, as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR is based on specific general design criteria and regulatory guides. An additional basis for determining the acceptability of the SWS will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. Listed below are specific criteria as they relate to the SWS.

The design of the service water system 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, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety seem capable of performing required safety functions.

4. General Design Criterion 44, to assure:

   a. The capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions.
   b. Component redundancy so that under LOCA conditions the safety function can be performed assuming a single active component failure coincident with the loss of offsite power.
   c. Component redundancy so that the safety function can be performed assuming a single active component failure coincident with the loss of offsite power.
   d. The capability to isolate components, subsystems, or piping if required so that the system safety function will not be compromised.

5. General Design Criterion 45, as related to design provisions made to permit inservice inspection of safety-related components and equipment.

6. General Design Criterion 46, as related to design provisions made to permit operational functional testing of safety-related systems and components to assure:

   a. Structural integrity and system leak tightness.
   b. Operability and adequate performance of active system components.
   c. Capability of the integrated system to perform required functions during normal, shutdown, and accident situations.

7. Regulatory Guide 1.26, as related to the quality group classification of systems and components.

8. Regulatory Guide 1.29, as related to the seismic design classification of system components.

9. Branch Technical Position APCSB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this review plan. For review of operating license (OL) applications, the review procedures and acceptance criteria 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 review procedures for OL applications include 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 as a result of the staff's review.

As a result of the various SWS designs provided, there will be variations in system requirements. For the purpose of this review plan, a typical system is assumed which has fully redundant systems, with each of the systems having an identical essential (safety features) portion and an identical non-essential portion (used for normal operation). For cases where there are variations from the typical arrangement, the reviewer will adjust the review procedures given below. However, the system design will be required to meet the acceptance criteria given in Section II of this review plan. Also, the reviewer will need to refer to review plans for other systems that would interface with the SWS, depending upon the nature and conditions of the ultimate heat sink cooling water (e.g., salt water).

1. The SAR is reviewed to determine that the system description section and piping and instrumentation diagrams (P&IDs) show the SWS equipment that is used for normal operation, and the minimum system heat transfer and flow requirements for normal plant operation. The system performance requirements section will also be reviewed to determine that it describes component allowable operational degradation (e.g., pump leakage) and describes the procedures that will be followed to detect and correct these conditions when they become excessive.

2. The reviewer, using the results of failure modes and effects analyses as appropriate, comparisons with previously approved systems, or independent calculations, determines that the system is capable of sustaining the loss of any active component and meeting minimum system requirements (cooling load and flow) for the failure conditions. The

system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed for the following points:

a. Essential portions of the SWS are correctly identified and are isolable from the non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion and indicate the required classification changes. System drawings are also 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. The drawings and descriptions are reviewed to verify that automatically operated isolation valves separate non-essential portions and components from the essential portions.
b. Essential portions of the SWS, including the isolation valves separating essential and non-essential portions, are classified Quality Group C or higher and seismic Category I. Components and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above seismic and safety classifications have been included, and that the P&IDs indicate any points of change in piping quality group classification.
c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

3. The reviewer determines that the safety function of the system will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. The reviewer uses engineering judgment, the results of a failure mode and effects analyses, and the results of reviews performed under other review plans to verify the following:

a. The failure of 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 that house, support, or are close to essential portions of the SWS, will not preclude operation of the essential portions of the SWS. Reference to SAR Chapter 2 describing site features and the general arrangement and layout drawings will be necessary as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)
b. The essential portions of the SWS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The reviewer will utilize the procedures identified in these review plans to assure that the analyses presented are valid. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the

effects of both flooding and missiles is acceptable. The location and the design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate.

c. The SWS pumps will have sufficient available net positive suction head (NPSH) at the pump suction locations, considering low water levels. Reference to SAR Section 2.4, which indicates the lowest probable water level of the heat sink, and to drawings indicating the elevation of service water pump impellers will be necessary. An independent calculation verifying the applicant's conclusion will be necessary for acceptance.

d. Provisions are made in the system to detect and control leakage of radioactive contamination into and out of the system. It will be acceptable if the system P&IDs show radiation monitors located on the system discharge and at components susceptible to leakage, and components can be isolated by one automatic and one manual valve in series.

e. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the SWS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding review plans.

f. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The system design will be acceptable if the SWS meets minimum system requirements as stated in the SAR assuming a concurrent failure of a single active component, including a single failure of an auxiliary electric power source. The SAR is reviewed to determine that for each SWS component or subsystem affected by the loss of offsite power, system flow and heat transfer capability meet or exceed minimum requirements. The results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This will be an acceptable verification of system functional reliability.

3. The descriptive information, P&IDs, SWS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the failure mode 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 contains verification that minimum system flow and heat transfer requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

IV. EVALUATION FINDINGS

The reviewer determines 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 service water system (SWS) includes all components and piping from the SWS pump intake to the points of cooling water discharge. The scope of review of the service water system for the ________________ plant includes layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the SWS and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the service water system regarding the requirements for continuous cooling during all conditions of plant operation. (CP)] [The review has determined that the applicant's analysis of the design of the service water system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the service water system and necessary auxiliary 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 service water system 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 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 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water Systems."

7. Regulatory Guide 1.26, "Quality Group Classification and Standards For Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants."

8. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

9. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.2.2 REACTOR AUXILIARY COOLING WATER SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)

I. AREA OF REVIEW

The APCSB reviews reactor auxiliary cooling systems that are required for safe shutdown during normal, operational transient, and accident conditions, and for mitigating the consequences of an accident, or preventing the occurrence of an accident. These include closed loop auxiliary cooling systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS).

The review of these systems includes the pumps, heat exchangers, valves and piping, expansion tanks, makeup piping, and points of connection or interfaces with other systems. Emphasis is placed on the cooling systems for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment.

1. The APCSB reviews the capability of the auxiliary cooling systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. The review includes the following points:

   a. The functional performance requirements of the system including the ability to withstand adverse environmental occurrences, operability requirements for normal operation, and requirements for operation during and subsequent to postulated accidents.

   b. Multiple performance functions (if required) assigned to the system and the necessity of each function for emergency core cooling and safe shutdown.

   c. The capability of the system to cope with liquid expansion or provide necessary makeup as required.

d. The requirements for adequate net positive suction head (NPSH) for the auxiliary cooling pumps.

e. The sizing of the system for core cooling and decay heat loads and the associated design margin.

2. The APCSB review verifies that system components and piping have sufficient physical separation or shielding to protect essential portions of the system from missiles and pipe whip or from jet impingement that may result from piping cracks or breaks.

3. Other system aspects that are reviewed are:

a. The use of design and fabrication codes consistent with the assigned quality group classification and seismic category.

b. The effects of non-seismic Category I component failures on the seismic Category I portion of the system.

c. The provisions for detection, collection, and control of system leakage and the means provided to detect leakage of activity from one system to another and preclude its release to the environment.

d. The provisions to control long-term corrosion and organic fouling.

e. The requirements for operational testing and inservice inspection of the system.

f. Instrumentation and control features necessary to accomplish design functions, including isolation of components to deal with leakage or malfunctions, and actuation requirements for redundant equipment.

4. The applicant's proposed technical specifications will be reviewed for operating license applications as they relate to areas covered in this review plan.

The review of the cooling water systems will involve secondary reviews performed by other branches. The results are used by the APCSB to complete overall evaluation of the system. The secondary reviews are as follows: the RSB will identify engineered safety feature components associated with the reactor coolant system and the emergency core cooling systems that are required for operation during normal operations and accident conditions. RSB will establish cooling load functional requirements and minimum time intervals and assure that the seismic and quality group classifications for system components are acceptable. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification of components and confirm that the system is designed in accordance with applicable codes and standards. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all essential electrical components required for proper operation.

II. ACCEPTANCE CRITERIA

Acceptability of the designs of cooling water systems as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR, is based on specific general design criteria and regulatory guides, and on independent calculations and staff judgments with respect to system functions and component selection. Listed below are specific criteria as they relate to the cooling water systems.

The design of a cooling water system 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, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 44, to include:

   a. The capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under both normal operating and accident conditions.

   b. Component redundancy so that safety functions can be performed assuming a single active component failure coincident with the loss of offsite power.

   c. The capability to isolate components, systems, or piping if required so that the system safety function will not be compromised.

5. General Design Criterion 45, as related to the design provisions to permit inservice inspection of safety-related components and equipment.

6. General Design Criterion 46, as related to the design provisions to permit operational functional testing of safety-related systems or components to assure:

   a. Structural integrity and system leak tightness.

b. Operability and adequate performance of active system components.

c. Capability of the integrated system to perform required functions during normal, shutdown, and accident situations.

7. Regulatory Guide 1.26, as related to the quality group classification of systems and components.

8. Regulatory Guide 1.29, as related to the seismic design classification of system components.

9. Branch Technical Position APCSB 3-1, as related to high and moderate energy breaks in piping systems outside containment.

An additional basis for determining the acceptability of a cooling water system will be the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience.

III. REVIEW PROCEDURES

The 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 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 review procedures and acceptance criteria will be 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 procedures for OL reviews include 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 as a result of the staff's review.

One of the main objectives in the review of a cooling water system (CWS) is to determine its function with regard to safety. Some cooling systems are designed as safety-related systems in their entirety, others have only portions of the system that are safety related, and others are classified as non-safety-related because they do not perform any safety function. In order to determine the safety category of a cooling water system, the APCSB will evaluate its necessity for achieving safe reactor shutdown conditions or for accident prevention or accident mitigation functions. The safety functions to be performed by these systems in all designs are essentially the same, however, the method used varies from plant to plant depending upon the individual designer.

In view of the various designs provided, the procedures set forth below are for a typical cooling water system designed entirely as a safety-related system. Any variance of the review procedures to take account of a proposed unique design will be such as to assure that the system meets the criteria of Section II. The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. The information provided in the SAR pertaining to the design bases and design criteria, and the system description section are reviewed to verify that the equipment used and the minimum system heat transfer and flow requirements for normal plant operations are identified. A review of the system piping and instrumentation diagrams (P&IDS) will show which components of the system are utilized to:

   a. Remove heat from the reactor primary coolant system equipment necessary to achieve a safe reactor shutdown.

   b. Provide essential cooling for containment components or systems such as the sprays, ventilation coolers, or sump equipment.

   c. Provide cooling for decay heat removal equipment.

   d. Provide cooling for emergency core cooling pump bearings or other emergency core cooling equipment necessary to prevent or mitigate the consequences of an accident.

2. The system performance requirements section is reviewed to determine that it limits allowable component operational degradation (e.g., pump leakage) and describes the procedures that will be followed to detect and correct these conditions when degradation becomes excessive.

3. The reviewer, using the results of failure modes and effects analyses, determines that the system is capable of sustaining the loss of any active component and, on the basis of previously approved systems or independent calculations, that the minimum system requirements (cooling load and flow) are met for these failure conditions. The system P&IDs layout drawings, and component descriptions and characteristics are then reviewed for the following points:

   a. Essential portions of the CWS are correctly identified and are isolable from the non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion and indicate required classification changes. System drawings are reviewed to see that they show the means for accomplishing isolation and the SAR description is reviewed to identify minimum performance of the isolation valves. The drawings and description are reviewed to verify that automatically operated isolation valves separate non-essential portions and components from the essential portions.

   b. Essential portions of the CWS, including the isolation valves separating seismic Category I portions from the non-seismic portions, are Quality Group C or higher and seismic Category I. System design bases and criteria, and the component classification tables are reviewed to verify that the heat exchangers, pumps, valves and piping of essential portions of the system will be designed to seismic Category I requirements in accordance with the applicable criteria.

c. The system is designed to cope with liquid expansion or to provide water makeup as necessary. Where the cooling water systems are closed loop systems, surge tanks are generally provided to accommodate liquid volume changes due to changes in temperature or leakage and to receive system makeup water as required. The surge tank and connecting piping are reviewed to assure that makeup water can be supplied to either header in a split header system. Redundant surge tanks (one to each header) or a divided surge tank design are acceptable to assure that in the event of a header rupture the loss of the entire contents of the surge tank will not result.

d. Net positive suction head (NPSH) requirements for the cooling water pumps are met during normal operations and accident conditions, including conditions of extreme low water levels. The review of the system design information and the system and station drawings locating the cooling water system in the facility identifies the components and water levels necessary to provide NPSH for the cooling water pump. Independent analyses and engineering judgment are used in conjunction with pump performance curves to assure that the design and the location of the pump and components are such as to maintain appropriate NPSH requirements.

e. The system is designed for removal of heat loads during normal operation and of emergency core cooling heat loads during accident conditions, with appropriate design margins to assure adequate operation. A comparative analysis is made of the system flow rates, heat levels, maximum temperature, and heat removal capabilities with similar designs previously found acceptable. To verify performance characteristics of the system, an independent analysis may be made.

f. Design provisions are made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

g. Essential portions of the system are protected from the effects of high energy and moderate energy line breaks. The system description and layout drawings will be reviewed to assure that no high or moderate energy piping systems are close to essential portions of the CWS or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and the procedures for reviewing this information are given in the corresponding review plans.

h. Essential components and subsystems (i.e., those necessary for safe shutdown) can function as required in the event of a loss of offsite power. The system design will be acceptable in this regard if the essential portions of the CWS meet minimum system requirements as stated in the SAR assuming a concurrent failure of a single active component, including a single failure of any auxiliary electric power source. The SAR is reviewed to determine that for each CWS component or subsystem affected by the loss of offsite power, system flow and heat transfer capability exceed minimum requirements. The results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This will be an acceptable verification of system functional reliability.

3. The system design information and drawings are analyzed to assure that the following features will be incorporated.

   a. A leakage detection system is provided to detect component or system leakage. An adequate means for implementing this criterion is to provide sumps or drains with adequate capacity and appropriate alarms in the immediate area of the system.

   b. Components and headers of the system are designed to provide individual isolation capabilities to assure system function, control system leakage, and allow system maintenance.

   c. Design provisions are made to assure the capability to detect leakage of radioactivity or chemical contamination from one system to another, to preclude long-term corrosion, organic fouling, or the spreading of radioactivity. Radioactivity monitors and conductivity monitors should be located in the system component discharge lines to detect leakage. An alternate means is to prevent leakage from occurring by operating the system at higher pressure to assure that leakage is in the preferred direction.

4. The reviewer verifies that the system has been designed so that system functions will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods. The reviewer evaluates the system using engineering judgment and the results of failure modes and effects analyses to determine the following:

   a. The failure of 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 that house, support, or are close to essential portions of the CWS, will not preclude essential functions. The review will identify these non-seismic category components or piping and assure that appropriate criteria are incorporated to provide isolation capabilities in the event of failure. Reference to SAR Chapter 2, describing site features, and the general arrangement and layout drawings will be necessary as well as to the SAR tabulation of seismic design classifications for structures and systems.

   b. The essential portions of the CWS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The reviewer will utilize the procedures identified in these review plans to assure that the analyses presented are valid. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate.

5. The descriptive information, P&IDs CWS drawings, and failure modes and effects analysis in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the failure mode 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 accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 reactor auxiliary cooling water systems include pumps, heat exchangers, valves and piping, expansion tanks, makeup piping, and the points of connection or interfaces with other systems. The scope of review of the cooling water systems for the ___________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the cooling water systems and the auxiliary supporting systems that are essential to operation of the cooling water systems. [The review has included the applicant's proposed design criteria and design bases for the cooling water systems, the adequacy of those criteria and bases, and the requirements for continuous cooling (if necessary) during all conditions of plant operation. (CP)] [The review has included the applicant's analysis of the manner in which the design of the cooling water systems and auxiliary supporting systems demonstrates conformance to the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the cooling water systems and necessary auxiliary 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 cooling water 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 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 50, Appendix A, General Design Criterion 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants."

8. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

9. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failure in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

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SECTION 9.2.3 DEMINERALIZED WATER MAKEUP SYSTEM

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)
Effluent Treatment Systems Branch (ETSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The APCSB reviews the demineralized water makeup system (DWMS) from the supply connection of the service or municipal water source to the points of discharge. The capability to provide an adequate supply of treated water of reactor coolant purity to other systems as makeup, and other plant demineralized water requirements is reviewed.

1. The APCSB review of the DWMS system includes the following considerations:
   a. Capability of the system to effectively store, handle, and dispense all chemicals utilized in the demineralizing and regeneration process.
   b. Capability of the DWMS to operate within the environment to which it is exposed.
   c. Provisions for the regeneration wastes to be directed to a suitable point in the radwaste system or other specified areas for subsequent processing prior to discharge to the environment and instrumentation and isolation capabilities provided, including the ability to detect corrosive solutions and the valving necessary to isolate the system.

2. The APCSB reviews the system function to determine whether portions of the system are safety-related.

3. The DWMS is also reviewed to assure that a malfunction or failure of a component will not have an adverse effect on any safety-related system or components.

4. The applicant's proposed technical specifications are reviewed at the operating license (OL) stage, as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows.

The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that the components, piping, and structures 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 compatibility of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of all essential electrical components (sensing, control, and power) required for proper operation. The ETSB verifies that the limits for radioactivity concentrations are met.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the DWMS, as described in the applicant's safety analysis report (SAR) is based on the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience. There are no general design criteria or regulatory guides that apply directly to the safety-related functional performance requirements for the DWMS. The APCSB assures that the system is capable of providing the required supply of reactor coolant purity water to all systems.

Several general design criteria and regulatory guides are used to evaluate the system design for those cases when a failure or malfunction of the DWMS could adversely effect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation). These are as follows:

1. General Design Criterion 2, as related to the safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, or floods as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to the system being capable of withstanding the effects of internally generated missiles.

3. General Design Criterion 5, in regards to the effect of sharing in multiple unit facilities.

4. Regulatory Guide 1.29, Position C-1, if any portion of the system is deemed to be safety-related, and Position C-2 for non-safety-related functions.

5. Appendix 1 of Regulatory Guide 1.56, for an acceptable standard for purity of the demineralized water produced by the DWMS.

6. Branch Technical Position APCSB 3-1, as it relates to high and moderate energy breaks or cracks in piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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 applications, the review procedures and acceptance criteria 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 procedures for OL reviews include 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 as a result of the staff's review.

The reviewer selects and emphasizes material from this review plan, as may be appropriate for a particular case. A determination will be made as to whether the DWMS or portions thereof are safety-related. In confirming this design aspect, an analysis is made in which it is assumed that any DWMS pipe fails or component malfunctions or fails in such a manner as to cause maximum damage to other equipment located nearby. The system will be considered non-safety-related if its failure does not affect the ability of the reactor facility to achieve and maintain safe shutdown conditions.

1. The APCSB evaluates the system design information and drawings and, utilizing engineering judgment, operational experience, and performance characteristics of similar, previously approved systems, verifies that:
   a. The system is capable of fulfilling the requirements of the facility for makeup water on a day-to-day basis.
   b. The component redundancy necessary for the system to perform its design function is provided.
   c. Precautions are taken or incorporated into the system design to properly store, handle, and dispense corrosive and toxic chemicals effectively and safely so that a hazardous condition does not result from mishandling or leakage.
   d. The components utilized are compatible with the associated chemicals.
   e. The potential for leakage and accidental spills has been minimized.
   f. In the event of a leak or spill, there would not be an adverse effect on safety-related systems or components.
   g. Instrumentation (e.g., a conductivity monitor) has been provided together with the capability to isolate the system should planned operating conditions be exceeded.
   h. Piping has been provided as necessary to direct solutions and regenerative wastes to the radwaste system or other specified areas for processing and disposal.

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 demineralized water makeup system includes all components and piping associated with the system from the service or numicipal water source to the points of discharge to other systems or to a discharge canal. The scope of review of the demineralized water makeup system for the ____________________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and for the auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the demineralized water makeup system, regarding safety related requirements (if any) for an adequate supply of reactor coolant purity water during all conditions of plant operation. (CP)] [The review has determined that the applicant's analysis of the designs of the demineralized water makeup system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the demineralized water makeup system and necessary auxiliary 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 and is acceptable."

V. REFERENCES

1. General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

2. General Design Criterion 4, "Environmental and Missile Design Bases."

3. General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

5. Regulatory Guide 1.56, Appendix, "Maintenance of Water Purity in Boiling Water Reactors."

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.

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SECTION 9.2.4 POTABLE AND SANITARY WATER SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

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, ETSB review consists of confirming the design accepted at the CP stage.

1. The system descriptions for the potable and sanitary water systems (PSWS) are reviewed. The piping and instrumentation drawings (P&IDs) are reviewed at the OL stage.

2. System design criteria to prevent connection to systems having the potential for containing radioactive material are reviewed.

II. ACCEPTANCE CRITERIA

ETSB accepts the PSWS design if there are no interconnections between the PSWS and systems having the potential for containing radioactive material.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from this review plan, as may be appropriate for a particular case.

In the review of the PSWS, ETSB considers the design criteria to prevent cross connections, as described in the SAR. The P&ID's are reviewed at the OL stage to verify the absence of interconnections with systems having the potential for containing radioactive materials.

IV. EVALUATION FINDINGS

ETSB determines 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 potable and sanitary water systems (PSWS) include all components and piping from the supply connection to the municipal or other water source to all points of discharge to sewage facilities or other plant systems. The scope of review of the PSWS included piping and instrumentation diagrams and descriptive information for the PSWS. The applicant's proposed design criteria bases for the PSWS have been reviewed.

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the potable and sanitary water systems to industry standards. Based on the foregoing evaluation, we conclude that the proposed potable and sanitary water systems are acceptable."

V. REFERENCES

None

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SECTION 9.2.5 ULTIMATE HEAT SINK

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Site Analysis Branch (SAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident, including LOCA.

The APCSB reviews the water sources which make up the ultimate heat sink. This includes the size, type of cooling water supply (e.g., ocean, lake, natural or man-made reservoir, river, or cooling tower), makeup sources to the ultimate heat sink, and the capability of the heat sink to deliver the required flow of cooling water at appropriate temperatures for normal or accident condition shutdown of the reactor. The UHS is reviewed to determine that design code requirements, as applicable to the assigned quality classifications and seismic categories, are met. A related area of review is the conveying system, which is generally the service water pumping system. The service water system is reviewed under Standard Review Plan (SRP) 9.2.1.

1. The ultimate heat sink is reviewed with respect to the following considerations:

   a. The type of cooling water supply.

   b. The ability to dissipate the total essential station heat load.

   c. The effect of environmental conditions on the capability of the UHS to furnish the required quantities of cooling water, at appropriate temperatures and with any required chemical and purification treatment, for extended times after shutdown.

   d. The effect of earthquakes, tornadoes, missiles, and hurricane winds on the availability of the source water.

e. Sharing of cooling water sources in multi-unit stations.

f. Applicable design requirements such as the high and low water levels of the source to determine their compatibility with the service water system.

2. APCSB reviews the station heat input provided in the SAR for the design of the UHS with respect to reactor system heat, sensible heat, and pump work, and station auxiliary system individual and total heat loads.

3. The proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this review plan.

Secondary reviews will be performed by other branches and the results used by the APCSB to complete overall evaluation of the UHS. The secondary reviews are as follows. The RSB assures that seismic and quality group classifications established for the system components are acceptable. The RSB also confirms heat loads from the reactor coolant and emergency core cooling systems. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that the system is designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components and, upon request, verifies the compatibility of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of electrical components and instrumentation required for UHS operation. The SAB verifies the ultimate heat sink water levels, meteorological and natural phenomena criteria and transient analysis of the cooling water inventory.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the ultimate heat sink, as described in the applicant's safety analysis report (SAR), including related sections of Chapters 2 and 3 of the SAR, is based on specific general design criteria and regulatory guides and on independent calculations and staff judgments with respect to system adequacy. An additional basis for acceptability is the degree of similarity of the UHS design with that for previously reviewed plants with satisfactory operating experience.

The design of the ultimate heat sink is acceptable if the system and the associated complex of water sources, including retaining structures and canals or conduits connecting the sources with the station, are 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, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, relative to structures housing the systems and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with high and moderate energy pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 44, as related to:

   a. The capability to transfer heat loads from safety-related structures, systems, and components to the heat sink under both normal operating and accident conditions.

   b. Suitable component redundancy so that safety functions can be performed assuming a single active component failure coincident with loss of offsite power.

   c. The capability to isolate components, systems, or piping if required so that safety functions are not compromised.

5. General Design Criterion No. 45, as related to the design provisions to permit inservice inspection of safety-related components and equipment.

6. General Design Criterion No. 46, as related to the design provisions to permit operation functional testing of safety-related systems or components.

7. Regulatory Guide No. 1.26, as related to quality group classification of system components.

8. Regulatory Guide No. 1.27, as related to the design and functional requirements of the ultimate heat sink.

9. Regulatory Guide No. 1.29, as related to the seismic design classification of system components.

10. Branch Technical Position APCSB 9-2, as related to the methods for calculating heat release due to fission product and heavy element decay.

III. REVIEW PROCEDURES

The 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 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 as set forth in the final safety analysis report.

The review procedures for OL applications include 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 as a result of the staff's review.

Availability of an adequate supply of water for the ultimate heat sink is a basic requirement for any nuclear power plant. There are various methods of satisfying the requirement, e.g., a large body of water such as an ocean, lake, or natural or man-made reservoir, a river, or cooling ponds or towers, or combinations thereof. The design of the ultimate heat sink tends to be unique for each nuclear plant, depending upon its particular geographical location. For the purpose of this plan, typical procedures are established for use in identifying the essential features of an ultimate heat sink. For installations where these general procedures are not completely adequate, the reviewer supplements them as necessary.

1. The SAR is reviewed for the overall arrangement and type of ultimate heat sink proposed. The reviewer verifies that the UHS is designed so that system function is maintained as required when subjected to adverse environmental phenomenon or a loss of offsite power. The reviewer evaluates the system to determine that:

   a. The heat inputs that are used in the design of the UHS are conservative. The reviewer makes an independent evaluation of the applicant's calculated heat loads. The UHS heat loads include heat due to decay of radioactive material, sensible heat, pump work, and the heat load from the operation of the station auxiliary systems serving and dependent upon the UHS.

   b. Operational data from plants of similar design confirm, where possible, the heat input values given for sensible heat, pump work, and station auxiliary systems.

2. The reviewer verifies that:

   a. The total essential station heat load and system flow requirements of the service water system are compatible with the heat rejection capability of the UHS.

   b. The UHS has the capability to dissipate the maximum possible total heat load, including LOCA under the worst combination of adverse environmental conditions and has provisions for cooling the unit (or units, including LOCA for one unit for a multi-unit station with one heat sink) for a minimum of 30 days without makeup unless acceptable makeup capabilities can be demonstrated. This capability is verified by independent check calculations.

   c. The connecting channels, structures, man-made embankments and dams, and conduits to and from the UHS are capable of withstanding design basis natural phenomena in combination with other site-related events and that a single failure resulting from such phenomena or events cannot prevent adequate cooling water flow or adversely effect the temperature of the water from the sink.

3. Plants utilizing cooling towers as the ultimate heat sink are reviewed as described above and in addition the reviewer determines that:

   a. The tower structure and basin design bases in the SAR include requirements for withstanding design basis natural phenomena or combinations of such phenomena at historically observed intensities. The natural phenomena to be considered include tornadoes, tornado missiles, hurricane winds, and the SSE.

   b. The results of failure modes and effects analyses show that the mechanical systems (fans, pumps, and controls) can withstand a single active failure in any of these systems, including failure of any auxiliary electric power source, and not prevent delivery of water in the quantities and at temperatures required for safe shutdown.

   c. Adequate net positive suction head (NPSH) can be provided to all essential pumps considering variations of water level in the basis. This is verified by performing independent calculations.

   d. The towers can provide the design cooling water temperature under the worst combination of adverse environmental conditions, and that the supply of water in the basins can provide a 30-day capability for long-term cooling at the required temperature without makeup unless acceptable makeup capabilities can be demonstrated. This is verified by independent calculations.

4. Reactor sites that utilize large natural or man-made water sources which for all practical purposes have an infinite supply of water are reviewed as described in items 1 and 2, above, and in addition the reviewer determines:

   a. By evaluation of the SAR information or independent calculations, that the water source is adequate taking into account the effects of design basis natural phenomena such as tornadoes, hurricane winds, probable maximum floods, tsunami, seiches, and the SSE.

   b. By reviewing the SAR preliminary site and plant arrangement sketches (CP) and (OL) site drawings and plant arrangement drawings that the design of the intake and outlet conduits (open or closed type) are properly separated to prevent recirculation or water temperature stratification.

   c. That man-made earth dam, dike, or other structure design bases in the SAR include requirements for withstanding the design basis natural phenomena or combinations of such phenomena at historically observed intensities. In the event of failure of a dam, dike, or other structure not designed to withstand the design basis natural phenomena (particularly the SSE), sufficient water must remain in the source pool to assure a cooling water supply for a minimum of 30 days, with adequate cooling capability so that the required cooling water temperature to the service water system inlet is not exceeded.

5. The reviewer verifies that essential portions of the UHS are classified seismic Category I Quality Group C, or higher and are tornado missile protected.

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 ultimate heat sink review included the size, type of cooling supply [i.e., large body of water, ocean, lake, natural or man-made reservoir, river, pond, or cooling tower], and makeup sources to the ultimate heat sink. The review for the ___________ plant included layout drawings, piping and instrumentation diagrams, process flow diagrams [if any], and descriptive information on the supporting systems that are essential to safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the ultimate heat sink and the requirements for delivering cooling water during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the ultimate heat sink and supporting systems is in conformance with 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 ultimate heat sink 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 ultimate heat sink 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 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 44, "Cooling Water System."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

7. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

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

9. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

RESIDUAL DECAY ENERGY FOR LIGHT WATER REACTORS FOR LONG-TERM COOLING

A. BACKGROUND

The Auxiliary and Power Conversion Systems Branch has developed acceptable assumptions and formulations that may be used to calculate the residual decay energy release rate for light water cooled reactors for long-term cooling of the reactor facility.

Experimental data (Refs. 1 and 2) on total beta and gamma energy releases for long half-life (> 60 seconds) fission products from thermal neutron fission of U-235 have been considered reliable for decay times of 10^3 to 10^7 seconds. Over this decay time, even with the exclusion of short-lived fission products, the decay heat rate can be predicted to within 10 percent of experimental data (Refs. 3, 7, and 8).

The short-lived fission products contribute appreciably to the decay energy for decay times less than 10^3 seconds. Although consistent experimental data are not as numerous (Refs. 4 and 5) and the results of various calculations differ, the effect of all uncertainties can be treated in the zero to 10^3 second time range by a suitably conservative multiplying factor.

B. BRANCH TECHNICAL POSITION

1. Fission Product Decay

For finite reactor operating time (t_o) the fraction of operating power, $\frac{P}{Po}(t_o, t_s)$, to be used for the fission product decay power at a time t_s after shutdown may be calculated as follows:

$$\frac{P}{Po}(\infty, t_s) = \frac{1}{200} \sum_{n=1}^{n=11} A_n \exp(-a_n t_s) \qquad (1)$$

$$\frac{P}{Po}(t_o, t_s) = (1 + K) \frac{P}{Po}(\infty, t_s) - \frac{P}{Po}(\infty, t_o + t_s) \qquad (2)$$

where:

$\frac{P}{Po}$ = fraction of operating power

t_o = cumulative reactor operating time, seconds

t_s = time after shutdown, seconds

K = uncertainty factor; 0.2 for $o \leq t_s \leq 10^3$ and 0.1 for $10^3 \leq t_s \leq 10^7$

A_n, a_n = fit coefficients having the following values:

n	A_n	a_n (sec^{-1})
1	0.5980	1.772×10^{0}
2	1.6500	5.774×10^{-1}
3	3.1000	6.743×10^{-2}
4	3.8700	6.214×10^{-3}
5	2.3300	4.739×10^{-4}
6	1.2900	4.810×10^{-5}
7	0.4620	5.344×10^{-6}
8	0.3280	5.716×10^{-7}
9	0.1700	1.036×10^{-7}
10	0.0865	2.959×10^{-8}
11	0.1140	7.585×10^{-10}

The expressions for finite reactor operation may be used to calculate the decay energy from a complex operating history; however, in accident analysis a suitably conservative history should be used. For example, end of first-cycle calculations should assume continuous operation at full power for a full cycle time period, and end of equilibrium cycle calculations should assume appropriate fractions of the core to have operated continuously for multiple cycle times.

An operating history of 16,000 hours is considered to be representative of many end-of-first or equilibrium cycle conditions and is, therefore, acceptable. In calculating the fission produce decay energy, a 20 percent uncertainty factor (K) should be added for any cooling time less than 10^3 seconds, and a factor of 10 percent should be added for cooling times greater than 10^3 but less than 10^7 seconds.

2. Heavy Element Decay Heat

The decay heat generation due to the heavy elements U-239 and N_p-239 may be calculated according to the following expressions (Ref. 6):

$$\frac{P\ (U\text{-}239)}{Po} = 2.28 \times 10^{-3}\ C\ \frac{\sigma_{25}}{\sigma_{f25}}\ [1 - \exp(-4.91 \times 10^{-4}\ t_o)]\ [\exp(-4.91 \times 10^{-4}\ t_s)] \qquad (3)$$

$$\frac{P(N_p\text{-}239)}{Po} = 2.17 \times 10^{-3}\ C\ \frac{\sigma_{25}}{\sigma_{f25}} \Big\{ 0.007\ [1 - \exp(-4.91 \times 10^{-4}\ t_o)]$$

$$\cdot\ [\exp(-3.41 \times 10^{-6}\ t_s) - \exp(-4.91 \times 10^{-4}\ t_s)]$$

$$+ [1 - \exp(-3.41 \times 10^{-6}\ t_o)]\ [\exp(-3.41 \times 10^{-6}\ t_s)] \Big\} \qquad (4)$$

where:

$\frac{P\ (U\text{-}239)}{Po}$ = fraction of operating power due to U-239

$\frac{P\ (N_p\text{-}239)}{Po}$ = fraction of operating power due to N_p-239

t_o = cumulative reactor operating time, seconds

t_s = time after shutdown, seconds

C = conversion ratio, atoms of Pu-239 produced per atom of U-235 consumed

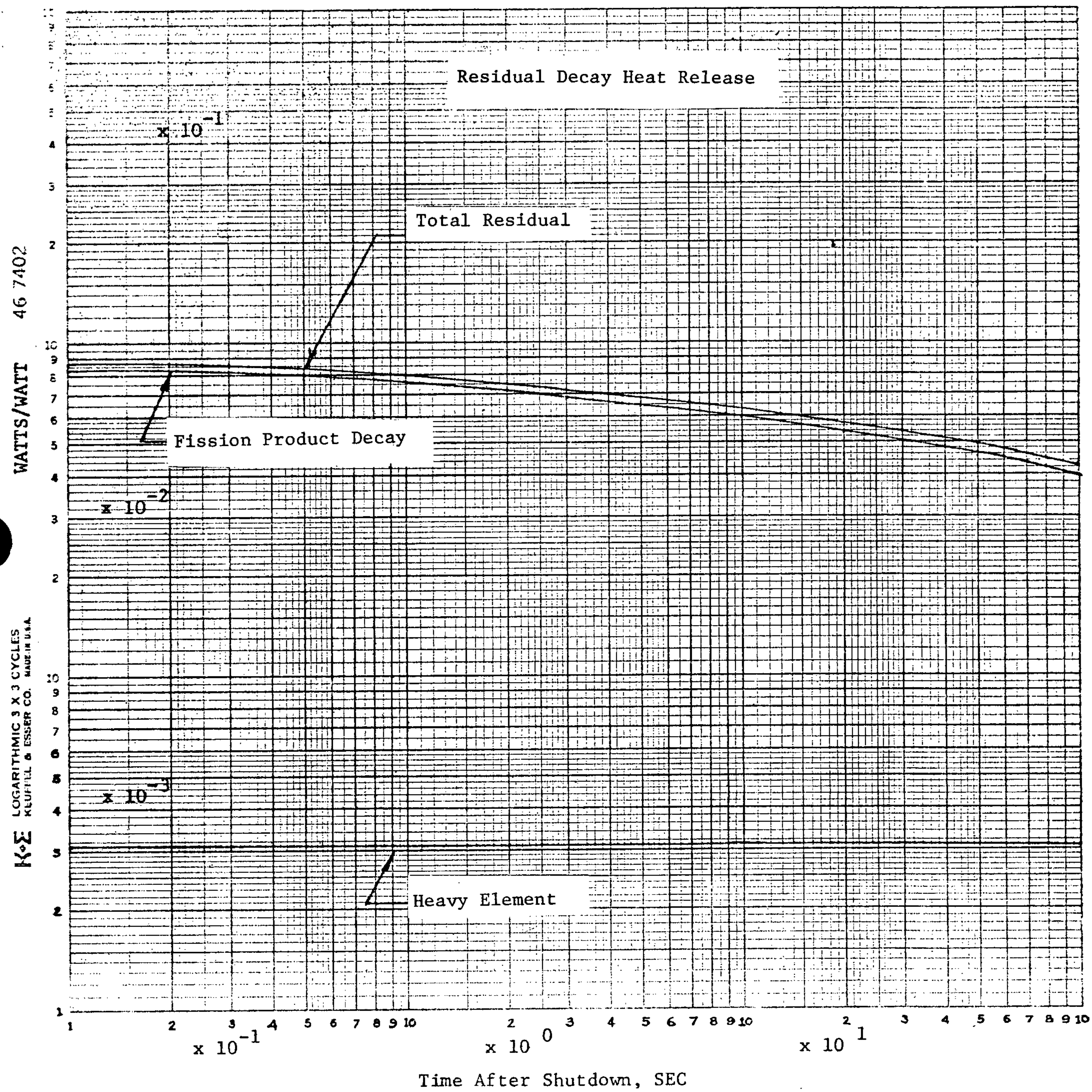
σ_{25} = effective neutron absorption cross section of U-235

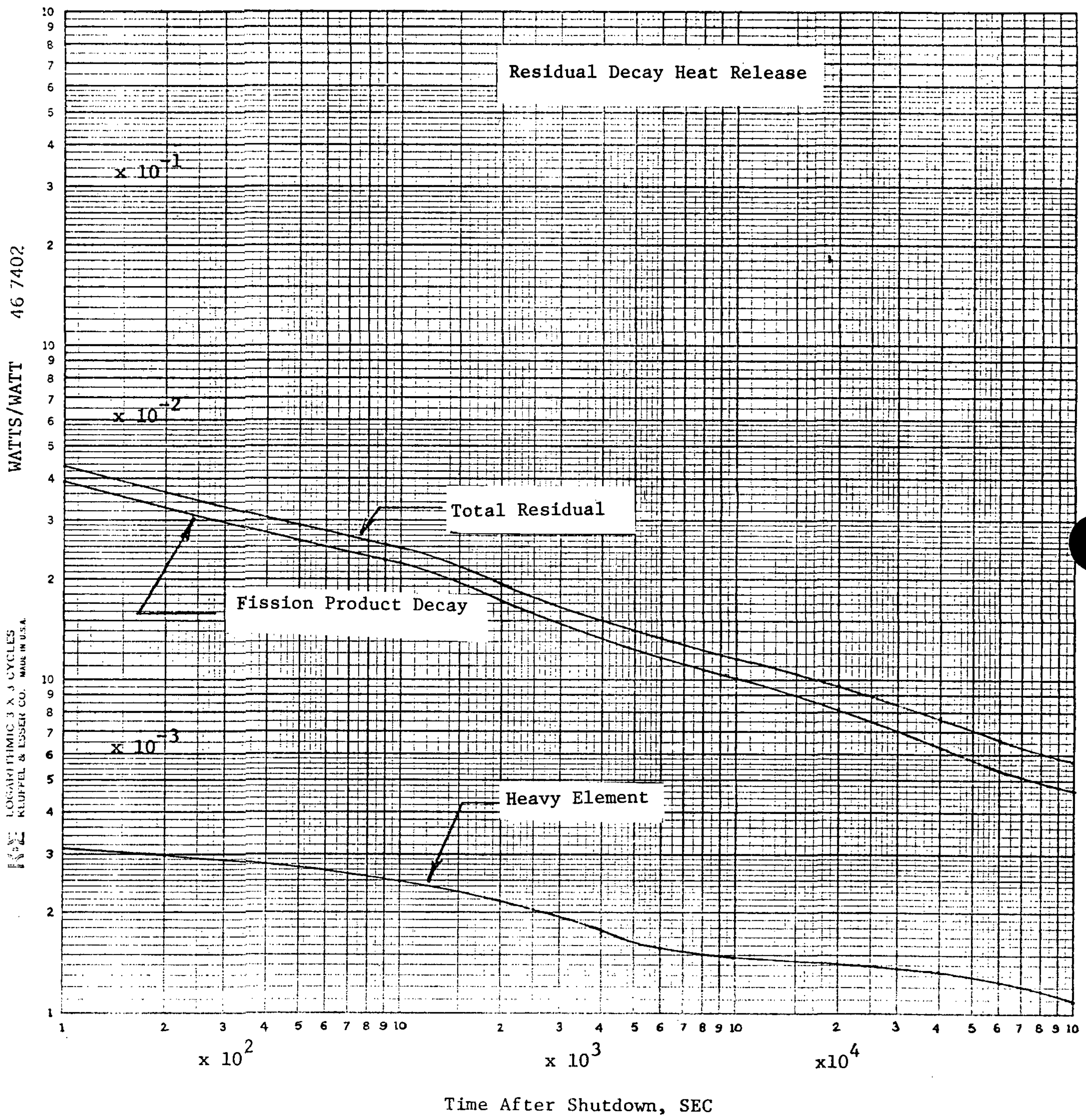
σ_{f25} = effective neutron fission cross section of U-235

The product of the terms $C \cdot \frac{\sigma_{25}}{\sigma_{f25}}$ can be conservatively specified as 0.7.

The nuclear parameters for energy production by the heavy elements U-239 and N_p-239 are relatively well known. Therefore, the heavy element decay heat can be calculated with a conservatively estimated product term of $C \cdot \frac{\sigma_{25}}{\sigma_{f25}}$ without applying any other uncertainty correction factor.

3. Figures 1, 2, and 3 give the residual decay heat release in terms of fractions of full reactor operating power based on a reasonably realistic reactor operating time of 16,000 hours.


Residual Decay Heat Release
Total Residual
Fission Product Decay
Heavy Element
x 10^{-1}
x 10^{-2}
x 10^{-3}
WATTS/WATT
46 7402
K-E LOGARITHMIC 3 X 3 CYCLES
KEUFFEL & ESSER CO. MADE IN U.S.A.
x 10^{-1}
x 10^{0}
x 10^{1}
Time After Shutdown, SEC


Residual Decay Heat Release
x 10^-1
x 10^-2
x 10^-3
Total Residual
Fission Product Decay
Heavy Element
WATTS/WATT
46 7402
LOGARITHMIC 3 X 3 CYCLES
KEUFFEL & ESSER CO. MADE IN U.S.A.
x 10^2
x 10^3
x10^4
Time After Shutdown, SEC

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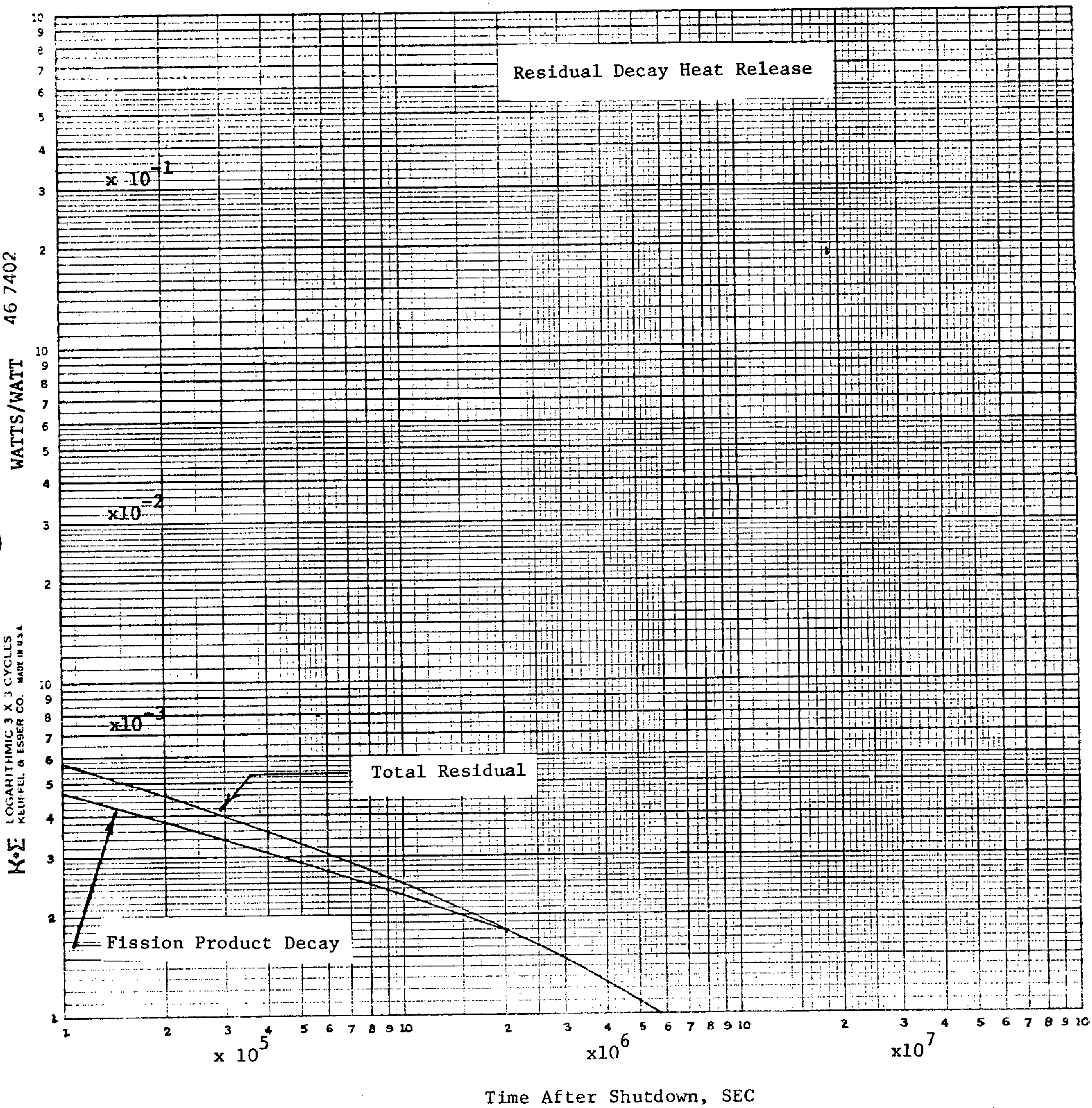
Residual Decay Heat Release
WATTS/WATT
x 10^-1
x10^-2
x10^-3
Total Residual
Fission Product Decay
x 10^5
x10^6
x10^7
Time After Shutdown, SEC
K•E LOGARITHMIC 3 X 3 CYCLES
KEUFFEL & ESSER CO. MADE IN U.S.A.
46 7402

9.2.5-13

C. REFERENCES

1. J. F. Perkins and R. W. King, "Energy Release From the Decay of Fission Products, Nuclear Science and Engineering," Vol. 3, 726 (1958).

2. A. M. Perry, F. C. Maienschein, and D. R. Vondy, "Fission-Product Afterheat: A Review of Experiments Pertinent to the Thermal-Neutron Fission of ^{235}U," ORNL-TM-4197, Oak Ridge National Laboratory, October 1973.

3. A. Tobias, "The Energy Release From Fission Products," Journal of Nuclear Energy, Vol. 27, 725 (1973).

4. J. Scobie, R. D. Scott, and H. W. Wilson, "Beta Energy Release Following the Thermal Neutron Induced Fission of ^{233}U and ^{235}U," Journal of Nuclear Energy, Vol. 25, 1 (1971).

5. L. Costa and R. de Tourreil, "Activite β et α Des Products d'une Fission de ^{235}U et ^{239}Pu," Journal of Nuclear Energy, Vol. 25, 285 (1971).

6. Proposed ANS Standard, "Decay Energy Release Rates Following Shutdown of Uranium - Fueled Thermal Reactors," American Nuclear Society, October 1973.

7. J. Scobie and R. D. Scott, "Calculation of Beta Energy Release Rates Following Thermal Neutron Induced Fission of ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu," Journal of Nuclear Energy, Vol. 25, 339 (1971).

8. K. Shure, "Fission Product Decay Energy," WAPD-BT-24, Westinghouse Electric Corporation, December 1961.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.2.6 CONDENSATE STORAGE FACILITIES

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Effluent Treatment Systems Branch (ETSB)
Materials Engineering Branch (MTEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The condensate storage facility (CSF) is provided to serve as a receiver for excess water generated by other systems such as the main condenser hotwell, the liquid radwaste low activity reprocessed condensate, and the makeup water treatment system, and also to serve as the water supply or makeup source for various auxiliary systems. The APCSB review covers the CSF from the condensate storage tank up to the connections to or interfaces with other systems or components.

1. The APCSB reviews the capability of the CSF to supply water to various auxiliary systems and to receive return water from other systems.

2. The APCSB reviews the CSF to verify that:

   a. Failures of CSF components connected to the emergency core cooling system (ECCS) or other safety-related systems do not adversely affect the safety function of the ECCS or other safety-related systems.

   b. Component redundancy necessary to assure CSF safety functions is provided.

   c. System components meet design code requirements consistent with the component quality group and seismic design classifications.

   d. Provisions for mitigating the environmental effects of system leakage or storage tank failure are provided.

e. Provisions for safe handling of storage tank overflow, the associated instrumentation necessary to detect high or low water level, and isolation means are provided.

3. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the CSF. The secondary reviews are as follows. The RSB will identify essential portions of the facilities that are required to function during normal operations and accident conditions, determine that the seismic and quality group classifications for the system components are acceptable, and assist in establishing the basis for minimum condensate storage capacity. The ETSB will verify that the limits for radioactivity concentrations are not exceeded. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification of components and confirm that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will verify the adequacy of the design, installation, inspection, and testing of all electrical systems (sensing, control, and power) required for proper operation. RAB reviews the facility to assure that radiation levels are as low as possible (ALAP).

II. ACCEPTANCE CRITERIA

Acceptability of the design of the condensate storage facility, 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 condensate storage facility will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

1. For reactor systems where the condensate storage facility is an ultimate means of water supply for safe shutdown or accident mitigation the CSF is acceptable if the integrated facility design is in accordance with the following criteria:

   a. General Design Criterion 44, to assure:

      (1) Redundancy of components so that under normal and accident conditions the safety function can be performed assuming a single active component failure coincident with the loss of offsite power.

      (2) The capability to isolate components, subsystems, or piping if required so that the system safety function will not be compromised.

(3) The capability to provide sufficient makeup water to safety related cooling systems.

b. General Design Criterion 45, as related to design provisions made to permit inservice inspection of safety-related components and equipment.

c. General Design Criterion 46, as related to design provisions made to permit operational functional testing of safety-related systems and components to assure structural integrity, system leak tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

d. General Design Criterion 2, as related to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena, external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

e. General Design Criterion 5, as related to the capability of shared systems and components to perform required safety functions.

f. Regulatory Guide 1.26, as related to the quality group classifications of components and systems.

g. Regulatory Guide 1.29, as related to the seismic design classification of system components.

h. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

2. For reactor systems where the condensate storage facility is not an ultimate means of water supply for safe shutdown or accident mitigation, the design of the CSF is acceptable if the integrated facility design is in accordance with the following criteria:

a. Regulatory Guide 1.29, as related to the seismic design classification of facility components.

b. The concentration of activity in the condensate storage tank is not in excess of the unrestricted levels for liquids given in 10 CFR Part 20, or the tank is provided with a seismic Category I retention basin to preclude the release of the stored liquids to the site in the event of tank failure.

III. REVIEW PROCEDURES

The 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 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 as set forth in the final safety analysis report.

The review of OL applications 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 as a result of the staff's review.

The condensate storage facility (CSF) may be designed either as a safety-related facility or as a non-safety-related facility, depending on the plant. The safety function performed by the facility is to ensure an adequate supply of water to the auxiliary feedwater system in the event that it is required for the safe shutdown of the reactor. Normal plant system functions performed by the CSF, such as makeup to the condenser hotwells and other auxiliary systems of the plant are reviewed to verify that failure will not have an adverse effect on the safety-related functions of the facility.

The review procedures given below are for a typical CSF system of the safety-related type. For cases where there are variations from this typical arrangement, the reviewer will adjust the review procedures given below. However, the system design will be required to meet the acceptance criteria given in Section II.

1. The safety analysis report is reviewed to determine that the facility description section and piping and instrumentation diagrams (PID's) delineate the CSF equipment that is used for normal operations, abnormal operations, and accident conditions as follows:

   a. The facility functional requirements and the minimum flow requirements for supplying water to the auxiliary feedwater system and other safety-related systems are described.

   b. Component allowable operational degradation (e.g., pump leakage) and the procedures that will be followed to detect and correct these conditions when they become excessive are described. The reviewer, using failure modes and effects analyses, comparisons with previously approved facilities, or independent calculations determines that the facility is capable of sustaining the loss of any active component and meeting minimum flow requirements to the safety-related systems.

2. The facility PID's, layout drawings, and component descriptions and characteristics are reviewed to determine the following:

   a. Essential portions of the CSF are correctly identified and are isolable from the non-essential portions of the system. The PID's are reviewed to verify that they clearly indicate the physical division between each portion. System drawings are also reviewed to see that they show the means for accomplishing isolation and the facility description is reviewed to identify minimum performance requirements for the isolation valves.

b. Essential portions of the CSF, including the isolation valves separating seismic Category I portions from the non-seismic portions are, at a minimum, classified Quality Group C or higher and seismic Category I.

c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

3. The reviewer verifies that the system has been designed so that facility functions are maintained, as required, in the event of adverse natural phenomena such as tornadoes, hurricanes, and floods, and a loss of offsite power. The reviewer evaluates the facility, using engineering judgment and the results of failure modes and effects analyses to determine the following:

a. The failure of portions of the facility or of other systems not designed to seismic Category I standards and located close to essential portions of the facility, or non-seismic Category I structures that house, support, or are close to essential portions of the CSF, do not preclude essential functions. Reference to SAR Chapter 2, describing site features, and the general arrangement and layout drawings, as well as to the SAR tabulation of seismic design classifications for structures and facilities will be necessary. Statements in the SAR to the effect that the above conditions are met are acceptable. (CP)

b. The essential portions of the CSF are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and design of the facility and structures are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the facility is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the facility will be located in individual structures that will withstand the effects of both flooding and missiles is acceptable.

c. The CSF provides sufficient net positive suction head (NPSH) at safety-related pump suction locations considering low condensate storage tank water levels. The SAR should indicate the minimum water level of the condensate storage tank and the elevation of the pump impellers. An independent calculation verifying the applicant's conclusion regarding pump NPSH may be necessary.

d. The condensate storage tank is equipped with instrumentation to monitor the water level in the tank and alarm when the water level reaches the low level setpoint which indicates the minimum reserve condensate storage for safety-related system supply.

e. The condensate storage tank overflow piping is connected to the radwaste system.

f. The essential portions of the facility are protected from the effects of high and moderate energy line breaks or cracks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the CSF, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and the procedures for reviewing this information are given in the corresponding review plans.

g. Functions of the essential components and subsystems of the CSF (i.e., those necessary for plant safe shutdown) will not be precluded by a loss of offsite power. The CSF design will be acceptable in this regard if minimum system requirements are met with onsite power.

4. The descriptive information, PID's, system drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the CSF will function as needed following design basis accidents, assuming a concurrent single active component failure. The reviewer evaluates the information presented in the SAR on the ability to function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that system flow requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system flow requirements are met.

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 condensate storage facility (CSF) includes all components and piping associated with the facility to the points of connection or interfaces with other systems. The scope of review of the CSF for the ______________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the facility and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the condensate storage facility and the requirements for sufficient water supply to safety-related systems during normal, abnormal, and accident conditions (CP).] [The review has determined that the design of the condensate storage facility and auxiliary supporting systems is in conformance with the 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 condensate storage facility 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 condensate storage facility 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 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 Criteria 44, "Cooling Water."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

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 I.

9. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.3.1 COMPRESSED AIR SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The compressed air system (CAS) provides air to safety-related equipment, and also to plant equipment used only for normal facility operation. APCSB reviews the entire compressed air system since there may be cases where two systems or subsystems are provided, i.e., a safety-related control air system (SRCAS), and a station service system for non-safety-related equipment. If the two systems are interconnected, then the area of review will extend from the safety-related portion to the outermost isolation valve on all interconnections between the two systems. If the systems are not connected, then the review will be limited to the SRCAS.

1. APCSB reviews the systems to identify the safety-related air operated devices that are supplied by the system, and whether each requires a source of supply air in order to perform the safety-related function.

2. APCSB then reviews to determine that:

   a. A failure of a component, or the loss of a compressed air source does not negate the safety-related functional performance of the system.

   b. The system components and pipes have sufficient physical separation or barriers to protect the essential portions of the system from missiles, and from the effects of breaks and cracks in high and moderate energy fluid system piping close to the SRCAS.

3. The APCSB reviews the system to determine that the effects of failure of non-seismic Category I equipment or components will not affect the functioning of the SRCAS.

4. APCSB reviews the design of the SRCAS with respect to the following:

   a. Capability to isolate portions or components of the system in case of component malfunction.

   b. Instrumentation and control features provided to determine and verify that the system is operating in a correct mode (e.g., valve position indication, pressure).

   c. Functional capability of the system in the event of adverse environmental phenomena, abnormal operational requirements, or accident conditions such as a loss-of-coolant accident (LOCA) or main steam line break concurrent with loss of offsite power.

5. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall review of the system. The secondary reviews are as follows. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification of components and confirm that the system is designed in accordance with applicable codes and standards. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all essential electrical components.

II. ACCEPTANCE CRITERIA

The acceptability of the design of the safety-related control air system, 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 acceptability of the system will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. The design of the SRCAS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapter 2 of the SAR.

2. General Design Criterion 4, with respect to structures housing the systems and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. Regulatory Guide 1.26, as related to the quality group classification of systems and components.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high energy piping or cracks in moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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 as set forth in the final safety analysis report. The procedures for OL reviews include 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 as a result of the staff's review.

As a result of various CAS designs provided for different plants, there will be variations in system requirements. For the purpose of this plan, a typical system is assumed which has two independent systems, the plant service air system, and a safety-related control air system (SRCAS). For cases where there are variations from this arrangement, the reviewer adjusts the review procedures given below. However, the system design would be required to meet the acceptance criteria in Section II. The reviewer will select and emphasize material from this plan as appropriate for a particular case.

1. The SAR is reviewed to identify from information in the system description section and the piping and instrumentation diagrams (P&IDs) the SRCAS equipment used for normal operation and for safety feature operation. The reviewer determines that the system design is acceptable, taking into account the worst expected component operational degradation (e.g., wet or dirty air). The procedures to be followed to detect and correct these conditions when degradation becomes excessive are also reviewed.

2. The reviewer, using the results of failure modes and effects analyses, determines that the system, when operating in the normal mode, is capable of sustaining the loss of any active component. The reviewer determines, on the basis of previously approved systems or independent calculations, that the minimum system requirements (as stated in the SAR) are met for these failure conditions.

3. The system P&IDs, layout drawings, and component descriptions and characteristics are reviewed to determine the following:

    a. Essential portions of the SRCAS are correctly identified and are isolable from the non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical division between each portion. System drawings are also reviewed to verify that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements of the isolation valves. For the typical system, the drawings and descriptions are reviewed to verify that two automatically operated isolation valves in series separate non-essential portions and components from the essential portions.

b. Essential portions of the SRCAS, including the isolation valves separating essential from non-essential portions, are classified Quality Group C or higher and seismic Category I. 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 change in any design classification.

4. The reviewer verifies that the system has been designed so that system function will be maintained, as required, in the event of adverse environmental phenomena, certain pipe breaks, or a loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

a. 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 SRCAS, or of non-seismic Category I structures that house, support, or are close to the SRCAS, will not preclude operation of the essential portions of the SRCAS. Reference to SAR Chapter 2 (which describes site features) and the general arrangement and layout drawings, as well as to the SAR tabulation of seismic design classifications for structures and systems will be necessary. Statements in the SAR to the effect that the above conditions are met are acceptable.

b. The essential portions of the SRCAS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Seismic design, flood protection, and missile protection criteria are discussed in detail in Chapter 3 of the SAR. The location and the design of the system, structures, or cubicles are reviewed to determine that the degree of protection is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of tornado winds, flooding, and missiles is acceptable.

c. An adequate SRCAS air supply source is available, considering the loss of offsite power. The system design will be acceptable if minimum performance requirements, as stated in the SAR, are met assuming a concurrent failure of a single active component, including an emergency power source. The SAR information is reviewed to verify that for each SRCAS component or subsystem affected by the loss of offsite power, system capability meets or exceeds the minimum requirements. Statements in the SAR and the results of failure modes and effects analyses are considered to assure that the system meets these requirements. This will be acceptable verification of system functional reliability.

d. The essential components of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the SRCAS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and procedures for reviewing the information are given in the corresponding review plans.

5. The descriptive information, P&IDs, SRCAS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that the SRCAS will function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates failure modes and effects analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum compressed air flow requirements are met for each degraded situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 compressed air system includes all components and piping and the points of connection or interfaces with other systems. The scope of the review of the compressed air system for the ______________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for operation of essential portions of the system. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the system with regard to the need to maintain a continous air supply to safety-related components during all conditions of plant operation. (CP)] [The review has determined that the applicant's design of the compressed air system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the compressed air system and necessary auxiliary 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 compressed air system 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 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. 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. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.3.2 PROCESS SAMPLING SYSTEM

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Containment Systems Branch (CSB)
Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

ETSB reviews the following information in the applicant's safety analysis report (SAR):

1. The design objectives and design criteria for the process sampling system (PSS) are reviewed at the construction permit (CP) stage. During the operating license (OL) stage of review, 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 review includes identification of the process streams to be sampled and the parameters to be determined through sampling (e.g., gross beta-gamma concentration, boric acid concentration).

2. The system description for the PSS is reviewed at the operating license (OL) stage. The review includes (a) piping and instrumentation diagrams (P&ID's), (b) provisions for obtaining representative samples, (c) location of sampling points and sample stations, and (d) provisions for purging sampling lines.

3. The seismic design and quality group classifications of piping and equipment, and the bases for the classifications chosen are reviewed at the CP stage. At the OL stage, the review includes design and expected temperatures and pressures and materials of construction of components of the system.

4. The isolation provisions for the system and the means provided to limit radioactive releases by limiting reactor coolant losses are reviewed at the CP stage.

Sampling and monitoring systems for radwaste processing systems are reviewed by ETSB under Standard Review Plan (SRP) 11.5. Secondary reviews are performed by the following branches and the results used by ETSB to complete the overall evaluation of the PSS. The CSB, under SRP 6.2.4, reviews the design of isolation provisions of those portions of the PSS that penetrate primary containment. The APCSB, under SRP 3.6.1, reviews the design with respect to the effects of externally or internally generated missiles, pipe whip, and jet impingement forces associated with postulated pipe breaks in high energy fluid systems or leakage cracks in moderate energy fluid systems.

II. ACCEPTANCE CRITERIA

1. The applicant's design should be such that the PSS has the capability for sampling all normal process systems and principal components, including provisions for obtaining samples from at least the following points:

   a. For a pressurized water reactor (PWR):

      Reactor primary coolant.
      Refueling (borated) water storage tank.
      ECCS core flooding tank.
      Concentrated boric acid storage tank.
      Boric acid mix tank.
      Boron injection tank.
      Chemical additive tank.
      Spent fuel pool.
      Secondary coolant.
      Pressurizer tank.
      Steam generator blowdown (if applicable).

   b. For a boiling water reactor (BWR):

      Reactor coolant.
      Standby liquid control system tank.

   The required tests and test frequencies should be given in the plant technical specifications.

2. ETSB will use the following guidelines for determining the acceptability of the system functional design:

   a. Provisions should be made to assure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. Provisions should be made for purging sample lines and for reducing plateout or precipitation in sample lines (e.g., heat tracing). Provisions for sampling should be in accordance with the guidelines in Regulatory Guide 1.21, paragraph C.6.

   b. Provisions should be made to assure representative samples from gaseous process streams and tanks in accordance with ANSI N13.1-1969.

c. Locations of sampling points should be described in the SAR at the OL stage and should be shown on P&IDs describing the system to be sampled.

d. Provisions should be made for purging sampling lines and for reducing plateout in sample lines.

e. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system.

f. Isolation valves should fail in the closed position.

g. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided.

3. The seismic design and quality group classification of sampling lines and components should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification) as described in Regulatory Guides 1.26 and 1.29. Components and piping downstream of the second isolation valve can be designed to Quality Group D and non-seismic Category 1 requirements.

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 review of the process sampling system, ETSB compares the list of process sampling points contained in the SAR with the sampling points identified in Section II, 1, above, to assure that the required process sampling points have been provided.

2. ETSB compares the capability of the system to obtain representative samples of process fluids and the locations of sampling points with the guidelines for obtaining representative samples of fluids contained in paragraph C.6 of Regulatory Guide 1.21 and with the principles for obtaining representative samples of gases contained in ANSI N13.1-1969.

3. ETSB compares the seismic design and quality group classifications of the PSS to the classifications of the fluid systems to which the sampling system is connected.

4. ETSB reviews the technical specifications for process sampling to determine that the content and intent of the technical specifications are in agreement with the requirements developed as a result of the staff's review.

5. ETSB verifies that provisions have been made to limit the potential for reactor coolant loss from the rupture of a sample line.

IV. EVALUATION FINDINGS

ETSB 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 process sampling system includes piping, valves, heat exchangers, and other components associated with the system from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed design. The review has included descriptive information for the process sampling system and the location of sampling points, as shown on piping and instrumentation diagrams.

"The basis for acceptance in our review has been conformance of the applicant's design for the process sampling system to applicable regulations and guides, as well as to branch technical positions and industry standards. Based on our evaluation, we find the proposed system to be acceptable."

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 1.26, "Quality Group Classifications and Standards For Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 2.

3. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

4. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969).

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SECTION 9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Containment Systems Branch (CSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The equipment and floor drainage system (EFDS) is designed to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The APCSB reviews the equipment and floor drainage system, including the collection and disposal of liquid effluents outside containment. This includes piping and pumps from equipment or floor drains to the sumps, and any additional equipment that may be necessary to route effluents to the drain tanks and then to the radwaste system.

1. The APCSB reviews the EFDS capability to collect and dispose of all waste liquid effluents so that they will be processed in a controlled and safe manner. APCSB will determine that:

   a. The system is capable of handling the volume of leakage expected, including the capacities of the sumps, drain tanks, and sump pumps.

   b. The system is capable of preventing a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment.

   c. There is no potential for inadvertent transfer of contaminated fluids to a non-contaminated drainage system.

2. The applicant's proposed technical specifications are reviewed at the operating license stage as they relate to areas covered in this review plan.

Secondary reviews will be performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The ETSB will provide verification that the radwaste system is capable of collecting, sampling, analyzing, and processing the effluents from the EFDS consistent with the requirements for disposal of radwaste material. The CSB will verify that portions of the drain system penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions including accidents. RAB will verify that the system will meet occupational radiation protection criteria of Regulatory Guide 8.8.

II. ACCEPTANCE CRITERIA

1. Acceptability of the design of the equipment and floor drainage system, as described in the applicant's safety analysis report (SAR) is based on the system being designed to prevent the flooding of areas housing safety-related equipment and to prevent the inadvertent transfer of contaminated fluids to non-contaminated drainage systems for disposal.

2. There are no general design criteria or regulatory guides that are directly applicable to the safety-related performance requirements for the EFDS. The APCSB uses the following criteria to determine if portions of the EFDS are safety-related:

   a. If the system is capable of detecting leaks in safety systems that utilize the drainage system sumps, and is the only means for such leakage detection, it is considered safety-related in this regard.

   b. If the system can cause the inundation of safety-related areas due to drain backflow that may result from blockage or the probable maximum flood, it is considered safety-related in this area.

   c. If the system is connected so that an inadvertent transfer of contaminated fluids to non-contaminated drainage systems can occur, it is considered safety-related in this area.

3. The general design criteria and regulatory guides utilized in review of those portions of the system where failure or malfunction could result in adverse effects on essential systems or components (i.e., necessary for safe shutdown, accident prevention, or accident mitigation) follow:

   a. General Design Criterion 2, as related to the capability of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

   b. General Design Criterion 4, with respect to the capability of withstanding the effects of external missiles and internally generated missiles, pipe whip and jet impingement forces associated with pipe breaks.

   c. Regulatory Guide 1.29, as related to the seismic design classification of components.

   d. Regulatory Guide 8.8 related to maintaining occupational radiation exposure as low as practicable.

e. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

4. An additional basis for determining the acceptability of safety-related portions of the EFDS will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

III. REVIEW PROCEDURES

The 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 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 procedures for OL applications include 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 as a result of the staff's review.

The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. The SAR is reviewed to see that the EFDS description section, layout drawings, and piping and instrumentation diagrams (P&IDs) show the EFDS layout and equipment, including pumps and valves necessary for routing effluents, the minimum drain tank capacity system flow requirements, connections to areas containing safety-related equipment or to non-contaminated drain systems, and any use made of the EFDS for leakage detection for safety-related systems. The reviewer determines which portions of the EFDS have safety functions or can adversely affect safety-related systems, using the criteria of Section II.2, above. These "essential" portions of the EFDS are then reviewed on the basis of the criteria of Section II.3, as is described in the paragraphs that follow.

2. The EFDS performance requirements section of the SAR is reviewed to confirm that it describes component allowable operational degradation (e.g., drain blockage, sump pump leakage, or failures) for safety-related portions of the system and describes the procedures that will be followed to detect and correct these conditions if they become excessive. The reviewer determines that essential portions of the system can sustain the loss of any active component and meet minimum system requirements. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed for the following points:

    a. Essential portions of the EFDS are correctly identified and are isolable from the non-essential portions of the system to the extent required by system performance requirements.

b. Essential portions of the EFDS are classified Quality Group C or higher and seismic Category I. Components and system descriptions in the SAR are reviewed to verify that the seismic and safety classifications have been included, and that the P&IDs indicate any points of change in piping quality group classification.

3. The reviewer verifies that the system safety functions will be maintained, as required, in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks. The reviewer evaluates the system, using engineering judgment, failure modes and effects analyses, and the results of reviews performed under other review plans, to determine that:

   a. 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 that house, support, or are close to essential portions of the EFDS, will not preclude operation of the essential portions of the EFDS. Reference to SAR Chapter 2 (which describes site features) and the general arrangement and layout drawings will be necessary. Statements in the SAR to the effect that the above conditions are met are acceptable.

   b. System capability to prevent drain or flood water from backing up in the drainage system into areas housing safety-related equipment has been incorporated. Statements in the SAR that this capability is provided are acceptable.

   c. Provisions are made in the system to control and direct the flow of radioactive waste fluids to the radwaste area. It will be acceptable if the system P&IDs and design criteria show that the potential for inadvertent transfer of contaminated fluids to noncontaminated drainage system for disposal has been precluded.

   d. Essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the EFDS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR, and the procedures for reviewing this information are given in the corresponding review plans.

4. The descriptive information, P&IDs, EFDS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function as required following design basis accidents, assuming a concurrent failure of a single active component. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system flow requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

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 equipment and floor drainage system includes all piping from equipment or floor drains to the sump, the sump pumps, and the associated pumps and piping network necessary to route effluents to the drain tanks and then to the radwaste system. The scope of review of the equipment and floor drainage system for the ____________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the equipment and floor drainage system and the auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the equipment and floor drainage system, and the requirements for continuous removal of liquids from areas containing safety-related equipment during normal, abnormal, and accident conditions. (CP)] [The review has determined that the applicant's design of the equipment and floor drainage systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the essential portions of the equipment and floor drainage system and necessary auxiliary 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 equipment and floor drainage system 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 Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases."

3. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

4. Regulatory Guide 1.26, "Quality Group Classifications and Standards For Water-, Steam-, And Radioactive-Waste-Containing Components of Nuclear Power Plants."

5. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

6. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM (PWR) (INCLUDING BORON RECOVERY SYSTEM)

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Core Performance Branch (CPB)
Reactor Systems Branch (RSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Effluent Treatment Systems Branch (ETSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

Pressurized water reactor (PWR) plants include a chemical and volume control system (CVCS) and boron recovery system (BRS). These systems maintain the required water inventory and quality in the reactor coolant system (RCS), provide seal-water flow to the reactor coolant pumps, control the boron neutron absorber concentration in the reactor coolant, and control the primary water chemistry. Further, the system provides recycled coolant for the demineralized water makeup system for normal operation and high pressure injection flow to the emergency core cooling system in the event of postulated accidents.

1. The APCSB reviews the systems from the letdown line of the primary system to the charging lines that provide makeup to the primary system and the reactor coolant pump seal-water system. The system is reviewed to the interfaces with the demineralized water makeup system and radioactive waste system.

2. The APCSB reviews the functional performance characteristics of CVCS components and the effects of adverse environmental occurrences, abnormal operational requirements, or accident conditions such as those due to a loss-of-coolant accident (LOCA).

3. The APCSB reviews the system to determine that a malfunction, a single failure of an active component, or the loss of a cooling source will not reduce the safety-related functional performance capabilities of the system.

4. The system is reviewed with respect to the effects of postulated breaks or leakage cracks in high and moderate energy piping.

5. The system is reviewed to determine that quality group and seismic design requirements are met. The effects of failure of equipment or components not designed to withstand seismic events on safety-related functions of the system are evaluated.

6. The APCSB reviews the system design with respect to the capability to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions. RAB reviews the system with respect to maintaining occupational radiation exposure as low as practicable.

7. The APCSB reviews the system features provided to prevent precipitation of boric acid in components and lines containing boric acid solutions, and the adequacy of the system design to protect personnel from the effects of toxic, irritating, or explosive chemicals that may be used.

8. Provisions for operational testing are evaluated, as are the instrumentation and control features that determine and verify that the system is operating in the correct mode.

9. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete overall evaluation of the system. The secondary reviews are as follows. The CPB determines the adequacy of the specified boron concentrations in the primary coolant for normal and accident conditions. The RSB determines that the assigned seismic and quality group classifications for system components are acceptable. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maxium flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components and upon request will verify the compatibility of the materials of construction with service conditions. The EICSB evaluates the controls, instrumentation, and power sources with respect to capability, capacity, and reliability to perform safety-related functions during normal and emergency conditions. The ETSB reviews the CVCS and BRS to determine the source terms for possible radioactive releases and the processing of radioactive effluent from the BRS by the waste management systems. The RAB will verify the system meets radiation protection criteria.

II. ACCEPTANCE CRITERIA

Acceptability of the CVCS and BRS design, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. Additional bases for determining the acceptability of the CVCS and BRS include the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience, and independent calculations by the staff. Listed below are specific criteria related to the CVCS and BRS.

The design of the CVCS and BRS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to structures housing the facility and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 26, as related to the CVCS capability to control the rate of reactivity changes resulting from normal power changes and the capability to maintain the reactor core subcritical under cold conditions.

5. General Design Criterion 27, as related to the CVCS capability to control reactivity changes so that under postulated accident conditions, and with appropriate margin for a stuck control rod, the capability to cool the core is maintained.

6. General Design Criterion 33, as related to the CVCS capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary so that specified fuel design limits are not exceeded.

7. General Design C-iterion 60, as related to the handling of radioactive contaminants.

8. Regulatory Guide 1.26, as related to quality group classifications.

9. Regulatory Guide 1.29, as related to seismic design classifications.

10. Regulatory Guide 8.8 as related to maintaining occupational radiation exposure as low as practicable.

11. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

II. <u>REVIEW PROCEDURES</u>

The 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 procedures for OL applications include 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 as a result of the staff's review.

For the purpose of this review plan, a typical system is assumed for use as a guide since the design of the CVCS will vary with each reactor plant supplier. It is assumed that the typical system consists of heat exchangers to cool the letdown flow from the RCS before processing through the demineralizers and to reheat it prior to reinjection into the RCS, demineralizers and filters for removal of suspended and dissolved impurities, high pressure charging pumps to inject makeup flow into the RCS, a volume control tank for system surge capacity and makeup volume, a boron makeup and storage system to provide neutron absorber to the RCS as needed, evaporators and tanks for boron recovery and demineralized water makeup, and a boron thermal regeneration subsystem to minimize the quantity of waste water and allow reactivity control by varying the temperature of demineralizers so as to remove or add boron to the CVCS. For cases where there are variations from this system the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II.

1. The SAR is reviewed to determine that the system description section and piping and instrumentation diagrams P&IDs show the CVCS equipment that is used for normal operation, and the minimum system heat transfer and flow requirements for normal plant operation. The system performance requirements section will also be reviewed to determine that it limits expected component operational degradation (e.g., pump leakage, heat exchanger scaling, resin deterioration) and describes the procedures that will be followed to detect and correct these conditions when they become excessive. The reviewer, using the results of failure modes and effects analyses, comparisons with previously approved systems, or independent calculations, as appropriate, determines that the system can sustain the loss of any active component and meet the minimum system requirements for site shutdown or accident mitigation. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed for the following points:

    a. Essential portions of the CVCS are correctly identified and are verified to be isolable from the non-essential portions of the system. The P&IDs will be reviewed to verify that they clearly indicate physical divisions between such portions and indicate design classification changes. System drawings are also 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.

    b. Essential portions of the CVCS, including the isolation valves separating essential portions from non-essential portions, are classified Quality Group C or higher and seismic Category I. Component and system descriptions in the SAR are reviewed to verify that the above seismic and safety classifications have been included, and that the P&IDs indicate any points of change in piping quality group classification.

    c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the connections and special piping and equipment required by this program.

d. The system description and drawings are reviewed in conjunction with the reactor coolant system to determine that the CVCS has sufficient pumping capacity to maintain the RCS water inventory within the allowable pressurizer level range for all normal modes of operation, including startup from cold shutdown, full power operation, and plant cooldown. It is further ascertained from a review of the P&IDs that makeup to the RSC can be accomplished via two redundant appropriately designed flow paths.

e. Using the results of evaluations performed by the CPB, the APCSB verifies the adequacy of the system for reactivity control in the following areas:

(1) Boration of the reactor coolant system is accomplished through either of two flow paths and from either of two boric acid sources. This is verified from the review of P&IDs and system description.

(2) The amount of boric acid stored in the CVCS exceeds the amount required to borate the reactor coolant system to cold shutdown concentration, assuming that the control assembly with the highest reactivity worth is held in the fully withdrawn position, and to compensate for subsequent xenon decay during any part of core life. This is verified from a review of the SAR.

(3) The CVCS is capable of counteracting the inadvertent positive reactivity insertion caused by the maximum boron dilution accident.

f. The adequacy of the CVCS for control of water chemistry is verified by examination of the information provided in the SAR, i.e., the allowable ranges for primary coolant activity, total dissolved solids, pH, and maximum allowable oxygen and halide concentrations.

g. The adequacy of resin overtemperature protection is verified by reviewing the system description and drawings to determine that temperature sensors are provided that will actuate the demineralizer bypass or isolation valves.

h. The boron thermal regeneration subsystem is reviewed to determine the maximum change in primary coolant boron concentration due to equipment or control errors as determined from failure modes and effects analyses.

i. The operating procedures and controls for boron addition and primary coolant dilution are reviewed for adequacy.

j. The system P&IDs are examined to determine that all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid.

2. The reviewer verifies that the safety function of the system will be maintained as required in the event of adverse environmental phenomena such as earthquakes, tornadoes,

hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. The reviewer uses engineering judgement, failure modes and effects analyses, and the results of reviews performed under other review plans, as applicable, to determine the following:

a. The failure of 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 that house, support, or are close to essential portions of the CVCS, will not preclude operation of the essential portions of the CVCS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)

b. The essential portions of the CVCS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and the design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. Essential portions of the system are protected from the effects of high energy line breaks and moderate energy line cracks. Layout drawings of the system are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the CVCS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing the information presented are given in the corresponding review plans.

d. Essential components and subsystems (i.e., those necessary for safe shutdown) can function as required in the event of loss of offsite power. The system design will be acceptable if the CVCS meets minimum system requirements as stated in the SAR assuming a failure of a single active component, within the system or in the auxiliary electric power source, which supplies the system. The SAR is reviewed to verify that for each CVCS component or subsystem affected by the loss of offsite power, boric acid addition and coolant charging capabilities meet or exceed minimum requirements. Statements in the SAR and the results of failure modes and effect analyses are considered in assuring that the system meets these requirements. This will be an acceptable verification of system functional reliability.

3. The descriptive information, P&IDs, layout drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system

will function following design basis accidents assuming a single active component failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system flow and heat transfer requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

4. The boron recovery system is not required for safe shutdown, or for the prevention or mitigation of postulated accidents. The BRS will be reviewed for the following:

   If the system tankage is of non-seismic Category I design, the results of analyses which postulate the rupture of tanks are reviewed to verify that the accident releases are in accordance with safe limits. The facility design, including P&IDs, are reviewed to assure that safety-related equipment will not be adversely affected by flooding.

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 chemical and volume control system (including boron recovery system) includes components and piping associated with the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system and the reactor coolant pump seal water system. The scope of review of the chemical and volume control system for the ___________ plant included process flow diagrams, layout drawings, piping and instrumentation diagrams, and descriptive information for the system and for the supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the chemical and volume control system, and the requirements for system performance of necessary functions during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the chemical and volume control system and supporting systems is in conformance with the 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 chemical and volume control system and its 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 chemical and volume control system 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 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 26, "Reactivity Control System Redundancy and Capability."

5. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

6. 10 CFR Part 50, Appendix A, General Design Criterion 33, "Reactor Coolant Makeup."

7. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

8. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," Revision 1.

9. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

10. Regulatory Guide 8.8 "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

11. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems OUtside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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STANDARD REVIEW PLAN

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SECTION 9.3.5 STANDBY LIQUID CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Core Performance Branch (CPB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

Boiling water reactor (BWR) plants include a standby liquid control system (SLCS) that provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. This system has the capability for controlling the reactivity difference between the steady-state operating condition at any time in core life and the cold shutdown condition. The review covers the SLCS design to the point where the system connects to the reactor coolant system (RCS). The APCSB reviews the system to determine its adequacy to perform the shutdown function. Other points reviewed by APCSB are as follows:

1. The functional performance characteristics of SLCS components and the effects of adverse environmental occurrences, abnormal operational conditions, or accident conditions such as those due to a loss-of-coolant accident (LOCA).

2. The system to determine that a malfunction or a single failure of a component will not reduce the safety-related functional performance capabilities of the system.

3. The system with respect to the effects of postulated breaks and cracks in high and moderate energy piping.

4. To determine that quality group and seismic design requirements are met for the system.

5. The system design with respect to the capability to detect, collect, and control system leakage and the capability to isolate portions of the system in case of excessive leakage or component malfunctions.

6. The capability of the system to prevent precipitation of the neutron absorber in components and lines containing the absorber solutions.

7. The provisions for operational testing and the instrumentation and control features that verify that the system is available to operate in the correct mode.

8. The applicant's proposed technical specifications for operating license applications as they relate to areas covered in this plan.

Secondary review evaluations are performed by other branches to complete the overall evaluation of the system. The secondary reviews are as follows. The CPB determines the adequacy of the specified boron neutron absorber quantities and concentrations required in the primary coolant to assure that the plant can be brought from rated power to cold shutdown at any time in core life with the control rods withdrawn in the rated power pattern. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification testing of components and confirms that components, piping, and structures 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 compatibility of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of electrical components (sensing, control, and power) required for proper operation.

II. ACCEPTANCE CRITERIA

Acceptability of the SLCS design, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. Additional bases for determining the acceptability of the SLCS include the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience and independent calculations by the staff. Listed below are specific acceptance criteria related to the SLCS.

The design of the SLCS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 21, as related to system design requirements for high functional reliability, inservice testability, and capability to meet the single failure criterion.

4. General Design Criterion 26, as related to the requirement that two independent reactivity control systems of different design principles be provided, and the requirement that one of the systems shall be capable of holding the reactor subcritical in the cold condition.

5. General Design Criterion 27, as related to the SLCS capability to control the rate of reactivity changes resulting from normal power changes and the capability to maintain the reactor core subcritical under cold conditions.

6. Regulatory Guide 1.26, as related to the quality group classification of system components.

7. Regulatory Guide 1.29, as related to the seismic design classification of system components.

8. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside the drywell.

III. REVIEW PROCEDURES

The 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 procedures for OL applications include a determination that the technical specifications prepared by the applicant are in agreement with the requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

For the purpose of this review plan, a typical system is assumed for use as a guide. It is assumed that the SLCS consists of a boron solution tank, a test water tank, two positive displacement pumps, two explosive valves, and associated local valves and controls. For cases where there are variations from this system, the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II.

1. The SAR is reviewed to determine that the system description section and piping and instrumentation diagrams (P&IDs) delineate the SLCS equipment. The reviewer, using the results of failure modes and effects analyses, comparisons with previously approved systems, or independent calculations, as appropriate, determines that the system can sustain the loss of any active component and meet the minimum system

requirements for the safe shutdown of accident mitigation. The system P&IDs, layout drawings, and component descriptions and characteristics are reviewed to determine the following:

a. The SLCS is classified Quality Group B or higher and seismic Category I. Component and system descriptions in the SAR should verify that these classifications have been included, and the P&IDs should indicate any points of change in piping quality group classification.

b. Design provisions have been made that permit appropriate inservice inspection and functional testing of the system. It will be acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the connections and special piping and equipment required by this program.

c. Using the results of the evaluation performed by the Core Performance Branch, the APCSB determines that the system has the capability to store the required quantity of neutron absorber in solution and that the injection rate is sufficient to bring the reactor from rated power to cold shutdown at any time in core life with the control rods remaining withdrawn in the rated power pattern, taking into account the reactivity gains from complete decay of the rated power xenon inventory, an allowance for imperfect mixing and leakage, and dilution by the residual heat removal system.

d. The system PID's indicate that adequate means are provided to maintain the system temperature above the saturation temperature of the neutron absorber solution.

e. The controls and the summary of operating and test procedures for neutron absorber addition are adequate.

2. The reviewer verifies that the safety function of the system will be maintained as required in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. The reviewer uses engineering judgment, failure modes and effects analyses, and the results of reviews performed under other review plans, as applicable, to determine the following:

a. The failure of systems not designed to seismic Category I standards and located close to essential portions of the system, or of non-seismic structures that house, support, or are close to essential portions of the SLCS, will not preclude operation of the SLCS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)

b. The SLCS is protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and the design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. Essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings of the system are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the SLCS or that protection from the effects of failure is provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing the information presented are given in the corresponding review plans.

d. Essential components and subsystems (i.e., those necessary for safe shutdown) can function as required in the event of loss of offsite power. The system design is acceptable if the SLCS meets minimum system requirements as stated in the SAR assuming a failure of a single active component within the system or in the auxiliary electric power source which supplies the system. Statements in the SAR and the results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This will be an acceptable verification of system functional reliability.

3. The descriptive information, PID's, layout drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents assuming a single active component failure. The reviewer evaluates the information in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system flow requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum systems requirements are met.

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 standby liquid control system (SLCS) includes storage tanks, pumps, valves, and piping to the point where the system connects to the reactor coolant boundary. The SLCS is provided on BWR's only. The scope of review of the SLCS for the ________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the systems and for the supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the standby liquid control system, and the requirements for system functions to provide reactivity control during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the standby liquid control system and supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's design, design criteria, and design bases for the SLCS 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 SLCS 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 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 21, "Protection System Reliability and Testability."

4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

5. General Design Criterion 27, "Combined Reactivity Control Systems Capability."

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. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.4.1 CONTROL ROOM AREA VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Reactor Systems Branch (RSB)
Accident Analysis Branch (AAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of main control room personnel and to assure the operability of main control room components during normal operating, anticipated operational transient, and design basis accident conditions.

The APCSB reviews the CRAVS from the air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents. The review includes components such as air intakes, ducts, air conditioning units, filters, blowers, isolation dampers or valves, and exhaust fan The review of the CRAVS covers the control room, switchgear and battery room, access control area, control building heating, ventilating, and air conditioning (HVAC) equipment room, and computer room.

1. The APCSB reviews the CRAVS to determine the safety significance of the system. Based on this determination, the safety-related part of the system is reviewed with respect to the functional performance required to maintain a habitable control room area during adverse environmental occurrences, during normal operation, anticipated operational occurrences, and subsequent to postulated accidents. The review includes the effects of radiation, combustion and other toxic products, and the coincidental loss of offsite power. The APCSB reviews safety-related portions of the system to assure that:

   a. A single active failure cannot result in loss of the system functional performance capability.

   b. Components and piping have sufficient physical separation or barriers to protect essential portions of the system from missiles and pipe whip.

   c. Failures of non-seismic Category I equipment or components will not affect the CRAVS.

2. The APCSB also reviews safety-related portions of the CRAVS with respect to the following:

    a. The ability of the control room heating and cooling subsystems to maintain a suitable ambient temperature for control room personnel and equipment.

    b. The capability to detect in-leakage of radioactivity or airborne chemical contaminants to the control room and the ability to isolate the system to preclude their entrance.

    c. The ability to detect, filter, or expedite safe discharge of airborne contaminants inside the control room.

    d. The provisions for the detection and isolation of portions of the system in the event of fires, failures, or malfunctions.

    e. The ability of essential equipment being serviced by the ventilation system to function under the worst anticipated degraded CRAVS performance.

3. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB, upon request, reviews the seismic qualification of components and confirms that components, piping, and structures 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 will verify the compatability of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of all essential electrical components. The AAB evaluates the concentrations of airborne contaminants in the vicinity of the intake and exhaust vents resulting from accident releases on the plant site. The ETSB verifies the effectiveness of the CRAVS filtration system to remove radioactive and chemical contaminants.

II. ACCEPTANCE CRITERIA

Acceptability of the CRAVS design, 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 CRAVS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of safety-related portions of the CRAVS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety.

4. General Design Criterion 19, as related to providing adequate protection to permit access and occupancy of the control room under accident conditions.

5. Regulatory Guide 1.26, as related to the quality group classification of systems and components.

6. Regulatory Guide 1.29, as related to the seismic design classification of system components.

7. Regulatory Guide 1.52, as related to system design requirements, maximum system flow requirements, system functional performance requirements, design provisions for radiation detection, and isolation provisions.

8. Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators Against An Accidental Chlorine Release."

9. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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. The procedures for OL reviews include a determination that the content and intent of the proposed technical specifications are in agreement with the requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

As a result of various CRAVS designs proposed by applicants, there will be variations in system requirements. For the purpose of this review plan, a typical system with redundant subsystems is assumed with each subsystem having an identical essential (safety features) portion. For cases where there are variations from this typical arrangement, the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II. The reviewer will select and emphasize material from this plan as may be appropriate for a particular case.

1. The SAR is reviewed to verify that the system description and piping and instrumentation diagrams (PID's) show the CRAVS equipment used for normal operation, the ambient temperature limits for the areas serviced, and the filtration capacities of the intake and exhaust filters. The system performance requirements section is reviewed to determine that it describes allowable component operational degradation (e.g., loss of cooling function, damper leakage) and describes the procedures that will be followed to detect and correct these conditions. The reviewer, using results from failure modes and effects analyses, determines that the safety-related portion of the system is capable of functioning in spite of the loss of any active component.

2. The system PID's, layout drawings, and component descriptions and characteristics are then reviewed to determine that:

   a. Essential portions of the CRAVS are correctly identified and are isolable from non-essential portions of the system. The PID's are reviewed to verify that they clearly indicate physical divisions between such portions and indicate design classification changes. System drawings are also reviewed to verify that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements for the isolation dampers. For the typical system, the drawings and description are reviewed to verify that two automatically operated isolation dampers in series separate non-essential portions and components from the essential portions.

   b. Essential portions of the CRAVS, including the isolation dampers separating essential from non-essential portions are classified Quality Group C or higher and seismic Category I. 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 PID's indicate points of change in design classification.

   c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around pumps or isolation valves that would be required by this program.

3. The reviewer verifies that the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or in the event of certain pipe breaks or loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

a. The failure of non-essential portions of the system or of other non-essential systems, structures or components located close to essential portions of the system will not preclude operation of the essential portions of the CRAVS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)

b. The essential portions of the CRAVS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail in Chapter 3 of the SAR. The location and the design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. The total system has the capability to detect and control leakage of airborne contamination into the system. It is acceptable if the following conditions are met:

   (1) The system PID's show monitors located in the system intakes that are capable of detecting radiation, smoke, and toxic chemicals. The monitors should actuate alarms in the control room.

   (2) The capability for isolation of non-essential portions of the CRAVS by two automatically actuated dampers in series is shown on the PID's.

   (3) The CRAVS has provisions for an internal recirculation filtering mode of operation or can discharge airborne contaminants from the control room area using a once-through ventilation mode, as applicable.

   (4) Provisions for isolation of the control room upon smoke detection at the air intakes are shown on the PID's. The isolation may be actuated manually for most cases. Automatic isolation may be required in special cases such as for fires resulting from aircraft crashes.

d. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the CRAVS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

e. Essential components and subsystems can function as required in the event of loss of offsite power. The system design will be acceptable if the CRAVS meets minimum system requirements as stated in the SAR assuming a failure of a single active component within the system itself or in the auxiliary electric power source which supplies the system. The SAR is reviewed to see that for each CRAVS component or subsystem affected by the loss of offsite power, the resulting system operation will not affect safety of control room personnel or the performance of any essential control room equipment. Statements in the SAR and the results of failure modes and effects analyses are considered in verifying that the system meets these requirements. This will be an acceptable verification of system functional reliability.

4. The descriptive information, PID's, CRAVS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system isolation or filtration requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 control room area ventilation system (CRAVS) includes all components and ducting from the intake vents to the exhaust structure. The scope of review of the CRAVS for the ____________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the control room area ventilation system and the requirements for system performance to maintain a suitable environment during all normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the control room area ventilation system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the CRAVS and necessary auxiliary 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 CRAVS 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 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 19, "Control Room."

5. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power PLants," Revision 1.

6. Regulatory Guide 1.29, "Seismic Design Classification."

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

8. 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."

9. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.4.2 SPENT FUEL POOL AREA VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Accident Analysis Branch (AAB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Reactor Systems Branch (RSB)
Effluent Treatment Systems Branch (ETSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents.

The APCSB reviews the SFPAVS from air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or the station vents. The review includes components such as air intakes, ducts, air conditioning units, filters, blowers, isolation dampers, and exhaust fans. The review of the SFPAVS covers all areas containing or adjacent to the spent fuel pool, including the spent fuel cooling pump room.

1. The APCSB reviews the SFPAVS to determine the safety significance of the system. Based on this determination, the safety-related part of the system is reviewed with respect to functional performance requirements during normal operation, during adverse environmental occurrences, and subsequent to postulated accidents including the loss of offsite power. The APCSB reviews safety-related portions of the system to assure that:

   a. A single active failure cannot result in loss of the system functional performance capability.

   b. Components and piping or ducting have sufficient physical separation or barriers to protect essential portions of the system from missiles and pipe whip.

c. Failures of non-seismic Category I equipment or components will not affect the SFPAVS.

2. The APCSB also reviews safety-related portions of the SFPAVS with respect to the following:

   a. The capability to detect and monitor radiation levels in the pool area.

   b. The capability to direct ventilation air from areas of low radioactivity to areas of potentially higher radioactivity.

   c. The capability to detect the need for isolation and to isolate portions of the system in the event of failures or malfunctions.

   d. The capability to actuate components not normally operating that are required to operate during accident conditions, and to provide necessary isolation.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The ETSB verifies that the system functional performance conforms to acceptable limits for radioactivity release during normal operations. The RAB reviews the system capability to monitor radiation levels in the pool. RAB also verifies the system meets the radiation protection criteria. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing or supporting the system to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will, upon request, review the seismic qualification of components and confirm that the components, piping, and structures 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, will verify the compatability of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of all essential electrical components. The AAB evaluates the radiological consequences resulting from a postulated fuel handling accident and the effectiveness of the filtration system to remove radioactive contaminants.

II. <u>ACCEPTANCE CRITERIA</u>

Acceptability of the SFPAVS design, 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 SFPAVS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of safety-related portions of the SFPAVS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety.

4. General Design Criterion 60, as related to the handling of radioactive materials in the SFPAVS.

5. General Design Criterion 64, as related to the monitoring of gaseous releases through the SFPAVS.

6. Regulatory Guide 1.13, as related to the system capability to limit releases of radioactive contaminants to the environment.

7. Regulatory Guide 1.26, as related to the quality group classification of systems and compenents.

8. Regulatory Guide 1.29, as related to the seismic design classification of system components.

9. Regulatory Guide 1.52, as related to system design requirements, maximum system flow requirements, and system functional performance requirements.

10. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

11. Branch Technical Positions SPCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping system outside containment.

III. REVIEW PROCEDURES

The 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 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 procedures for OL reviews include 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 as a result of the staff's review.

As a result of various SFPAVS designs proposed by applicants, there will be variations in system requirements. For the purpose of this review plan, a typical system is assumed which has fully redundant subsystems, each having an identical essential (safety features) portion. For cases where there are variations from this typical arrangement, the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II. The reviewer will select and emphasize material from this plan as may be appropriate for a particular case.

1. The SAR is reviewed to verify that the system description section and piping and instrumentation diagrams (P&IDs) show the SFPAVS equipment used for normal operation, the ambient temperature limits for the area serviced, and the filtration capacity of the exhaust filters. The system performance requirements section is reviewed to determine that it describes allowable component operational degradation (e.g., loss of cooling function, damper leakage) and describes the procedures that will be followed to detect and correct these conditions. The reviewer, using results from failure modes and effects analyses as appropriate, determines that the safety-related portion of the system is capable of functioning in spite of the loss of any active component.

2. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed to determine that:

   a. Essential portions of the SFPAVS are correctly identified and are isolable from non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical divisions between such portions and indicate design classification changes. System drawings are also reviewed to verify that they show the means for accomplishing isolation and the system description is reviewed to identify minimum performance requirements for the isolation dampers. For the typical system, the drawings and description are reviewed to verify that two automatically operated isolation dampers in series separate non-essential portions and components from the essential portions.

   b. Essential portions of the SFPAVS, including the isolation dampers separating essential from non-essential portions, are classified Quality Group C or higher and seismic Category I. 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 any points of change in design classification.

   c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around fans or isolation dampers that would be required by this program.

3. The reviewer verifies that the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or in the event of certain pipe breaks or loss of offsite power. The reviewer evaluates the system, using engineering judgement and failure modes and effects analyses to determine that:

   a. 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 that house, support, or are close to essential portions of the SFPAVS, will not preclude operation of the essential portions of the SFPAVS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems.

   b. The essential portions of the SFPAVS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail in Chapter 3 of the SAR. The location and the design of the system, structures, and fan rooms (cubicles) are reviewed to determine that the degree of protection is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

   c. The total system has the capability to detect and control leakage of radioactive contamination from the system. It is acceptable if the following conditions are met:

      (1) The system PID's delineate the location of monitors and alarms in the system that are capable of detecting and warning of radiation from the fuel pool area.

      (2) The capability for isolating non-essential portions of the SFPAVS by two automatically actuated dampers in series is shown on the P&IDs.

      (3) The SFPAVS has provisions to filter radioactive contaminants from the spent fuel area by automatically isolating the normal ventilation system and actuating the emergency exhaust system before the first contaminated airborne particles and gases reache the normal ventilation exhaust ducts. A statement in the SAR that the technical specifications will require that the SFPAVS be operating whenever fuel handling operations are in progress is required.

   d. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the SFPAVS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

e. Components and subsystems necessary for preventing the release of radioactive contaminants can function as required in the event of loss of offsite power. The system design will be acceptable if the SFPAVS meets minimum system requirements as stated in the SAR assuming a failure of a single active component, within the system itself or in the auxiliary electric power source which supplies the system. The SAR is reviewed to see that for each SFPAVS component or subsystem affected by the loss of offsite power, the resulting system flow capacity will not cause the loss of air flow from areas of low potential radioactivity to areas of higher potential radioactivity. Statements in the SAR and the results of failure modes and effects analyses are considered in verifying that the system meets these requirements. This will be an acceptable verification of system functional reliability.

4. The descriptive information, P&IDs SFPAVS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system isolation or filtration requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 spent fuel pool area ventilation system (SFPAVS) includes all components and ductwork from air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents. The scope of the review of the SFPAVS for the ____________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to its safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the spent fuel pool area ventilation system and the requirements for system performance to prevent an unacceptable release of contaminants to the environment during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the spent fuel pool area ventilation system and supporting systems is in conformance with the 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 SFPAVS and its 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 SFPAVS 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 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 60, "Control of Releases of Radioactive Materials to the Environment."

5. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

6. Regulatory Guide 1.13, "Fuel Storage Facility Design Basis."

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. 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."

10. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

11. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 9.4.3 AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Effluent Treatment Systems Branch (ETSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The APCSB reviews the auxiliary and radwaste area ventilation system (ARAVS) from air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents. The review includes components such as air intakes, ducts, air conditioning units, blowers, isolation dampers, and roof exhaust fans. The review of the ARAVS covers the radwaste areas and controlled access nonradioactive areas and their relationship to safety-related areas in the auxiliary building.

1. The APCSB reviews the functional performance requirements and the air treatment equipment for the ARAVS to determine whether the ventilation system or portions of the system have been designed or need to be designed as a safety-related system. Based on this determination, the safety-related part of the system is reviewed with respect to functional performance requirements during normal operation, during adverse environmental occurrences, and during and subsequent to postulated accidents, including the loss of offsite power. The APCSB reviews safety-related portions of the system to assure that:

   a. A single active failure cannot result in loss of the system functional performance capability.

   b. Components and piping have sufficient physical separation or shielding to protect essential portions of the system from missiles and pipe whip.

   c. Failures of non-seismic Category I equipment or components will not result in unfiltered releases of radioactive contaminants.

2. The APCSB also reviews safety-related portions of the ARAVS with respect to the following:

a. The capability to detect and monitor radiation levels.

b. The capability to direct ventilation air from areas of low radioactivity to areas of progressively higher radioactivity.

The capability to detect the need for isolation and to isolate safety-related portions of the system in the event of fires, failures, or malfunctions, and the capability of the isolated system to function under such conditions.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The MEB will, upon request, review the seismic qualification of components and confirm that the components, piping, and structures are designed in accordance with applicable codes and standards. The ETSB will verify that the system functional performance meets acceptable limits for radioactivity releases during normal operation. The RAB reviews the system capability to monitor radiation levels. RAB also verifies the system meets the radiation protection criteria. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all electrical components required for proper operation.

II. <u>ACCEPTANCE CRITERIA</u>

Acceptability of the ARAVS design, 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 ARAVS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. The design of safety-related portions of the ARAVS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety.

4. General Design Criterion 60, as related to the handling of radioactive materials in the ARAVS.

5. General Design Criterion 64, as related to the monitoring of gaseous releases through the ARAVS.

6. Regulatory Guide 1.26, as related to quality group classification of systems and components.

7. Regulatory Guide 1.29, as related to seismic design classification of system components.

8. Regulatory Guide 1.52, as related to system functional performance requirements.

9. Regulatory Guide 8.8, "Information Relavent To Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

10. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 procedures for OL reviews include 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 as a result of the staff's review.

As a result of various ARAVS designs proposed by applicants, there will be variations in system requirements. For the purpose of this review plan, a typical system is assumed which has fully redundant subsystems, each having an identical essential (safety features) portion. For cases where there are variations from this typical arrangement, the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II. The reviewer will select and emphasize material from this plan as may be appropriate for a particular case.

1. The SAR is reviewed to verify that the system description section and piping and instrumentation diagrams (P&IDs) show the ARAVS equipment used for normal operation, the ambient temperature limits for the areas serviced, and the filtration capacity of the system filters. The system performance requirements section is reviewed to determine that it describes allowable component operational degradation (e.g., loss of function, damper leakage) and describes the procedures that will be followed to detect and correct these conditions. The reviewer, using results from failure modes and effects analyses as appropriate, determines that the safety-related portion of the system is capable of functioning in spite of the failure of any active component.

2. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed to determine that:

   a. Essential portions of the ARAVS are correctly identified and are isolable from non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical divisions between such portions and indicate design classification changes. System drawings are also reviewed to verify that they show the means for accomplishing isolation and the description is reviewed to identify minimum performance requirements for the isolation dampers. For the typical system, the drawings and description are reviewed to verify that two automatically operated isolation dampers in series separate non-essential portions and components from the essential portions.

b. Essential portions of the ARAVS, including the isolation dampers separating essential from non-essential portions, are classified seismic Category I and Quality Group C or higher. Component and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above seismic classification has been included, and that the P&IDs indicate any points of change in design classification.

3. The reviewer verifies that the essential portion of the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or in the event of certain pipe breaks or loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

a. 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 that house, support, or are close to essential portions of the ARAVS, will not preclude operation of the essential portions of the ARAVS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable. (CP)

b. The essential portions of the ARAVS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and the design of the system, structures, and fan rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. The total system has the capability to detect and control leakage of radioactive contamination from the system. It is acceptable if the system P&ID indicates monitors and alarms located in the system intakes that are capable of detecting radioactive leakage and provisions for automatically isolating the ARAVS before the first contaminated airborne particles and gases reach the normal ventilation exhaust ducts. Two automatically actuated dampers in series should be provided for isolation of non-essential portions of the system.

d. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the ARAVS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

e. Components and subsystems necessary for preventing the release of radioactive contaminants can function as required in the event of loss of offsite power. The system design will be acceptable if the ARAVS meets minimum system requirements as stated in the SAR assuming a failure of a single active component within the system or in the auxiliary electric power source which supplies the system. The SAR is reviewed to see that for each ARAVS component or subsystem affected by the loss of offsite power, the resulting system flow capacity will not cause the loss of preferred direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity. Statements in the SAR and the results of failure modes and effects analyses are considered in verifying that the system meets these requirements. This will be an acceptable verification of system functional reliability.

4. The descriptive information, P&IDs, ARAVS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system isolation or filtration requirements are met for each accident situation for the required time spans. For each case, the design will be acceptable if minimum system requirements are met.

IV. EVALUATION FINDINGS

The reviewer determines 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 auxiliary and radwaste area ventilation system (ARAVS) includes all components and ductwork from air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents. The scope of the review of the ARAVS for the _______________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the auxiliary supporting systems that are essential to its safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the auxiliary and radwaste area ventilation system and the requirements for system performance to preclude an unacceptable release of contaminants to the environment during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the auxiliary and radwaste area ventilation system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the ARAVS and necessary auxiliary 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 ARAVS 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 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 60, "Control of Releases of Radioactive Materials to the Environment."

5. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

7. Regulatory Guide 1.29, "Seismic Design Classification."

8. 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."

9. Regulatory Guide 8.8, "Information Relevant To Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

10. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.4.4 TURBINE AREA VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Effluent Treatment Systems Branch (ETSB)
Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The APCSB reviews the turbine area ventilation system (TAVS) from air intake to the point of discharge. The review includes components such as air intakes, ducts, cooling units, blowers, isolation dampers, and roof exhaust fans. The review of the TAVS includes systems contained in the turbine building and their relationship, if any, to safety-related equipment areas.

1. The APCSB reviews the functional performance requirements and the methods and equipment provided for air treatment for the TAVS to determine whether the ventilation system or portions of the system have been designed or need to be designed as a safety system. In making this determination, systems provided for heating, ventilating, and air conditioning of the turbine area, designed to normal industrial standards and those systems that provide for control and filtration of small quantities of radioactive gas leakage in the turbine area during normal plant operation, are not considered safety-related for the purpose of this plan. Based on this determination, any safety-related portions of the system are reviewed with respect to funtional performance requirements during adverse environmental occurrences, during normal operation, and subsequent to postulated accidents, including the loss of offsite power. The APCSB reviews the safety-related portions of the system to assure that:

   a. A single active failure cannot result in loss of the system functional performance capability.

   b. Components and piping have sufficient physical separation or barriers to protect essential portions of the system from missiles and pipe whip.

   c. Failures of non-seismic Category I equipment or components will not result in an unacceptable release of radioactive contaminants.

2. The APCSB also reviews safety-related portions of the TAVS with respect to the following:

   a. Provisions to detect and monitor radiation levels.

   b. The capability of the system to direct ventilation air from areas of low radioactivity to areas of higher radioactivity levels.

   c. The capability to detect the need for isolation and to isolate safety-related portions of the system in the event of fires, failures, or malfunctions, and the capability of the isolated system to function under these conditions.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The MEB will, upon request, review the seismic qualification of components and confirm that the components, piping, and structures are designed in accordance with applicable codes and standards. The ETSB will verify that the system functional performance meets acceptable limits for radioactive releases during normal operations. The EICSB will, upon request, determine the adequacy of the design, installation, inspection, and testing of all electrical components required for proper operation. The RAB reviews the system capability to monitor radiation levels. RAB also verifies the system meets the radiation protection criteria.

II. ACCEPTANCE CRITERIA

Acceptability of the TAVS design, 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 TAVS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. The design of safety-related portions of the TAVS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety.

4. General Design Criterion 60, as related to the handling of radioactive materials in the TAVS.

5. General Design Criterion 64, as related to the monitoring of gaseous releases through the TAVS.

6. Regulatory Guide 1.26, as related to the quality group classification of systems and components.

7. Regulatory Guide 1.29, as related to seismic design classification of systems and components.

8. Regulatory Guide 1.52, as related to system functional performance requirements.

9. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

10. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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 procedures for OL reviews include a determination that the proposed technical specifications are in agreement with the requirements for testing, minimum performance, and surveillance developed by the staff.

As a result of various TAVS designs proposed by applicants, there will be variations in system requirements. For the purpose of this review plan, a typical system is assumed which has fully redundant subsystems, each having an identical essential (safety-related) portion. For cases where there are variations from this typical arrangement, the reviewer adjusts the review procedures given below. However, in such cases, the system design must still meet the acceptance criteria given in Section II. The reviewer selects and emphasizes material from this plan as may be appropriate for a particular case.

1. The SAR is reviewed to verify that the system description section and piping and instrumentation diagrams (P&IDs) show the TAVS equipment used for normal operation, the ambient temperature limits for the areas serviced, and the filtration capacity of the exhaust filters. The system performance requirements are reviewed to determine the allowable component operational degradation (e.g., loss of function, damper leakage) and the procedures that will be followed to detect and correct these conditions. The reviewer, using results from failure modes and effects analyses as appropriate, determines that the system is capable of sustaining the failure of any active component that is required for the prevention of unacceptable releases of radioactive contaminants to the environment.

2. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed to determine that:

a. Essential portions of the TAVS are correctly identified and are isolable from non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical divisions between each portion and indicate the changes in design classification. System drawings are also reviewed to verify the means provided for accomplishing isolation and to identify minimum performance requirements for the isolation dampers. For the typical system, the drawings and descriptions are reviewed to verify that two automatically operated isolation dampers in series separate non-essential portions and components from the essential portions.

b. Essential portions of the TAVS, including the isolation dampers separating essential from non-essential portions, are classified Quality Group C or higher and seismic Category I. Component and system descriptions in the SAR that identify mechanical and performance characteristics are reviewed to verify that the above seismic classifications have been included, and that the P&IDs indicate any points of change in design classification.

3. The reviewer verifies that the safety-related portion of the system has been designed so that system function will be maintained as required, in the event of adverse environmental phenomena or in the event of certain pipe breaks or loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

   a. 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 that house, support, or are close to essential portions of the TAVS, will not preclude operation of the essential portions of the TAVS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. A commitment in the SAR confirming that the above conditions are met is acceptable. (CP)

   b. The essential portions of the TAVS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Seismic design, flood protection, and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and design of the system, structures, and fan rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A commitment in the SAR to the effect that the system is located in a seismic Catogory I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles, is acceptable.

c. The system has the capability to detect and control unacceptable levels of leakage of radioactive contamination from the system. The reviewer verifies that the following conditions are met:

(1) The system P&ID shows monitors and alarms located in the system discharge.

(2) The capability for isolation of the TAVS by two automatically actuated dampers in series is shown on the P&IDs.

(3) The system is capable of processing normal releases in a controlled manner.

d. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the TAVS or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

e. Components and subsystems necessary for preventing unacceptable releases of radioactive contaminants can function as required in the event of loss of off-site power. The system design will be acceptable if the TAVS meets minimum system requirements as stated in the SAR assuming a failure of a single active component, within the system itself, or in the auxiliary electric power source which supplies the system. The SAR is reviewed to see that, for each TAVS component or subsystem affected by loss of offsite power, the resulting system flow capacity will not cause the loss of direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity. Statements in the SAR and the results of failure modes and effects analyses are considered in verifying that the system meets these requirements. This will be an acceptable verification of system functional reliability.

4. The descriptive information, P&IDs, TAVS drawings, and failure modes effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system isolation or filtration requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 turbine area ventilation system (TAVS) includes all components and ducting from air intake to the point of discharge. The scope of review of the TAVS for the ___________________________ plant included layout drawings, piping and instrumentation diagrams and descriptive information for safety-related portions of the system and the auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the turbine area ventilation system and the requirements (if any) for system performance to preclude any adverse effect on safety-related functions during all conditions of plant operation. (CP)] [The review has determined that the design of the turbine area ventilation system and auxiliary 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 and design criteria for the TAVS and necessary auxiliary 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 TAVS 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 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 60, "Control of Releases of Radioactive Materials to the Environment."

5. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

6. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

7. Regulatory Guide, 1.29, "Seismic Design Classification."

8. Regulatory Guide, 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Light-Water-Cooled Nuclear Power Plants."

9. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable (Nuclear Reactors)."

10. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.4.5 ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents.

The APCSB reviews the ESFVS from air intake to the point of discharge to the atmosphere. The review includes components such as air intakes, ducts, air conditioning units, flow control devices, isolation dampers, exhaust vents, and exhaust fans.

The review of the ESFVS covers all ventilation systems utilized to maintain a controlled environment in areas containing safety-related equipment. These include the service water pump house, diesel generator area, emergency core cooling system (ECCS) pump rooms, component cooling water pump room, auxiliary feedwater pump area, and other areas containing equipment essential for the safe shutdown of the reactor or necessary to prevent or mitigate the consequences of an accident.

1. The APCSB reviews the ESFVS to determine the safety significance of the various portions and subsystems. Based on this determination, the safety-related portions of the system are reviewed with respect to functional performance requirements associated with engineered safety feature areas during normal operation, during adverse environmental occurrences, and during and subsequent to postulated accidents, including the loss of offsite power. The APCSB reviews safety-related portions of the system to assure that:

   a. A single active failure cannot result in loss of the system functional performance capabilities.

b. Components and piping or ducting have sufficient physical separation or barriers to protect essential portions of the system from missiles and pipe whip.

c. Failures of non-seismic Category I equipment or components will not result in damage to essential portions of the ESFVS.

2. The APCSB also reviews safety-related portions of the ESFVS with respect to the following:

a. The ability of the heating and cooling systems to maintain a suitable ambient temperature range in the areas serviced, assuming proper performance of equipment contained in these areas.

b. Capability to detect leakage of radioactivity or airborne contaminants from the engineered safety feature areas, and the ability to isolate the system to prevent uncontrolled discharge to the environment.

c. Provisions to detect the need for isolation and to isolate portions of the system in the event of failures or malfunctions.

d. The ability of the safety features equipment in the areas being serviced by the ventilation system to function under the worst anticipated degraded ESFVS system performance.

e. Capability of the system to circulate sufficient air to prevent accumulation of inflammable or explosive gas or fuel-vapor mixtures from components such as storage batteries and stored fuel.

f. The capability of the system to automatically actuate components not operating during normal conditions, or to actuate standby components (redundant equipment) in the event of a failure or malfunction, as needed.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results are used by the APCSB to complete the overall evaluation of the system. The SEB determines the acceptability of design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification of components and confirms that components, piping, and structures 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 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 EICSB determines the adequacy of the design, installation, inspection, and testing of all electrical components required for proper operation.

II. ACCEPTANCE CRITERIA

Acceptability of the ESFVS design, 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 ESFVS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of safety-related portions of the ESFVS is acceptable if the integrated design of the systems 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety.

4. Regulatory Guide 1.26, as related to the quality group classification of system components.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Regulatory Guide 1.52, as related to system design requirements, maximum system flow requirements, and system functional performance requirements.

7. Branch Technical Positions APCSB 3-1, and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 procedures for OL reviews include 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 as a result of the staff's review.

As a result of various ESFVS designs proposed by applicants, there will be variations in system requirements. For the purpose of this review plan, a typical system is assumed which has fully redundant subsystems, each having an identical essential (safety features) portion. For cases where there are variations from this typical arrangement, the reviewer would adjust the review procedures given below. However, the system design would be required to meet the acceptance criteria given in Section II. The reviewer will select and emphasize material from this review plan as may be appropriate for a particular case.

1. The SAR is reviewed to verify that the system description section and piping and instrumentation diagrams (P&IDs) show the ESFVS equipment used for normal operation, the ambient temperature limits for the areas serviced, and the filtration capacity of the intake and exhaust filters. The system performance requirements section is reviewed to determine that it limits allowable component operational degradation (e.g., loss of function, damper leakage) and describes the procedures that will be followed to detect and correct these conditions. The reviewer, using results from failure modes and effects analyses as appropriate, will determine that the safety-related portion of the system is capable of sustaining the failure of any active component.

2. The system P&IDs, layout drawings, and component descriptions and characteristics are then reviewed to determine that:

    a. Essential portions of the ESFVS are correctly identified and are isolable from non-essential portions of the system. The P&IDs are reviewed to verify that they clearly indicate the physical divisions between such portions and indicate design classification changes. System drawings are also 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 dampers. For the typical system, the drawings and description are reviewed to verify that two automatically operated isolation dampers in series separate non-essential portions and components from the essential portions.

    b. Essential portions of the ESFVS, including the isolation dampers separating essential from non-essential portions, are classified Quality Group C or higher and seismic Category I. 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 change in design classification.

    c. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary test recirculation loops around fans or isolation dampers that would be required by this program.

3. The reviewer verifies that the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or in the event of certain pipe breaks or loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

a. The failure of non-essential portions of the system or of other non-seismic systems, components or structures located close to essential portions of the system will not preclude operation of the essential portions of the ESFVS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems.

b. The essential portions of the ESFVS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail in Chapter 3 of the SAR. The location and the design of the system, structures, and fan rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. The total system has the capability to detect and control leakage of airborne contamination from the system. It is acceptable if the following conditions are met:

(1) The system P&ID shows monitors and alarms located in the system that are capable of detecting and warning of radioactive contaminants. Smoke detection may be required in special cases, such as for fires resulting from aircraft crashes.

(2) The capability for isolating non-essential portions of the ESFVS by two automatically actuated dampers in series is shown on the P&IDs.

(3) The ESFVS has provisions to actuate ventilation equipment in the engineered safety feature areas before ambient temperatures exceed design rated temperatures of components..

d. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the ESFVS or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6.of the SAR and procedures for reviewing this information are given in the corresponding review plans.

e. Essential components and subsystems can function as required in the event of loss of offsite power. The system design will be acceptable if the ESFVS meets minimum system requirements as stated in the SAR assuming a failure of a single active component within the system itself or in the auxiliary electric power source which supplies the system. The SAR is reviewed to see that for each ESFVS component or subsystem affected by the loss of offsite power, the resulting system performance will not affect the capability of any engineered safety feature equipment. Statements in the SAR and results of failure modes and effects analyses are considered in verifying that the system meets these requirements. This will be an acceptable verification of system functional reliability.

4. The descriptive information, P&IDs, ESFVS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system can function following design basis accidents assuming a concurrent single active failure. The reviewer evaluates the analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that the SAR contains verification that minimum system isolation or filtration requirements are met for each accident situation for the required time spans. For each case the design will be acceptable if minimum system requirements are met.

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 engineered safety feature ventilation system (ESFVS) includes all components and ducting associated with the system from air intake to the point of discharge to the atmosphere. The scope of review of the ESFVS for the ____________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to its safe operation. [The review determined the adequacy of the applicant's proposed design criteria and design bases for the engineered safety feature ventilation system and the requirements for system performance to preclude equipment malfunction in the engineered safety feature areas due to a failure of the system during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the engineered safety feature ventilation system 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 ESFVS and its 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 ESFVS 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 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. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-,Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants."

5. Regulatory Guide 1.29, "Seismic Design Classification."

6. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Light-Water-Cooled Nuclear Power Plants."

7. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.5.1 FIRE PROTECTION SYSTEM

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

The fire protection system (FPS) provides essential protection to combat fires that may occur in the plant and safety-related components and systems. It provides for warning, alarm, and the initiation of automatic and manual systems for containment or control, suppression, and extinguishing of fires.

The APCSB review of the FPS includes an evaluation of the fire potential as described in the applicant's safety analysis report (SAR), and a review of the design layout of the FPS showing the system characteristics and locations which define the "fire prevention" and "fire protection" portions of the system. The review also includes the description, identification, and types of fire hazards that can exist and fire risk evaluations for each of the postulated hazards.

1. The APCSB reviews the total integrated FPS and its subsystems with regard to the following:

   a. The selection of fire fighting methods, manual or automatic equipment, and safety devices, including the detection, suppression, control, and extinguishing systems as described in the SAR.

   b. The selection of appraisal and trend evaluation systems to be used, and the design of the fire detection and alarm system.

   c. The general plans for performing inspection checks and the frequency of testing to maintain a reliable detection and alarm system.

d. Offsite fire station provisions and plans to provide assistance to the plant as necessary.

2. The APCSB reviews the building and facility arrangements and structural design features which control selection of the methods for fire prevention, control and extinguishing, and control of fire hazards. Fire barriers, use of fire retardant materials, egress routes, fire walls, and the isolation and containment features provided are included.

3. The APCSB determines from the SAR if appreciable amounts of combustibles are to be located on site, and reviews analyses of the effects of these hazards on safety-related equipment located nearby. APCSB verifies that these analyses include a conservative selection of design basis fires, as determined from the quantities of stored combustible materials.

4. The functional performance of extinguishing systems, as described in the SAR, is reviewed to verify the adequacy of the FPS to protect electrical equipment. On multiple unit applications, the additional fire protection and control provisions during construction of the remaining units are reviewed to verify that the integrity of the fire protection system is maintained.

5. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The RSB will identify essential facilities and systems associated with the reactor that are required to operate during both normal and accident conditions, and determine the appropriate seismic and quality classification for system components. The SEB will verify the acceptability of the design analyses, procedures, and criteria used for seismic Category I supporting structures for the FPS. The MEB will review the seismic qualification of components and confirm that system components, piping, and structures are designed in accordance with applicable codes and standards. The EICSB will evaluate the consequences of failure of the FPS on safety-related electrical equipment and cables and provide consultation on matters concerning the adequacy of instrumentation.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the fire protection system, as described in the SAR, including appropriate sections of Chapters 2, 3, 6, 7, and 9, is based on specific general design criteria and regulatory guides, the use of consensus safety standards and engineering codes applicable to the FPS, and industrial standards and practices with respect to system functions and component selection. An additional basis for determining the acceptability of the FPS will be the compatability of design with the performance requirements.

The design of the fire protection system 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.

2. General Design Criterion 3, as related to the design and operation of prevention, protection, and detection systems provided to protect the safety-related structures, systems, and components of the reactor facility.

3. General Design Criterion 5, as related to the capability of shared systems and components to perform required functions.

4. Regulatory Guide 1.22, as related to the FPS detection and actuation devices.

5. Regulatory Guide 1.29, as related to the system seismic design classification.

6. Regulatory Guide 1.58, as related to personnel qualifications for inspection and testing of the FPS.

7. Regulatory Guide 1.78, as related to habitable areas such as the control room and the use of specific extinguishing agents.

8. American National Standards Institute (ANSI) consensus standards, Underwriters Laboratories (UL) ratings, and National Fire Prevention Association (NFPA) consensus codes used to evaluate the selection of components and the system design and functions (Refs. 8 and 9).

III. REVIEW PROCEDURES

The 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 procedures for OL applications include 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 as a result of the staff's review.

Since more than one designer will be responsible for the design and selection of the FPS, plant-to-plant variations in the design of the FPS and the selection of components will occur. The reviewer will select and emphasize material from the paragraphs below, appropriate to the particular design under review.

1. The information in the SAR is reviewed to determine if it adequately describes the design and operational performance for the total system and subsystems. The review includes system degradation evaluations and the procedures that are followed to detect and correct conditions when such degradation becomes intolerable. The results of a failure mode and effects analysis will be used to determine the capability of the total system and the subsystems of the FPS.

2. The performance specifications for the design of the FPS are reviewed to determine the acceptability of material and component selections, and detection and alarm devices. The review will verify that applicable consensus standards, engineering codes, and reference information have been used to select and develop the FPS.

3. The SAR analysis of the fire potential and the hazard of fires is reviewed to determine that:

   a. Potential fire characteristics for all individual plant areas containing combustible materials have been described and the design basis fires are in accordance with NFPA requirements. This includes maximum fire loading, hazards of flame spread, smoke generation, toxic contaminants, and explosions.

   b. Design characteristics for the suppression systems for smoke, heat, flame control, combustible and explosive gas control, and toxic material and contamination control are acceptable and provide adequate protection for safety-related structures, systems, and components. The reviewer determines that ventilating and exhaust system operations are consistent with these considerations.

   c. Performance requirements of detection systems, alarm systems, automatic suppression systems, manual systems, chemical systems, and gaseous systems for fire detection, confinement, control, and extinguishing are consistent with codes and standards requirements.

   d. Features of facility arrangements and buildings, and structural and containment features which affect the methods used for fire prevention, fire control, and control of hazards are acceptable for the protection of safety-related equipment.

   e. The essential electric circuit integrity needed to mitigate unacceptable consequences of fires corresponds with the standards established by the EICSB, and the protection systems are capable of maintaining the required integrity. Reference to established tests performed by standards organizations will be utilized during the review to determine acceptability.

   f. For multiple unit sites, protection is provided to operating units during concurrent construction of other units. This includes an evaluation of the total fire protection system for each plant or a total overall system for the site. The reviewer determines that the proposed methods for compliance with this arrangement are acceptable.

   g. The testing and inspection proposed during construction, installation and operation stages of the FPS will demonstrate the system is consistent and in compliance with code and standards appropriate for the safety function to be performed.

   h. A program is established for training, updating, and maintaining competence of the appointed station fire fighting staff, and is consistent with appropriate codes and standards.

i. Offsite fire control provisions are readily available if called upon for assistance.

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 fire protection system includes all piping, pumps, valves, manual and automatic controls, and safety devices associated with the system. The scope of the review of the fire protection system for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to its safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the fire protection system, and the requirements for system performance during all conditions of plant operation. (CP)] [The review has determined that the design of the fire protection system and supporting systems is in conformance with the 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 fire protection system and its 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 fire protection system 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 Phenomena."

2. 10 CFR Part 50, Appendix A, General Design Criterion 3, "Fire Protection."

3. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

4. Regulatory Guide 1.22, "Periodic Testing of Protection System Functions."

5. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

6. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

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

8. ANSI consensus standards, as applicable, A series, B series, E series, K series, and N series, American National Standards Institute.

9. Engineering society standards, codes, or guides as applicable, from, but not limited to the following: American Nuclear Society (ANS), American Society of Mechanical Engineers (ASME), American Society for Testing and Materials (ASTM), Institute of Electronics and Electrical Engineers (IEEE), National Fire Protection Association (NFPA), National Electrical Manufacturers Association (NEMA), Automatic Fire Alarm Association (AFAA), Underwriters Laboratories (UL), and American Water Works Association AWWA).

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SECTION 9.5.2 COMMUNICATIONS SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Industrial Security and Emergency Planning Branch (ISEPB)

I. AREAS OF REVIEW

The APCSB review of the communication system is limited to that portion of the system used in intra-plant and plant-to-offsite communications during accident conditions. The system is reviewed with respect to the following considerations: capability of the system to provide effective intra-plant communications and effective plant-to-offsite communications during accident conditions, including loss of offsite power.

The review of the communication system involves secondary review evaluations performed by other branches. The conclusions from their evaluations are used by the APCSB to complete the overall evaluation of the system. The evaluations provided by other branches are as follows. The ISEPB verifies that the offsite communication system provided will satisfy emergency plan requirements, including notification of personnel and implementation of evacuation procedures (Standard Review Plan 13.3). The EICSB will determine, upon request, the adequacy of the communication system with respect to its dependency upon a reliable power source during various operating conditions.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the communication system, as described in the applicant's safety analysis report (SAR), is based on the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. There are no general design criteria or regulatory guides that directly apply to the safety-related performance requirements for the communication system. The APCSB will use the following criterion to assess the system design capability: the communication system is acceptable if the integrated design of the system will provide effective communication between plant personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels.

III. REVIEW PROCEDURES

The information provided in the SAR pertaining to the design of the communication system will be evaluated to determine that intra-plant communication equipment needed in vital areas during recovery actions from transient or accident conditions is provided. The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

The design basis, design criteria, system description sections, and the analyses that demonstrate the effectiveness of the system when maximum plant noise levels are being generated during incident and accident conditions are reviewed to verify that the communication system will function effectively. The reviewer uses engineering judgment and compares the system capabilities with equipment provided for previously approved plants. The APCSB will accept the communication system if a statement in the SAR commits the applicant to perform a functional test under conditions that simulate the maximum plant noise levels being generated during the various operating conditions, including the accident condition, to demonstrate system capabilities.

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 communication system includes all components for intra-plant and plant-to-offsite communications. The scope of review of the communications system for the ___________ plant included verification that offsite equipment is capable of providing for notification of personnel and implementation of evacuation procedures, and verification that onsite communications are adequate in the event of an emergency. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the communication system and the requirements for all conditions of plant operation. (CP)] [The review has determined that the design of the communications system and auxiliary supporting systems is in conformance with the 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 communications system and necessary auxiliary supporting systems to staff positions and industry standards, and the ability of the systems to provide effective communications between plant personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels.

"The staff concludes that the design of the communications system conforms to all applicable staff positions and industry standards and is acceptable."

V. REFERENCES

1. None.

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SECTION 9.5.3 LIGHTING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The APCSB review of the lighting system is limited to the emergency or supplementary lighting systems. The system is reviewed with respect to the following considerations: capability of the system to provide adequate emergency lighting during all operating conditions, including transients and accident conditions, and the effect of the loss of offsite power on the emergency lighting system.

The review of the lighting system involves secondary review evaluations performed by other branches. The conclusions from their evaluations will be used by the APCSB to complete the overall evaluation of the system. The evaluations provided by the other branches are as follows. The EICSB will assure that the lighting system is capable of being powered by the onsite emergency power system discussed in Standard Review Plan 8.3.1.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the lighting system, as described in the applicant's safety analysis report (SAR), is based on the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. There are no general design criteria or regulatory guides that directly apply to the safety-related performance requirements for the lighting system. The APCSB will use the following criterion to assess the system design capability: the emergency lighting system is acceptable if the integrated design of the system will provide adequate emergency station lighting in all areas required for control and maintenance of safety-related equipment and the access routes to and from these areas.

III. REVIEW PROCEDURES

The information provided in the SAR pertaining to the design of the emergency lighting system is evaluated to determine that the lighting in vital areas and essential passageways to and from these areas is adequate. Engineering judgment, in conjunction with a comparison to equipment provided on previously approved plants, is used as a basis for determining acceptability.

IV. EVALUATION FINDINGS

The reviewer determines 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 lighting system includes all components necessary to provide adequate lighting during both emergency and normal operating conditions. The scope of review of the lighting system for the ______________ plant included assessment of the adequacy of the emergency power sources and verification of adequacy in accident conditions. [The review has determined the adequacy of the applicant's proposed design criteria and design bases regarding the requirements for lighting during accident conditions. (CP)] [The review has determined that the design of the emergency lighting system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the emergency lighting system and necessary auxiliary supporting systems to staff positions and industry standards.

"The staff concludes that the design of the lighting system conforms to all applicable staff positions and industry standards, and is acceptable."

V. REFERENCES

1. None.

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SECTION 9.5.4 EMERGENCY DIESEL ENGINE FUEL OIL STORAGE AND TRANSFER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

Nuclear power plants are required to have redundant onsite emergency power sources of sufficient capacity to power safety-related equipment. In almost all cases, the onsite power sources include diesel engine-driven generator sets. The standard review plans (SRP) numbered 9.5.4 through 9.5.8 cover the review of various essential elements of the emergency diesel engine sets. This plan, SRP 9.5.4, deals with the fuel oil storage and transfer system for these diesel engines up to the engine housing.

The APSCB review of the emergency diesel engine fuel oil storage and transfer system includes all piping up to the connection to the engine, the fuel oil storage tanks, the fuel oil transfer pumps, and the tank storage vaults. In addition, the review includes the quality and the quantity of fuel oil stored on site, and the availability and procurement of additional fuel from offsite sources.

1. The diesel engine fuel oil storage and transfer system is reviewed to determine that:

   a. The system meets appropriate seismic design requirements.

   b. The system will be designed, fabricated, erected, and tested to acceptable quality standards.

   c. Sufficient space has been provided to permit inspection, cleaning, maintenance, and repair of the system.

   d. A minimum of seven days supply of fuel oil had been provided onsite to meet the engineered safety feature load requirements following a loss of offsite power and a design basis accident.

e. Adequate and acceptable sources of fuel oil are available, including the means of transporting and recharging the fuel storage tank, following a design basis accident (DBA) so as to enable the diesel engines to supply uninterrupted emergency power for as long as may be required.

f. Seismic Category I structures housing the system protect it from natural phenomena and external missiles.

2. The APCSB verifies that suitable precautions will be taken to prevent deleterious material from degrading the stored fuel and that periodic tests will be performed to verify that fuel degradation does not proceed to the point where engine performance is affected.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

The review of the diesel engine fuel oil storage and transfer system will involve secondary review evaluations performed by other branches. Their evaluations are used by the APCSB to complete the overall system evaluation. The evaluations performed by other branches are as follows. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all electrical components (sensing, control and power) required for reliable operation. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of structures to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic design qualification of components and confirm that components, piping, and structures are designed in accordance with applicable codes and standards. The RSB will determine that the assigned seismic and quality group classifications for system components are acceptable. The MTEB will verify that inservice inspection requirements are met for system components and upon request will verify the compatibility of the materials of construction with service conditions.

II. ACCEPTANCE CRITERIA

Acceptability of the diesel engine fuel oil storage and transfer system, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides and on the proposed standard ANSI N195, "Fuel Oil Systems for Standby Diesel Generators." The review will also utilize information obtained from other federal agencies and reports, industry standards, military specifications, available technical literature, and operational performance data obtained from similarly designed systems at other plants having satisfactory operational experience.

The design of the diesel engine fuel oil storage and transfer system is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to the ability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and component important to safety to perform required safety functions.

4. Regulatory Guide 1.26, as related to quality group classification of the system components.

5. Regulatory Guide 1.29, as related to the seismic design classification of the system.

6. ANSI proposed standard N195, "Fuel Oil Systems for Standby Diesel Generators."

7. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design meet the acceptance criteria given in Section II of this plan. For the review of 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. The OL review includes verification that the content and intent of the technical specifications prepared by the applicant are in agreement with requirements for system testing, minimum performance, and surveillance developed as a result of the staff's review.

Plant-to-plant variations in the design of fuel oil storage and transfer systems will occur due to the number of architect-engineering companies having design responsibility in this area. Differences may occur in the number of redundant systems, in piping interconnections between diesel engines, and in sharing requirements between units. The reviewer will select and emphasize material from the paragraphs below to fit the particular design under review.

1. The SAR is reviewed to verify that the diesel engine fuel oil storage and transfer system description and related diagrams clearly indicate all modes of system operation, including the means for indicating, controlling, and monitoring fuel oil level, temperature, and pressure as required for uninterrupted operation.

2. The reviewer verifies that the system is designed to withstand the effects of seismic events, other design basis, natural phenomena, and internally and externally generated missiles. The review of internally generated missiles will consider the relative locations and orientations of components as placed in the facility.

3. Piping and interconnections between systems are reviewed to verify that single active failures will not cause unacceptable results. The associated drawings are examined to ascertain that sufficient space has been provided around the components to permit inspection, cleaning, maintenance, and repair.

4. The reviewer verifies that the design is such as to minimize the chance of deleterious material entering the system during recharging, or by operator error, or due to natural phenomena. The reviewer will ascertain that provisions or a program have been incorporated to assure that the quality of the stored fuel oil meets minimum requirements at all times.

5. The descriptive information and drawings in the SAR are reviewed to verify that:

   a. Each storage tank is equipped with an outside vent line, located so as to minimize the chance of damage, and with the vent point higher than the PMF flood level and the storage tank fill pipe opening.

   b. The minimum onsite inventory of fuel oil is sufficient to enable the diesel generators to power required engineered safety features for a period of seven days following any design basis accident and loss of offsite power.

   c. The day or integral tank associated with each diesel generator set is located at an elevation to assure a slight positive pressure at the engine fuel pumps.

   d. An overflow line is provided to return excess fuel oil delivered by the transfer pump back to the fuel oil storage tank.

   e. A low level alarm is provided to enable the operator to accomplish minor repairs or maintenance before all fuel in the day or integral tank is consumed (assuming full power operation).

6. The reviewer verifies that suitable precautions will be taken, once the fuel oil tank has been filled, to exclude sources of ignition such as open flames or hot surfaces, and that protective measures such as compartmentalization of redundant elements are used to minimize the potential causes and consequences of fires and explosions.

7. The reviewer verifies that the system function will be maintained as required in the event of failure of non-seismic Category I systems or structures located near the system. Reference to the SAR sections describing site features and the general arrangement and layout drawings will be necessary in this determination. Plant arrangement features, in conjunction with the protection obtained by location and the design of the system and structures, are considered in determining the ability of the system to maintain function in the event of such failures.

8. The diesel engine fuel oil storage and transfer system is reviewed to verify that protection from the effects of breaks in high and moderate energy lines has been

provided. Layout drawings are reviewed to assure that no high or moderate energy piping systems are located close to the fuel oil system, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding review plans.

9. The descriptive information, related system drawings, and the results of failure modes and effects analyses in the SAR are reviewed to verify that minimum system requirements will be met following design basis accidents assuming a concurrent single active component failure. For each case the design will be acceptable if minimum system requirements are met.

IV. EVALUATION FINDINGS

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

"The diesel engine fuel oil storage and transfer system includes storage tanks, fill, vent, drain, and overflow return lines, fuel oil transfer pumps, strainers, filters, valves, day tanks, and all components and piping up to the connections to the engine. The scope of review of the diesel engine fuel oil storage and transfer system for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and auxiliary supporting systems essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the diesel engine fuel oil storage and transfer system, and the requirements for system performance during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the diesel engine fuel oil storage and transfer system and auxiliary 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 design criteria and bases for the diesel engine fuel oil storage and transfer system and necessary auxiliary 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 diesel fuel oil storage and transfer system conforms to all applicable regulations, guides, staff positions, and industry standards, and is acceptable."

V. REFERENCE

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. 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. ANSI proposed standard N195, "Fuel Oil Systems for Standby Diesel Generators," American National Standards Institute.

7. 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 9.5.5 EMERGENCY DIESEL ENGINE COOLING WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)
Materials Engineering Branch (MTEB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The emergency diesel engine cooling water system (EDECWS) provides cooling water to the station emergency diesel engines. The APCSB review includes those portions of the EDECWS that receive heat from components essential for proper operation of the diesel engines and that are housed within their respective diesel engine compartments, and those additional parts of the system that transfer the heat to a heat sink. The system includes all valves, heat exchangers, pumps and piping up to the engine housing.

1. The APCSB reviews the functional performance characteristics of the EDECWS and the effects on those characteristics of adverse environmental occurrences, abnormal operational requirements, accident conditions, and loss of offsite power.

2. The system is reviewed to determine that a malfunction or single failure of a component, or the loss of a cooling source, will not reduce the safety-related functional performance capabilities of the system. The APCSB verifies that:

   a. System components and piping have sufficient physical separation or shielding to protect the system from internally or externally generated missiles and from pipe whip and jet impingement caused by cracks or breaks in high and moderate energy piping.

   b. System components are designed in accordance with the design codes required by the assigned quality group and seismic category classifications.

   c. The system is housed in structures designed to seismic Category I requirements.

   d. Failures of non-seismic Category I structures and components would not affect the safety-related functions of the EDECWS.

3. The APCSB reviews the design of the EDECWS with respect to the following:

   a. Functional capability during periods of abnormally high water levels (the probable maximum flood).

   b. Capability to detect and control system leakage, including isolating portions of the system in the event of excessive leakage or component malfunction.

   c. Measures to preclude long-term corrosion and organic fouling that would degrade system cooling performance, and the compatibility of any corrosion inhibitors or antifreeze compounds used with the materials of the system.

   d. The capacity of the EDECWS with regard to the manufacturer's recommended engine temperature differentials under adverse operating conditions.

   e. Provision of proper instruments and testing systems to permit operational testing of the system.

   f. Provisions to assure that normal protective interlocks do not preclude engine operation during emergency conditions.

4. The APCSB will review the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this plan.

Secondary reviews will be performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of the Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as a safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification testing of components and will determine that components, piping, and structures are designed in accordance with applicable codes and standards. 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 RSB will determine that the seismic and quality group classifications for system components are acceptable. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all electrical components (sensing, control, and power) required for proper operation of the system, including interlocks (EICSB BTP-17).

II. ACCEPTANCE CRITERIA

Acceptability of the diesel engine cooling system design, 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 system will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. Listed below are the specific criteria as they relate to the EDECWS.

The system is acceptable if the 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety being capable of performing required safety functions.

4. General Design Criterion 44, to assure:

    a. The capability to transfer heat from systems and components to a heat sink under transient or accident conditions.

    b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure.

    c. The capability to isolate components of the system or piping, if required to maintain the system safety function.

5. General Design Criterion 45, as related to design provisions to permit periodic inspection of safety-related components and equipment of the system.

6. General Design Criterion 46, as related to design provisions to permit appropriate functional testing of safety-related systems or components to assure structural integrity and leaktightness, operability and performance of active components, and the capability of the system to function as intended under accident conditions.

7. Regulatory Guide 1.26, as related to the quality group classification of system components.

8. Regulatory Guide 1.29, as related to the seismic design classification of system components.

9. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

10. Branch Technical Position EICSB-17, diesel-generator protective trip circuit bypasses as it relates to engine cooling water protective interlocks during accident conditions.

III. REVIEW PROCEDURES

The 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 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 procedures for OL reviews include 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 as a result of the staff's review.

The design of the diesel engine cooling water system may vary considerably from plant to plant due to the requirements of various diesel engine manufacturers, the number and type of secondary cooling loops used for heat removal, and the number of intermediate cooling loops required to transfer the rejected heat to the ultimate heat sink. Variations in design may also occur due to preferences of various architect-engineer firms. Therefore, for the purpose of this review plan, a typical system is assumed. Any variance in the review procedure, to suit a particular design, will be such that the system review areas in Section I are covered, and the system will meet the criteria in Section II.

1. The SAR is reviewed to establish that the EDECWS description and related diagrams clearly delineate system operation, individual and total heat removal rates required by components, and the margin in the design heat removal rate capability. The reviewer verifies the following:

   a. Failure of a piping interconnection, as shown on system piping and instrumentation diagrams (P&IDs), between subsystems does not cause total degradation of the EDECWS. The results of failure modes and effects analyses are used as a basis of acceptance.

   b. Provisions have been made to permit inspection of components, as shown on system layout drawings.

   c. The performance and water chemistry of the EDECWS is in conformance with the engine manufacturer's recommendations.

   d. The engine "first try" starting reliability has been increased by providing an independent loop for circulating heated water while the engine is in the standby mode.

   e. Temperature sensors have been provided to alert the operator when cooling water temperatures exceed the limits recommended by the manufacturer. Protective interlocks in this system are acceptable if the SAR indicates that the interlocks are in conformance with EICSB Branch Technical Position-17.

2. The reviewer verifies that the EDECWS can be vented to assure that all spaces are filled with water. Statements in the SAR to the effect that the system design satisfies the above requirement are acceptable.

3. The reviewer verifies that system function will be maintained in the event of adverse environmental phenomena and loss of offsite power. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses to determine that:

   a. Failure of non-essential portions of the system or of other systems not designed to seismic Category I requirements and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the EDECWS, will not preclude essential functions. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR to the effect that the above conditions are met are acceptable.

   b. The essential portions of the system are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

4. The reviewer verifies that there are no high or moderate energy piping systems located close to the EDECWS or that the EDECWS is protected from the effects of postulated breaks in these systems. The means of providing such protection are given in Chapter 3 of the SAR and procedures to review the information presented are given in the standard review plans for that chapter.

5. The descriptive information, P&IDs, onsite emergency power supply drawings, and system analyses are reviewed to assure that essential portions of the system will function following design basis accidents, assuming a concurrent single active component failure. The reviewer evaluates the results of failure modes and effects analyses presented in the SAR to ensure the functioning of required portions of the system.

6. The performance requirements of the diesel engine are reviewed to determine the time available to provide cooling water to the diesels and the other systems that have to operate to assure onsite power capability.

7. The reviewer verifies that the EDECWS and the diesel generator can perform during periods when less than full electrical power generation is required.

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 emergency diesel engine cooling water system includes all piping, valves, heat exchangers, and pumps up to the points where the cooling water piping connects to the engine housings. The scope of review of the diesel engine cooling water system for the ________________ plant included layout drawings, process flow diagrams, piping and instrumentation diagrams, and descriptive information for the system and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the emergency diesel engine cooling water system, and the requirements for continuous cooling during all conditions of plant operation. (CP)] [The review has determined that the design of the diesel engine cooling water system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the diesel engine cooling water system and necessary auxiliary 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 diesel engine cooling water system 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 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 44, "Cooling Water System."

5. 10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

6. 10 CFR Part 50, Appendix A, General Design Criterion 46, "Testing of Cooling Water System."

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 Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

10. Branch Technical Position EICSB-17, "Diesel-Generator Protective Trip Circuit Bypasses."

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SECTION 9.5.6 EMERGENCY DIESEL ENGINE STARTING 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)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The APCSB review of the emergency diesel engine starting system EDESS includes those system features necessary to assure reliable starting of the emergency diesel engine following a loss of offsite power. The review includes the system air compressors, air receivers, devices to crank the diesel engine, valves, piping, filters, and associated ancillary instrumentation and control systems.

1. The APCSB reviews the EDESS to verify that:

   a. Each emergency diesel engine has reliable, redundant starting systems of adequate starting capacity.

   b. The system complies with appropriate seismic requirements and quality standards, and has been properly designed, fabricated, erected, and tested.

   c. Essential portions of the system are housed within seismic Category I structures capable of protecting the system from extreme natural phenomena, missiles, and the effects of pipe whip or jet impingement from high and moderate energy pipe breaks.

   d. A single failure in an emergency engine air starting system will not lead to a loss of function of more than one diesel engine.

2. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The evaluation performed by others 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 to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification testing of components and confirms that components, piping, and structures 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, will verify the compatibility of the materials of construction with service conditions. The EICSB determines the adequacy of the design, installation, inspection, and testing of all essential electrical components (sensing, control and power).

II. ACCEPTANCE CRITERIA

Acceptability of the diesel engine starting system, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. An additional basis for acceptability is the similarity of the EDESS design with that of previously reviewed plants having satisfactory operating experience.

The design of the EDESS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion No. 2, as related to the ability of structures housing the system to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion No. 4, with respect to structures housing the systems and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion No. 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. Regulatory Guide 1.26 as related to quality group classification of the system components.

5. Regulatory Guide No. 1.29, as related to the system seismic design classification.

6. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

7. Branch Technical Position EICSB-17, Diesel Generator Protective Trip Circuit Bypasses.

8. The EDESS should also meet the following specific criteria:

   a. Each diesel engine should be provided with an air compressor and with independent and redundant starting systems, each consisting of an air receiver, injection lines and valves, and devices to crank the engine.

b. As a minimum, each of the redundant starting systems shouuld be capable of cranking a cold diesel engine five times without recharging the receiver. Each cranking cycle duration should be approximately 3 seconds, or consist of 2 to 3 engine revolutions.

c. Alarms should be provided which alert operating personnel if the air receiver pressure falls below the minimum allowable value.

d. Provisions should be made for the periodic or automatic blowdown of accumulated moisture and foreign material in the air receivers.

II. <u>REVIEW PROCEDURES</u>

The 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 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 review procedures for OL applications include 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 during the review. The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

1. The reviewer establishes that the EDESS description and piping and instrumentation drawings (P&IDs) clearly delineate all modes of operation and include the means for monitoring, indicating, and controlling receiver air pressure as required by the engine starting service. The P&IDs are reviewed to determine that each receiver has been provided with a pressure gauge, relief valve, drain valve, an automatic means of maintaining the receiver pressure within an allowable range, and suitable low pressure alarms. If there are piping interconnections between shared systems, they are reviewed to verify that failure could not lead to the loss of starting of more than one diesel engine. The building layout drawings are examined to ascertain that sufficient space has been provided around the components to permit inspection. The reviewer verifies that essential portions of the EDESS are classified seismic Category I.

2. The SAR is reviewed to assure that each diesel engine has its own compressor and that the compressor capacity is adequate with respect to the air receiver capacities of the redundant starting systems.

3. The reviewer verifies that the system has been designed to be operated and maintained in the event of adverse environmental conditions such as hurricanes, tornadoes, or floods, and is protected against the effects of internally or externally generated missiles.

4. The reviewer determines that the failure of non-seismic Category I systems, structures, or components located close to the EDESS will not preclude operation of the system.

5. The reviewer determines that essential portions of the EDESS are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to the system, or that protection from the effects of failure are provided. The means of providing such protection are discussed in Section 3.6 of the SAR and the procedures for reviewing this information are given in the corresponding review plans.

6. The SAR information, P&IDs, related system drawings, and failure modes and effects analyses are reviewed to assure that minimum requirements of the system will be met following design bases accidents, assuming a concurrent single active failure and loss of offsite power. The analyses presented in the SAR are reviewed to assure function of required components following postulated accidents. Utilizing the descriptions, related drawings, and analyses, the reviewer verifies that minimum system requirements are met for each degraded situation over the required time spans. For each case the design is considered acceptable if minimum system requirements are met.

IV. EVALUATION FINDINGS

The reviewer verifies that the information provided and his review support conclusions of the following type, to be included in the staff's safety evaluation report:

"The emergency diesel engine starting system includes the features necessary to assure that the system will be available and capable of starting the diesel engine following a loss of offsite power. The scope of review of the system for the ______________ ______________ plant included layout drawings, flow diagrams, piping and instrumentation diagrams, and descriptive information for the emergency diesel engine starting system and supporting systems essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the system, and the provisions necessary for diesel engine starting during all conditions of plant operation. (CP)] [The review has determined that the design of the emergency diesel engine starting system and supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the review has been conformance of the applicant's designs and design criteria for the emergency diesel engine starting system and necessary 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 emergency diesel engine starting system 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 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. 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 Desing Classification," Revision 1.

6. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 9.5.7 EMERGENCY DIESEL ENGINE LUBRICATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The emergency diesel engine lubrication system (EDELS) provides essential lubrication to the components of the emergency diesel engines. The APCSB reviews the EDELS and associated auxiliary systems. The review includes system piping, pumps, components, and associated ancillary equipment essential for system operation up to the engine housing.

1. The APCSB reviews the characteristics of the EDELS and system components with respect to the effect on functional performance of adverse environmental occurrences, abnormal operational requirements, and accident conditions.

2. The APCSB determines that a malfunction or failure of a component, or the loss of a cooling source does not reduce the safety-related functional performance capabilities of the emergency powered systems. Further, the APCSB review assures that:

   a. System components and piping have sufficient physical separation or barriers to protect the system from internally and externally generated missiles.

   b. The system is protected from the effects of pipe cracks or breaks in high and moderate energy piping.

   c. System components are designed in accordance with the design codes required by the assigned quality group and seismic category classifications.

   d. The system is housed in structures designed to seismic Category I requirements.

   e. Failure of non-seismic Category I structures or components will not affect the safety-related functions of the system.

3. The APCSB will also review the design of the EDELS with respect to the following:

   a. Functional capability during abnormally high site water levels (probable maximum flood).

   b. Capability for detection and control of system leakage.

   c. Measures to assure the quality of the lubricating oil.

   d. Capability for isolating portions of the system in the event of excessive leakage or component malfunction.

   e. Instrumentation and control features provided to permit operational testing of the system and to assure that normal protective interlocks do not preclude engine operation during emergency conditions.

4. The APCSB will review the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this plan.

Secondary reviews will be performed by other branches and the results of their reviews will be used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification testing of components and will determine that the components, piping, and structures are designed in accordance with applicable codes and standards. The RSB will determine that the seismic and quality group classifications for system components are acceptable. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will determine the adequacy of the design, installation, inspection, and testing of all electrical components (sensing, control, and power) required for proper operation of the system.

II. <u>ACCEPTANCE CRITERIA</u>

Acceptability of the emergency diesel engine lubrication system, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. The reviewer will also utilize information obtained from other sources such as other federal agencies, published reports, industry standards, military specifications, and technical literature on commercially available products. An additional basis for the acceptability of the system will be the degree of similarity with systems in previously reviewed plants with satisfactory operating experience.

The design of the EDELS is acceptable if the integrated design of the system 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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. Regulatory Guide 1.26, as related to quality group classification of the system components.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping outside containment.

7. Branch Technical Position EI&CSB-17 Diesel-Generator Protective Trip Circuit Bypasses.

8. Specific design criteria as follows:

    a. The operating pressure, temperature differentials, flow rate, and heat removal rate of the system external to the engine are in accordance with recommendations of the engine manufacturer.

    b. The system has been provided with sufficient protective measures to maintain the required quality of the oil during engine operation.

    c. Protective measures (such as relief ports) have been taken to prevent unacceptable crankcase explosions and to mitigate the consequences of such an event.

    d. The temperature of the lubricating oil is automatically maintained above a minimum value by means of an independent recirculation loop including its own pump and heater, to enhance the "first try" starting reliability of the engine in the standby condition.

III. REVIEW PROCEDURES

The 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 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 as a result of the staff's review.

The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. The SAR is reviewed to establish that the EDELS description and related diagrams clearly delineate system operation, including the means provided for indicating and monitoring oil levels, temperatures, and pressures required for continuous operation of the system. The reviewer verifies the following:

   a. Failure of a piping interconnection, as shown on the system piping and instrumentation diagrams (P&IDs) between subsystems will not cause total degradation of the lube oil system function. The results of failure modes and effects analyses will be used in this determination.

   b. The system layout drawings are examined to ascertain that sufficient space has been provided to permit inspection of components.

   c. The system has been designed to preclude the entry of deleterious material into the system due to operator error or extreme natural phenomena during recharging or normal operation. The system is acceptable if it is shown in the SAR that the system is locked closed, or if entry is administratively controlled.

   d. The design contains an independent circulation loop to maintain the temperature of the crankcase oil above a minimum value during the standby mode.

   e. The system P&IDs indicate the temperature, pressure, and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer.

   f. Essential portions of the EDELS are classified seismic Category I.

2. The reviewer determines that the system is designed to maintain function under adverse environmental phenomena. The reviewer, using engineering judgment and the results of failures modes and effects analyses, determines that:

   a. The failure of systems not designed to seismic Category I requirements or of non-seismic Category I structures that house, support, or are close to the EDELS, will not preclude function of the system. Chapters 2 and 3 of the SAR describe related site features and provide the general structural arrangement and layout drawings and a tabulation of seismic design classifications for the structures and systems. Statements in the SAR to the effect that the above design requirements are met are acceptable.

b. The essential portions of the system are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles.

3. The review verifies that the EDELS is protected from the effects of breaks in high and moderate energy lines. The system description in the SAR is reviewed to verify that there are no high or moderate energy piping systems close to the lube oil system, or that protection from effects of failure will be provided. The means of providing such protection are given in Chapter 3 of the SAR and procedures to review the information presented are given in the corresponding standard review plans.

4. The descriptive information, P&IDs, related system drawings, and system analyses in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents, assuming a concurrent single active component failure. The reviewer evaluates the results of failure modes and effects analyses presented in the SAR to assure function of required components, traces the availability of these components on system drawings, and checks that minimum system requirements are met for each degraded situation over required time spans. For each case, the design is acceptable if minimum system requirements are met.

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 emergency diesel engine lubrication system includes the pumps, heat exchangers, valves, piping, makeup piping, and the points of connection or interfaces with other systems. The scope of review of the emergency diesel engine lubrication system for the ________________________ plant included layout drawings, flow diagrams, 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 bases for the emergency diesel engine lubrication system and the requirements for system performance under all conditions of plant operation. (CP)] [The review has determined that the design of the emergency diesel engine lubrication system and auxiliary supporting systems is in conformance with the design criteria and bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's designs and design criteria for the emergency diesel engine lubrication system and necessary 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 emergency diesel engine lubrication system 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 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. 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. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to the Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

7. Branch Technical Position EICSB-17, "Diesel-Generator Protective Trip Circuit Bypasses."

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SECTION 9.5.8 EMERGENCY DIESEL ENGINE COMBUSTION AIR INTAKE AND EXHAUST SYSTEM

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

The emergency diesel generator combustion air intake and exhaust system (EDGCAIES) supplies combustion air of reliable quality to the diesel engines, and exhausts the products of combustion from the diesel engines to the atmosphere. The APCSB reviews the system from the outside air intake to the combustion air supply lines connected to the diesel engines, and from the exhaust connections at the diesel engines to the discharge point outside the building.

1. The APCSB reviews the EDGCAIES to verify that:

   a. The system design meets appropriate seismic design classification requirements and the components are designed, fabricated, erected, and tested to acceptable quality standards.

   b. The essential portions of the system are housed in or on a seismic Category I structure that is capable of protecting the system from extreme natural phenomena and external missiles.

   c. Each diesel engine has an independent combustion air intake and exhaust system.

   d. The consequenes of a single active failure in an engine combustion air intake or exhaust system will not lead to the loss of function of more than one diesel generator.

2. The applicant's proposed technical specifications are reviewed for operating license applications, as they relate to areas covered in this plan.

Secondary reviews will be performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The SEB will determine the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB will review the seismic qualification of components and confirm that system components, piping, and structures are designed in accordance with applicable codes and standards. The RSB will determine that the seismic and quality group classifications for system components are acceptable. The MTEB will verify that inservice inspection requirements are met for system components and, upon request, will verify the compatibility of the materials of construction with service conditions. The EICSB will verify the adequacy of the design, installation, inspection, and testing of all electrical systems (sensing, control, and power) required for proper system operation.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the emergency diesel generator combustion air intake and exhaust system, 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 EDGCAIES will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of the EDGCAIES is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to the ability of structures housing the system and system components to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, with respect to structures housing the systems and the system components being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to shared systems and components important to safety being capable of performing required safety functions.

4. Regulatory Guide 1.26, as related to quality group classification of the system components.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Each emergency diesel engine should be provided with an independent and reliable combustion air intake and exhaust system. The system should be sized and physically

arranged such that no degradation of engine function will be experienced when the diesel generator set is required to operate continuously at the maximum rated power output.

7. The combustion air intake system shall be provided with a means of reducing airborne particulate material over the entire time period that emergency power is required assuming the maximum airborne particulate concentration at the combustion air intake.

8. Suitable design precautions have been taken to preclude degradation of the diesel engine power output due to exhaust gases and other dilutents that could reduce the oxygen content below acceptable levels.

III. REVIEW PROCEDURES

The 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 review procedures for OL applications include 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 as a result of the staff's review.

The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the EDGCAIES description and related diagrams clearly delineate the system components and the modes of system operation. The reviewer verifies that essential portions of the EDGCAIES are designed to appropriate seismic and quality group classification standards.

2. The SAR is reviewed to ascertain that sufficient space has been provided around the components to permit inspection of the system components.

3. The SAR is reviewed to assure that the arrangement and location of the combustion air intake and exhaust are such that dilution or contamination of the intake air by exhaust products or other gases that may intentionally or accidentally be released on site will not preclude operation of the diesel engines at rated power output.

4. The SAR is reviewed to verify that if the intake air flow or engine exhaust is dependent upon the actuation of flow control devices (louvers, dampers), the EDGCAIES will function if there is a failure of an active component.

5. The SAR is reviewed to assure that system components exposed to atmospheric conditions (ice, snow) are protected from possible clogging during standby or operation of the system.

6. The review verifies that the system will function as required in the event of other adverse natural phenomena. The reviewer evaluates the system, using engineering judgment and failure modes and effects analyses to determine that:

   a. The failure of non-essential portions of the system or of other systems not designed to seismic Category I requirements and located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the EDGCAIES, will not preclude operation of the system. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable.

   b. The essential portions of the system are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and the design of the systems and structures are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

   c. The essential portions of the system are protected from the effects of high and moderate energy line breaks. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to the essential portions of the system, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

7. The descriptive information, P&IDs, EDGCAIES layout drawings, and failure modes and effects analyses in the SAR are reviewed to assure that functional requirements of the system will be met following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the effects of failure of components, traces the availability of redundant components on system drawings, and checks that the SAR contains verification that the system functional requirements are met.

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 emergency diesel generator combustion air intake and exhaust system (EDGCAIES) includes all components and piping of the air intake system from the atmospheric air intake to its connection to the engine and all components and piping of the exhaust system from its connection to the engine to the point where it exhausts to the atmosphere. The scope of the review of the EDGCAIES for the ______________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and auxiliary supporting systems that are essential to its safe operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the emergency diesel generator combustion air intake and exhaust system and requirements for safe operation of the system during normal, abnormal and accident conditions. (CP)] [The review has determined that the design of the emergency diesel generator combustion air intake and exhaust system and auxiliary supporting systems is in conformance with the 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 emergency diesel generator combustion air intake and exhaust system and its 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 emergency diesel generator combustion air intake and exhaust system 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 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. 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. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

7. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

8. Branch Technical Position EICSB Diesel-Generator Protective Trip Circuit Bypasses.

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SECTION 10.2 TURBINE GENERATOR

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Radiological Assessment Branch (RAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

Nuclear reactor plants include a turbine generator system (TGS) to convert the energy in steam from the nuclear steam supply system into electrical energy. The TGS consists essentially of the turbine unit and the automatic devices, alarms, and trips which control and regulate turbine action, and the generator unit and its controls. The turbine control system and the steam inlet stop and control valves, the low pressure turbine steam intercept and inlet control valves, and the extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant.

The turbine generator systems installed in nuclear plants are typically equipped with redundant overspeed protection instrumentation and controls and the main steam and reheat steam control and stop valving arrangements typically provide redundancy in the valves essential for overspeed protection. The intent of the review under this plan is to verify that such redundancy, in conjunction with inservice inspection and testing of the essential valves, makes a turbine overspeed condition above the design overspeed very unlikely. Assessment of the risk to essential plant systems and structures from potential turbine missiles is reviewed under SRP 3.5.1.3.

1. The APCSB reviews the turbine generator system and the components and subsystems normally provided with this equipment with respect to the following considerations:

   a. The general arrangement of the turbine and associated equipment with respect to safety-related structures and systems and balance of plant.

b. The types and locations of main steam stop and control valves, reheat stop and intercept valves, and associated piping arrangements.

c. The capability of the turbine generator control and overspeed protection systems to detect a turbine overspeed condition and to actuate appropriate system valves or other protective devices to preclude an overspeed condition above the design overspeed.

2. The inservice inspection and operability assurance program for valves essential for overspeed protection is reviewed.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this review plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows: the RSB determines that appropriate seismic and quality group classifications have been established for system components where appropriate. The MEB confirms that the components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components, and will verify the compatibility of the materials of construction with service conditions. The RAB determines if any radiation shielding is necessary to assure safe access to turbine equipment. The EICSB reviews the overspeed protection instrumentation and controls with respect to capabilities, reliability, and redundancy.

II. ACCEPTANCE CRITERIA

There are no general design criteria or regulatory guides that are directly applicable to the design evaluation of the turbine generator. Acceptability of the design of the turbine generator system, as described in the applicant's safety analysis report (SAR), is based on the specific criteria listed below and on the similarity of the design to that of plants previously reviewed and found acceptable.

1. A turbine control and overspeed protection system should be provided to control turbine action under all normal or abnormal operating conditions, and to assure that a full load turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown either by use of the turbine bypass system and main steam relief system or other engineered safety systems. The overspeed protection system should meet the single failure criterion.

2. Turbine main steam stop and control valves and reheat steam stop and intercept valves should be provided to protect the turbine from exceeding set speeds and to protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to assure that turbine overspeed is controlled.

within acceptable limits. The valve arrangements and valve closure times should be such that a failure of any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal.

3. Extraction steam stop valves should be provided at each extraction connection. The valves shall be capable of closing within an appropriate time limit to maintain stable turbine speeds in the event of a TGS trip signal.

4. The TGS should be provided with the capability to permit periodic testing of components important to safety while the unit is operating at rated load.

5. The inservice inspection program for main steam and reheat valves should include the following provisions:

   a. At approximately 3-1/3-year intervals, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by Section XI of the ASME Code for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled and visual and surface examinations conducted of valve seats, disks, and stems. If unacceptable flaws or excessive corrosion are found in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance.

   b. Main steam stop and control valves and reheat stop and intercept valves should be exercised at least once a week by closing each valve and observing by the valve position indicator that it moves smoothly to a fully closed position. At least once a month, this examination should be made by direct observation of the valve motion.

6. Unlimited access to all levels of the turbine area under all operating conditions should be provided. Radiation shielding should be provided as necessary to permit access.

7. Connection joints between the low pressure turbine exhaust and the main condenser should be arranged to prevent adverse effects on any safety-related equipment in the turbine room in the event of rupture (it is preferable not to locate safety-related equipment in the turbine room).

8. Branch Technical Position APCSB 3-1 should be used to determine the acceptability of the effects of postulated TGS piping failures on safety-related equipment.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in Section II of this plan. For 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 review procedures for OL applications include 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 as a result of the staff's review.

The review procedures given are for a typical turbine generator system. Any variance of the review, to take account of a proposed unique design, will be such as to assure that the system meets the criteria of Section II. The reviewer evaluates the TGS, subsystems, and components of the unit that are considered essential for the safe integrated operation of the reactor facility. The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (PID's) show the turbine generator system. The general arrangement of the TGS and associated equipment with respect to safety-related structures, systems, and components is noted.

2. The reviewer consults with the EICSB to verify the adequacy of the control and overspeed protection system and to determine that:

    a. Support systems, subsystems, control systems, and alarms and trips will function for all abnormal conditions, including a single failure of any component or subsystem, and will preclude an unsafe turbine overspeed. The indepth defense that is provided by the turbine generator protection system to preclude excessive overspeeds should be designed with diverse protection means.

    b. For normal speed-load control, the speed governor action of the electro-hydraulic control system fully cuts off steam at approximately 103 percent of rated turbine speed by closing the control, stop, and intercept valves.

    c. A mechanical overspeed trip device is provided that will actuate the control, stop, and intercept valves at approximately 111 percent of rated speed.

    d. An independent and redundant backup electrical overspeed trip circuit is provided that senses the turbine speed by magnetic pickup and closes all valves associated with speed control at approximately 112 percent of rated speed. This backup electrical overspeed trip system may utilize the same sensing techniques as the electro-hydraulic control system. However, the circuitry is reviewed to determine that the control signals from the two systems are isolated from and independent of one another.

3. The main steam stop and control and the reheat stop and intercept valving arrangements and valve closure times are reviewed to ensure that no single valve failure can disable the overspeed control function.

4. The extraction steam valving arrangements and valve closure times are reviewed to see that stable turbine operation will result after a TGS trip.

5. The capability for testing of essentail components during TGS operation is reviewed.

6. The proposed inservice inspection program for essential speed control valves is reviewed to verify that it includes the provisions of item 5 of Section II.

7. The reviewer consults with RAB with regard to expected radiation levels around the TGS and the degree of access to TGS components during operation.

8. If there are safety-related systems or portions of systems located close to the TGS, the physical layout of the system is reviewed to assure that protection has been provided from the effects of high and moderate energy TGS piping failures or failure of the connections from the low pressure turbine section of the main condenser. The means of providing such protection will be given 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 turbine generator system includes all components and equipment normally provided including turbine main steam stop and control valves and reheat steam stop and intercept valves. The scope of review of the turbine generator system for the ________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and for control and supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the turbine generator system and the requirements for safe operation of the system during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the turbine generator system and supporting systems is in conformance with the 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 turbine generator system and supporting systems to applicable staff technical positions and industry standards.

"The staff concludes that the design of the turbine generator system conforms to all applicable staff positions and industry standards, and is acceptable."

V. REFERENCES

1. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1.

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SECTION 10.2.3 TURBINE DISK INTEGRITY

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 turbine disks have large masses and rotate at relatively high speeds during normal reactor operation, failure of a disk could result in high energy missiles and excessive vibration of the turbine rotor assembly.

The following areas of the applicant's safety analysis report (SAR) relating to turbine disk integrity are reviewed:

1. Materials Selection

The high pressure turbine rotor is generally a one piece forging. Stresses at normal operating speeds are relatively low. The low pressure turbine rotor assembly usually consists of a rotor shaft with shrunk-on disks. Low pressure disk stresses are due to thermal gradients, the interference fit, and centrifugal forces. These stresses are relatively high. The low pressure turbine also operates at lower temperatures than the high pressure turbine. Thus, it is particularly important that low pressure disks be made of a tough material. The use of suitable material, adequate design, and inservice inspection can greatly reduce the probability of a turbine rotor or disk failure.

The materials properties, including descriptions of the procedures to minimize flaws and improve fracture toughness, are reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the low pressure disk materials. The materials properties such as creep, stress rupture, and toughness are reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the high pressure turbine rotor materials.

Included in this information are:

a. A discussion of the ductile-brittle transition temperatures (FATT or NDT) of the materials and the tests and standards used to determine them.

b. The Charpy V-notch test program used to establish minimum upper shelf energies of the disk materials.

c. The fracture toughness test program used to establish minimum upper shelf toughness of the disk materials.

2. Fracture Toughness

The fracture toughness of the materials and the materials tests or correlations of Charpy and tensile data to toughness properties are reviewed to establish that the rotor and disk materials exhibit adequate fracture toughness at normal operating temperature and during startup.

3. High Temperature Properties

The creep and stress-rupture properties of the high pressure rotor materials are reviewed to establish that these materials exhibit adequate high-temperature, long-term properties.

4. Preservice Inspection

The preservice inspection program information is reviewed to verify that the disk forgings are first machined with minimum excess stock prior to heat treatment, that visual and surface inspections are performed on all finished machined surfaces, and that a 100% volumetric (ultrasonic) examination is performed.

5. Turbine Disk Design

The high and low pressure turbine rotor design information, including allowable stresses, temperature distributions, and design overspeed considerations, is reviewed.

6. Inservice Inspection

Descriptions of the baseline and inservice phases of the inservice inspection program, including types of inspections, areas to be inspected, frequencies of inspection, and acceptance criteria, are reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described in Section I of this plan are as follows:

1. Materials Selection

The applicant's selection of a disk or rotor material is acceptable if in accordance with the following:

The turbine disk or rotor should be made from a material and by a process that tends to minimize flaw occurrence and maximize fracture toughness properties, such as a NiCrMoV alloy processed by vacuum melting or vacuum degassing. The material should be examined and tested to meet the following criteria:

a. Chemical analysis should be made for each forging. Elements that have a deleterious effect on toughness, such as sulfur and phosphorus, should be controlled to low levels.

b. The fracture appearance transition temperature (50% FATT) as obtained from Charpy tests performed in accordance with specification ASTM A-370 should be no higher than 0°F for low pressure disks and 50°F for high pressure rotors. Nil-ductility transition (NDT) temperature obtained in accordance with specification ASTM E-208 may be used in lieu of FATT. NDT temperatures should be no higher than -30 and 20°F, respectively.

c. The Charpy V-notch (C_V) energy at the minimum operating temperature of each low pressure disk in the tangential direction should be at least 60 ft-lbs. The C_V energy of high pressure rotor materials at minimum operating temperature should be at least 50 ft-lbs. A minimum of three C_V specimens should be tested in accordance with specification ASTM A-370.

2. Fracture Toughness

The low pressure turbine disk and high pressure rotor fracture toughness properties are acceptable if in compliance with the following criteria:

The ratio of the fracture toughness (K_{Ic}) of the disk and rotor materials to the maximum tangential stress at speeds from normal to design overspeed should be at least two $\sqrt{\text{in.}}$, at minimum operating temperature. Bore stress calculations should include components due to centrifugal loads, interference fit, and thermal gradients. Sufficient warmup time should be specified in the turbine operating instructions to assure that toughness will be adequate to prevent brittle fracture during startup. Fracture toughness properties can be obtained by any of the following methods:

a. Testing of the actual material of the turbine disk to establish the K_{Ic} value at normal operating temperature.

b. Testing of the actual material of the turbine disk with an instrumented Charpy machine and a fatigue precracked specimen to establish the K_{Ic} (dynamic) value at normal operating temperature. If this method is used, K_{Ic} (dynamic) shall be used in lieu of K_{Ic} (static) in meeting the toughness criteria above.

c. Estimating of K_{Ic} values at various temperatures from conventional Charpy and tensile data on the disk material using methods presented by J. A. Begley and W. A. Logsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1 (Ref. 5). This method of obtaining K_{Ic} should be used only on materials which exhibit a well-defined Charpy energy and fracture appearance transition curve and are strain-rate insensitive. The test data and the calculated toughness curve should be submitted to the staff for review.

d. Estimating "lower bound" values of K_{Ic} at various temperatures using the equivalent energy concept of F. J. Witt and T. R. Mager, ORNL-TM-3894 (Ref. 6). Load-displacement data from the compact tension specimens and the calculated toughness data should be submitted to the staff for review.

3. High Temperature Properties

The stress-rupture properties of the high pressure rotor material are acceptable if they provide sufficient assurance of rotor integrity for the lifetime of the turbine. The applicant can demonstrate compliance by submitting stress-rupture data of an equivalent material with similar properties to the staff for evaluation.

4. Preservice Inspection

The applicant's preservice inspection program is acceptable if in compliance with the following criteria:

a. Disk forgings should be rough machined prior to heat treatment.

b. Each finished disk should be subjected to 100% volumetric (ultrasonic), surface, and visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in the ASME Boiler and Pressure Vessel Code, Sections III and V.

c. Finish machined bores, keyways, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. No flaw indications in keyway or hole regions are allowable.

d. Each turbine rotor assembly should be spin tested at the maximum speed anticipated during a turbine trip following loss of full load.

5. Turbine Disk Design

The applicant's design is acceptable if in compliance with the following:

The turbine assembly should be designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. The design of the turbine assembly should meet the following criteria:

a. The design overspeed of the turbine should be 5% above the highest anticipated speed resulting from a loss of load. The basis for the assumed design overspeed should be submitted to the staff for review.

b. The combined stresses of low pressure disks or high pressure rotors at design overspeed due to centrifugal forces, interference fit, and thermal gradients should not exceed 0.75 of the minimum specified yield strength of the material, or 0.75 of the measured yield strength in the weak direction of the materials if appropriate tensile tests have been performed on the actual disk material.

c. The turbine shaft bearings should be able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in turbine trip.

d. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed should be controlled in the design and operation so as to cause no distress to the unit during operation.

e. The turbine rotor and disk design should facilitate inservice inspection of all high stress regions, including bores and keyways, with the need for removing the disks from the shaft.

6. <u>Inservice Inspection</u>
The applicant's inservice inspection program is acceptable if in compliance with the following criteria:

The inservice inspection program for the steam turbine assembly should provide assurance that disk flaws that might lead to brittle failure of a disk at speeds up to design speed will be detected. The inservice inspection program for the turbine assembly should include the following:

a. Disassembly of the turbine at approximately 10-year intervals, during plant shutdown coinciding with the inservice inspection schedule as required by ASME Boiler and Pressure Vessel Code, Section XI, and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, low pressure turbine blades, low pressure disks, and high pressure rotors. This inspection should consist of visual, surface, and volumetric examinations, as required.

The applicant should keep abreast of technological advances in volumetric examination techniques so that when improved methods for inspection of turbine disks are developed they can be incorporated into the inservice inspection program.

b. An in-place visual examination of the turbine assembly at accessible locations should be conducted during refueling shutdowns at intervals not exceeding three years.

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.

For each area of review, the following review procedures are followed:

1. <u>Materials Selection</u>
The materials properties and the procedures to minimize flaws and improve fracture toughness, as described by the applicant, are reviewed and compared with the requirements of Section II.1 of this plan. If a new material not used in prior licensed cases

is utilized, the applicant's materials selection is reviewed and evaluated to establish its acceptability. Such an evaluation is based on the acceptance criteria of Section II of this review plan.

2. Fracture Toughness
The fracture toughness properties of the low pressure disk and high pressure rotor materials, including materials specimen test data, where applicable, are reviewed and compared with the requirements of Section II.2 of this plan. The applicant is permitted any of three alternates for deriving the fracture toughness of the disk materials.

3. High Temperature Properties
The high temperature properties of the high pressure rotor materials, including specimen test data, where applicable, are reviewed and compared with the requirements of Section II.3 of this plan.

4. Preservice Inspection
The preservice inspection program, including finish machining, ultrasonic inspection, surface inspection, visual inspection, and spin testing, is reviewed and compared with the requirements of Section II.4 of this plan. The extent to which the ultrasonic inspections and the acceptance criteria in the SAR agree with ASME Boiler and Pressure Vessel Code, Section III, NB-2530 for plate materials or NB-2540 for forgings, is reviewed.

5. Turbine Disk Design
The design and stress analysis procedures used for the high and low pressure turbine disks 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 at design overspeed and allowable stresses.

The SAR data are compared and evaluated against Section II.5 of this plan.

6. Inservice Inspection
The inservice inspection program described by the applicant, 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 failure of a turbine disk or rotor can be minimized by the use of suitable materials, adequate design, and preservice and inservice inspections. The applicant has described his program for assuring the integrity of low pressure turbine disks and high pressure turbine rotors by use of suitable materials of adequate fracture toughness, conservative design practices, and preservice and inservice inspections. The staff concurs that these provisions, as described in the SAR, provide reasonable assurance that the turbine disks will not fail during normal operation, including transients up to design overspeed."

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 III, V, and XI, American Society of Mechanical Engineers.

3. ASTM E-208, "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 Materials.

4. ASTM A-370, "Standard Methods and Definitions for Mechanical Testing of Steel Products," Annual Book of ASTM Standards, Parts 1, 2, 3, 4, or 31, American Society for Testing Materials.

5. J. A. Begley and W. A. Logsdon, Scientific Paper 71-1E7-MSLRF-P1, Westinghouse Electric Corp., July 26, 1971.

6. F. J. Witt and T. R. Mager, ORNL-TM-3894, Oak Ridge National Laboratory (1972).

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SECTION 10.3 MAIN STEAM SUPPLY SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Mechanical Engineering Branch (MEB)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The main steam supply system (MSSS) for both boiling water reactor (BWR) and pressurized water reactor (PWR) plants transports steam from the nuclear steam supply system to the power conversion system and various safety-related or non-safety-related auxiliaries. Portions of the MSSS may be used as a part of the heat sink to remove heat from the reactor facility during certain operations and may also be used to supply steam to drive engineered safety feature pumps. The MSSS may also include provisions for secondary system pressure relief in PWR plants.

The MSSS for the BWR direct cycle plant extends from the outermost containment isolation valves up to (but not including) the turbine stop valves, and includes connected piping of 2-1/2 inches nominal diameter or larger up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation. The MSSS for the PWR indirect cycle plant extends from the connections to the secondary sides of the steam generators up to (but not including) the turbine stop valves, and includes the containment isolation valves, safety and relief valves, connected piping of 2-1/2 inches nominal diameter or larger up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation and the steam line to the auxiliary feedwater pump turbine.

1. The APCSB reviews the MSSS to determine which, if any, portions of the system are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents. The system is reviewed to verify that:

   a. A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse enviornmental occurrences, and accident conditions, including loss of offsite power.

b. Appropriate quality group requirements and seismic design requirements are met for safety-related portions of the system.

c. Failures of non-seismic Category I equipment or structures, or pipe cracks or breaks in high and moderate energy piping will not preclude essential functions of safety-related portions of the system.

d. The system is capable of performing multiple functions such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine-driven auxiliary feedwater pumps), as may be specified for a particular design.

2. The APCSB reviews the MSSS with regard to measures provided to limit blowdown of the system in the event of a steam line break.

3. The APCSB also reviews the design of the MSSS with respect to the following:

   a. The functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.

   b. The capability to detect and control system leakage, and to isolate portions of the system in case of excessive leakage or component malfunctions.

   c. The capability to preclude accidental releases to the environment.

   d. Provisions for functional testing for safety-related portions of the system.

4. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The RSB identifies essential components associated with the main steam supply system inside the primary containment that are required for normal operations and accident conditions, establishes shutdown cooling load requirements versus time, determines the appropriate seismic and quality group classifications for system components and verifies the design transient used in establishing the flow capacity and set point(s) of steam generator relief and safety valves. The SEB determines the acceptability of the design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles. The MEB reviews and seismic qualification of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies 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 EICSB reviews the electrical portions of the MSSS with respect to the adequacy design, installation, inspection, and testing of essential electrical components and instrumentation and control functions.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the main steam supply system, 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 MSSS will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of the MSSS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. General Design Criterion 2, as related to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 34, as related to the system function of transferring residual and sensible heat from the reactor system in indirect cycle plants.

4. Regulatory Guide 1.26, as related to the quality group classification of the system.

5. Regulatory Guide 1.29, as related to the seismic design classification of system components.

6. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 review of 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.

The procedures for OL applications include 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 as a result of the staff's review.

The review procedures below are written for typical main steam supply systems for both direct and indirect cycle plants. The reviewer will select and emphasize material from this plan, as may be appropriate for a particular case.

1. There are significant differences in the design of the main steam supply system for an indirect cycle (PWR) plant as compared to that for a direct cycle (BWR) plant. Further, different portions of the MSSS are safety-related in different plant designs, although the safety functions of the system are much the same in all PWR plants, and also in all BWR plants. The first step in the review of the MSSS, then, is to determine which portions are designed to perform a safety function. For this purpose, the system is evaluated to determine the components and subsystems necessary for achieving safe reactor shutdown in all conditions or for performing accident prevention or mitigation functions.

2. The reviewer determines that essential (safety-related) portions of the MSSS are correctly identified and are isolable to the extent required from non-essential portions of the system. The system description and piping and instrumentation diagrams (P&IDs) are reviewed to verify that they clearly indicate the physical division between each portion. System arrangement drawings are reviewed to identify the means provided for accomplishing system isolation.

3. The SAR is reviewed to verify that essential portions of the MSSS are designed to Quality Group B or higher and seismic Category I requirements, and to verify that the design classifications specified meet the acceptance criteria. In general, the main steam lines from the steam generators to the containment isolation valves in PWR plants are classified seismic Category I and Quality Group B, and the main steam lines in BWR plants from the outer containment isolation valves to the main steam system shutoff valves or the turbine stop valves are classified seismic Category I and Quality Group B. In this regard APCSB will use the results of the RSB review under Standard Review Plan (SRP) 3.2.2.

4. The SAR is reviewed to assure that design provisions have been made to permit appropriate functional testing of system components important to safety. It is acceptable if the SAR delineates a testing and inspection program and the system drawings show any test recirculation loops and special connections around isolation valves that would be required by this program.

5. The system description, safety evaluation, component table, and P&IDs are reviewed to verify that the system has been designed to:

   a. Provide the necessary quantity of steam to any turbine-driven safety system pumps. The reviewer refers to the pump performance curves and verifies that the design is capable of providing the required steam flow to the turbine so that an adequate supply of water can be pumped. (OL)

   b. Assure safe plant operation by including appropriate design margins for pressure relief capacity and set points for the secondary system, and for removal of decay heat during various accident conditions, as may be applicable in a particular case. The review is done on a case-by-case basis, and system acceptability is based on a comparison of system flow rates, heat loads, maximum temperatures, and heat removal capabilities to those of similarly designed systems for previously reviewed plants. For PWR's the design is reviewed to verify system capability for controlled cool-down to about 350°F to allow actuation of RHR system.

c. Provide leakage detection means for steam or radioactivity leakage from the system in the event of a steam line break. Radioactivity monitors or temperature and pressure sensors are acceptable means for initiating signals to close the main steam line isolation valves and/or turbine stop valves to limit the release of steam during a steam line break accident.

d. Assure that in the event of a postulated break in a main steam line in a PWR plant, the design will preclude the blowdown of more than one steam generator, assuming a concurrent single component failure. In this regard the turbine stop and control valves are considered to be functional. The reviewer should verify that the main steam isolation valves and turbine stop and bypass valves can close against maximum steam flow.

6. The reviewer verifies that the system is designed so that essential functions will be maintained, as required, in the event of adverse environmental phenomena, certain pipe breaks, or loss of offsite power. The reviewer uses engineering judgment and the results of failure modes and effect analyses to determine that:

a. Failure of non-seismic Category I portions of the MSSS or of other systems located close to essential portions of the system, or of non-seismic Category I structures that house, support, or are close to essential portions of the MSSS, do not preclude operation of the essential portions of the MSSS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that confirm that the above conditions are met are acceptable.

b. Essential portions of the MSSS are protected from the effects of floods, hurricanes, tornadoes, and internally and externally generated missiles. Flood protection and missile protection criteria are evaluated under the standard review plans for Chapter 3 of the SAR. The locations and the design of the system and structures are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or that components of the system will be located in individual cubicles or rooms that will withstand the effects of winds, flooding, and tornado missiles is acceptable.

c. Essential portions of the MSSS are protected from the effects of high and moderate energy line breaks and cracks, including pipe whip, jet forces and environmental effects. Layout drawings are reviewed to assure that no high or moderate energy piping systems are close to essential portions of the MSSS, or that protection from the effects of failure will be provided. The means of providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

d. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The SAR is reviewed to verify that for each MSSS component or subsystem affected by a loss of offsite power the system functional capability meets or exceeds minimum design requirements. Statements in the SAR and results of failure modes and effects analyses are considered in assuring that the system meets these requirements. This is an acceptable verification of system functional reliability.

7. The descriptive information, P&IDs, MSSS drawings, and failure modes and effects analyses in the SAR are reviewed to assure that essential portions of the system will function following design basis accidents assuming a concurrent single active component failure. The reviewer evaluates the analyses presented in the SAR to assure 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 accident situation for the required time spans. For each case the design is acceptable if minimum system requirements are met.

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 supply system (MSSS) includes all components and piping from the outermost containment isolation valves to the turbine stop valves. The scope of review of the main steam supply system for the ___________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the MSSS and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for safety-related portions of the MSSS and system performance requirements for normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of safety-related portions of the MSSS and auxiliary supporting systems is in conformance with the 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 main steam supply system 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 main steam supply system 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 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. 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. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 10.3.6 STEAM AND FEEDWATER SYSTEM MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials Engineering Branch (MTEB)

Secondary - None

I. AREAS OF REVIEW

General Design Criterion 1 requires that systems important to safety shall be designed to quality standards commensurate with the importance of the safety functions to be performed. The steam and feedwater systems consist of NRC Quality Group B or C (ASME Code, Section III, Class 2 or 3) components.

The following areas relating to the general materials considerations for ASME Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, Class 2 and 3 components of the steam and feedwater systems are reviewed: (The review procedures for materials considerations for steam generators are given in Standard Review Plan 5.4.2.1.) The Class 2 and 3 components of the steam and feedwater systems include those portions beyond the outermost containment isolation valves in boiling water reactors (BWR's) and the secondary coolant system of pressurized water reactors (PWR's), except for those portions of the steam generator that come in contact with the primary coolant.

1. Fracture Toughness of Class 2 and 3 Components

   The fracture toughness properties and requirements for Class 2 and 3 components are reviewed. Typical components in this review include steam generator shells in PWR's, as well as carbon or low alloy steel portions of steam and feedwater lines in both PWR's and BWR's.

2. Materials Selection and Fabrication for Class 2 and 3 Components

   The materials selected for all Class 2 and 3 components and their fabrication are reviewed.

   For austenitic stainless steel components, the following points are reviewed:

   a. The steps taken to control the use of sensitized stainless steel.
   b. The controls placed on the composition of any nonmetallic external thermal insulation.

c. The controls placed on the welding procedures.
d. For all components, the qualification procedures for welds in areas of limited accessibility are reviewed.
e. For carbon or low alloy steel components, the controls placed on the welding procedures are reviewed.

II. ACCEPTANCE CRITERIA

1. Fracture Toughness of Class 2 and 3 Components

The fracture toughness properties of the ferritic materials of these components must meet the following requirements of the Code, Section III:

NB-2300 - "Fracture Toughness Requirements for Materials"
NB-2331 - "Test Requirements and Acceptance Standards - Material for Vessels"
NB-2332 - "Test Requirements and Acceptance Standards - Material for Piping (Pipe, Tubes, and Fittings), Pumps and Valves Excluding Bolting Materials." Paragraph NB-2332(b) states additionally that pressure-retaining materials (other than bolting) with nominal thickness over 2-1/2 in. must meet the requirements of NB-2331, and that the lowest service temperature must be not lower than the nil-ductility transition reference temperature, RT_{NDT}, plus 100°F unless a lower temperature is justified by following methods similar to those contained in Article G-2000 of Section III.
NC-2310 of the Summer 1972 Addenda to Section III - "Impact Testing (Class 2)"
ND-2310 of the Summer 1972 Addenda to Section III - "Impact Testing (Class 3)"

2. Materials Selection and Fabrication for Class 2 and 3 Components

The mechanical properties of materials specified for use in Class 2 and Class 3 components must be either as stated in Appendix I to Section III of the Code, or alternately, as indicated in Parts A, B, and C of Section II of the Code.

The following criteria are applicable to all austenitic stainless steel components:

a. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," describes acceptable criteria for preventing stress corrosion of stainless steel components of the steam and feedwater systems. Furnace-sensitized material should not be used, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling and storage, testing materials prior to fabrication, and determining the degree of sensitization that occurs during welding.
b. The composition of nonmetallic thermal insulation for the austenitic stainless steel components should be controlled as described in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Concentrations of leachable contaminants should be controlled as specified in position C.2.b and Figure 1 of this guide, especially with regard to sodium silicate inhibitor concentrations, to minimize the probability of stress-corrosion cracking of these components when the insulation is moistened.

c. Regulatory Guide 1.31, "Control of Stainless Steel Welding," describes acceptable criteria for assuring the integrity of welds in stainless steel components. The control of delta ferrite content of weld filler metal described in this guide has been modified by Branch Technical Position MTEB 5-1 (Ref. 12), which details acceptable standards for delta ferrite content of weld metal.

The following criteria are applicable to all components:

d. Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," provides the following criteria for assuring the integrity of welds in remote areas where inspection is difficult:
   (1) The performance qualification should require testing of the welds when conditions of accessibility to production welds are less than 30 to 35 cms (12-14 inches) in any direction from the joint.
   (2) Requalification is required for different accessibility conditions or when other essential variables listed in the Code, Section IX, are changed.

e. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," describe acceptable procedures for cleaning and handling Class 2 and 3 components of the steam and feedwater systems.

The following criterion is applicable to all carbon or low alloy steel components, in addition to the above-cited fracture toughness criteria:

f. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," describes acceptable criteria for supplementing Section IX of the Code, to prevent crack formation in the underbead areas and heat-affected zones of welds in these materials. The welding procedures should be qualified at the minimum preheat temperature, and production welds should be monitored to verify that the limits on preheat and interpass temperature are maintained.

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 areas listed in Section I for the required information, in accordance with the following procedures:

1. Fracture Toughness of Class 2 and 3 Components

The reviewer determines which components of the steam and feedwater systems will be made of carbon or low alloy steels, and determines that their fracture toughness properties are in conformance with Section II.1. It should be noted that Code Case 1576, and NC-2310 and ND-2410 of the 1974 Edition of Section III state that the lowest service temperature for Class 2 and Class 3 components must be not lower than RT_{NDT} + 30°F. This is unacceptable to the staff.

2. Materials Selection and Fabrication for Class 2 and 3 Components

The reviewer determines that the mechanical properties of the materials proposed for the steam and feedwater systems are in conformance with either Appendix I to Section III or to parts A, B, or C of Section II of the Code.

For austenitic stainless steel components, the following procedures are followed:

a. The reviewer examines the methods of controlling sensitized stainless steel and determines that they comply with the acceptance criteria stated in Section II.2.a, especially with respect to cleaning and protection from contamination during handling and storage, verification of nonsensitization of the material, and qualification of welding procedures. If alternative methods of testing 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. The reviewer may ask the applicant to justify technically his departures from the above-cited criteria. Alternative tests of qualification welds that have been previously accepted by the MTEB include the use of ASTM A-393-63 for determining the degree of sensitization of the heat affected zones of the qualification welds, and the use of ASTM A-262-70, as amended by Westinghouse Process Specification 84201 MW, for qualifying welds and testing raw materials for nonsensitization.

b. The reviewer determines whether nonmetallic thermal insulation will be used on any austenitic stainless steel components of the steam and feedwater systems, and verifies that the leachable impurities in this insulation lie within the "acceptable analyses" area of Figure 1 of Regulatory Guide 1.36, as discussed in Section II.2.b.

c. The reviewer examines the methods of controlling and measuring the amount of delta ferrite in stainless steel weld deposits, in accordance with the criteria stated in Section II.2.c, especially with respect to the filler metal acceptance procedures for delta ferrite content, and the examination of production welds for average content of delta ferrite.

d. The reviewer determines that the methods for qualifying welders for making welds in remote areas, and the methods for monitoring and certification of production welds in remote areas are in accordance with the acceptance criteria stated in Section II.2.d.

e. The reviewer determines that the methods for cleaning and handling the Class 2 and 3 components are in accordance with acceptance criteria stated in Section II.2.e.

f. For all carbon or low alloy steel components, the reviewer verifies that the minimum preheat and interpass temperatures for welding are specified in accordance with Section II.2.f.

3. General

If the information contained in the safety analysis report 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 OF 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 Class 2 and 3 components of the steam and feedwater systems satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, or Parts A, B, or C of Section II of the Code. The fracture toughness properties of ferritic materials satisfy the requirements of the Code and minimum service temperatures for these materials are set at 100°F above the nil-ductility transition reference temperatures.

"The controls imposed upon austenitic stainless steel are in accordance with 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 reasonable assurance that stress-corrosion cracking will not occur during the design life of the plant. The controls placed upon concentrations of leachable impurities in nonmetallic thermal insulation used on austenitic stainless steel components of the steam and feedwater systems are in accordance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

"The welding procedures used in limited access areas conform to Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The onsite cleaning and cleanliness controls during fabrication satisfy the positions given in Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." The precautions taken in controlling and monitoring the preheat and interpass temperatures during welding of carbon and low alloy steel components conform to Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel."

"Conformance with the codes, standards, and Regulatory Guides mentioned constitutes an acceptable basis for assuring the integrity of steam and feedwater systems, and for meeting in part the requirements of General Design Criterion 1."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. ASME Boiler and Pressure Vessel Code; Section III, Articles NB, NC, and ND, Appendix I and Appendix G; Section II, Parts A, B, and C; and Section IX; American Society of Mechanical Engineers.

3. ANSI Standard N45.2.1-1973 (Draft 2, Rev. 0), "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," November 15, 1973.

4. ASTM A-262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptability to Intergranular Attack in Stainless Steel," Annual Book of ASTM Standards, Part 3, American Society for Testing and Materials.

5. 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.

6. Process Specification 84201 MW, "Corrosion Testing of Wrought Austenitic Stainless Steel," Westinghouse Electric Corporation.

7. Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

8. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Nuclear Power Plants."

9. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

10. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

11. Standard Review Plan 5.4.2.1, "Steam Generator Materials."

12. 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.

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SECTION 10.4.1 MAIN CONDENSERS

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system.

1. The APCSB reviews the performance requirements of the main condenser for both direct and indirect cycle plants during all operating conditions. Emphasis will be placed on the review of direct cycle facilities with regard to the prevention of loss of vacuum, galvanic corrosion, and hydrogen buildup.

2. The APCSB reviews the design of the MC system with respect to the following:

   a. The means to detect and control system leakage, to detect radioactive leakage into or out of the system, and to preclude accidental releases of radioactive materials to the environment in amounts in excess of established limits.

   b. Instrumentation and control features that determine and verify that the MC is operating in a correct mode.

   c. The means provided to deal with flooding from a complete failure of the MC and to preclude damage to safety-related equipment from the flooding.

   d. The capability of the MC to withstand the blowdown effects of steam from the turbine bypass system.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows: the ETSB evaluates the inventory of radioactive contaminants in the MC during power operation and during shutdown.

II. ACCEPTANCE CRITERIA

There are no general design criteria or regulatory guides that directly apply to the main condenser. Acceptability of the design of the main condenser system, as described in the applicant's safety analysis report (SAR), is based on the system being designed such that failures do not cause unacceptable flooding of areas housing safety-related equipment, or result in excessive releases of radioactivity to the environment. An additional basis for determining the acceptability of the MC system will be the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to determine that the design criteria and bases and the preliminary design meet the acceptance criteria given in Section II of this plan. For the review of 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. The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the system description delineates the main condenser system capabilities including the minimum system heat transfer and system flow requirements for normal plant and turbine bypass operation. Measures provided to prevent loss of vacuum, galvanic corrosion of MC tubes and components, and hydrogen buildup in the MC are reviewed, with particular emphasis on these measures in direct cycle (boiling water reactor) plants. System performance requirements are reviewed to determine that they satisfactorily limit possible system degradation conditions (e.g., leakage, partial loss of vacuum) and describe the procedures that are followed to detect and correct these conditions. The SAR is also reviewed to determine that any allowed MC system degraded operation does not have an adverse effect on the reactor primary system.

2. The reviewer evaluates the MC system design to verify that:

    a. Means have been provided for detecting and controlling condenser leakage.

    b. Measures have been provided to detect radioactive leakage into and out of the MC system and to preclude unacceptable accidental releases of radioactivity to the environment from the system.

    c. The system is provided with instrumentation and control features that determine and verify that the MC is operating in a correct mode.

3. The reviewer uses engineering judgment and the results of failure modes and effects analyses to determine that:

a. The failure of a main condenser and the resulting flooding will not preclude operation of any essential systems. Reference to sections of the SAR describing plant features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems. Statements in the SAR that verify that the above conditions are met are acceptable.

b. The system, in conjunction with the main steam system, has provisions to detect loss of condenser vacuum and to effect isolation of the steam source. For direct cycle plants, it will be acceptable if the detection system in the MC can actuate the main steam isolation valves to limit the quantity of steam lost out of the condenser.

c. Design provisions have been incorporated into the MC that will preclude component or tube failures from turbine bypass system steam blowdown.

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 condenser system includes all components and equipment from the turbine exhaust to the connections and interfaces with the main condensate and other systems. The scope of review of the main condenser system for the ____________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the main condenser system and supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the main condenser system and the requirements to preclude safety-related equipment malfunctions or failures due to rupture of the main condensers. (CP)] [The review has determined that the design of the main condenser system and supporting systems is in conformance with the proposed design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's design criteria and design bases for the main condenser system and supporting systems to applicable staff technical positions and industry standards.

"The staff concludes that the design of the main condenser system conforms to all applicable staff positions and industry standards, and is acceptable."

V. REFERENCES

None.

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SECTION 10.4.2 MAIN CONDENSER EVACUATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

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.

1. The main condenser evacuation system (MCES) generally consists of two subsystems: the "hogging" or startup system which initially establishes main condenser vacuum, and the normal system which maintains condenser vacuum once it has been established. The review of each MCES subsystem includes the design, design objectives, capacity, method of operation, and factors that influence gaseous radioactive material handling, e.g., system interfaces and potential bypass routes. The ETSB review includes the system piping and instrumentation diagrams (P&IDs).

2. The quality group classifications of piping and equipment, and the bases governing the design criteria chosen are reviewed.

3. Design features to preclude the possibility of an explosion if the potential for explosive mixtures exists are reviewed.

Provisions incorporated to sample and monitor radioactive materials in gaseous effluents from the MCES are reviewed in Standard Review Plan (SRP) 11.5.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The MCES capacity should be consistent with the industry guidelines given in Reference 2. Either mechanical vacuum pumps or steam jet air ejectors may be used for hogging (startup) or normal evacuation of the main condenser.

2. The components of the system may be designed to Quality Group D as defined in Regulatory Guide 1.26 (Ref. 3) and to a non-seismic design classification.

3. If there is a potential that explosive mixtures may exist, the MCES should be designed to withstand the effects of an explosion or provide redundant instrumentation to detect and annunciate the buildup of potentially explosive mixtures. Instrumentation with automatic alarm and control functions should be provided to continuously monitor concentrations of the appropriate gases in portions of the system having the potential for containing explosive mixtures. The design should include precautions to stop continuous leakage paths, i.e., provisions for liquid seals downstream of rupture discs and for prevention of permanent loss of the liquid seals in the event of an explosion.

4. Provisions to control and monitor releases of radioactivity to the environment from the MCES must conform to General Design Criteria 60 and 64 (Ref. 1).

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 of the MCES, the P&IDs are reviewed to determine the flow paths of gases through the system, including all bypasses, and the points of release of gaseous wastes to the environment or other systems. This information is used in SRP 11.3 to calculate the quantity of radioactive material released annually in gaseous effluents during normal operations, including anticipated operational occurrences. ETSB verifies that water from the mechanical vacuum pumps and condensate from the steam jet air ejectors are classified as radioactive liquids and treated accordingly.

2. ETSB reviews the equipment quality group classifications.

3. If there is a potential that explosive mixtures may exist, ETSB determines whether the applicant has designed the MCES to withstand the effects of such an explosion, or has provided redundant instrumentation to detect, annunciate, and prevent the buildup of potentially explosive mixtures. ETSB will also determine if the applicant's design includes adequate provisions to stop continuous leakage paths after an explosion.

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 main condenser evacuation system includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive material to the environment. The scope of our review included the system capability to transfer radioactive gases to the gaseous waste or ventilation systems, the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with General Design Criteria 60 and 64 and the quality group classification of equipment and components used to collect gaseous radioactive wastes relative to the guidelines of Regulatory Guide 1.26. We have reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the components of the main condenser evacuation system. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the main condenser evacuation system to the applicable regulations and regulatory guides referenced above, as well as to branch technical positions and industry standards. Based on our evaluation, we find the proposed main condenser evacuation system acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and Criterion 64, "Monitoring Radioactivity Releases."

2. "Standards for Steam Surface Condensers," 6th Ed., Heat Exchanger Institute (1970).

3. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 2.

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SECTION 10.4.3 TURBINE GLAND SEALING SYSTEM

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Auxiliary and Power and Conversion Systems Branch (APCSB)

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.

The turbine gland sealing system design, design objectives, method of operation, and factors that influence gaseous radioactive material handling, e.g., source of sealing steam, system interfaces, and potential leakage paths are reviewed. The ETSB review includes piping and instrumentation diagrams (P&IDs).

Provisions incorporated to sample and monitor radioactive materials in gaseous effluents are reviewed in Standard Review Plan (SRP) 11.5.

Provisions for controlling the release of radioactive materials from the gland seal condenser vent are reviewed in SRP 11.3.

During the OL stage, the APCSB reviews the potential effect of high energy pipe breaks within this system on safety-related equipment.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

The turbine gland sealing system should be designed to provide for the collection and condensation of sealing steam and the venting and treatment (as required in Ref. 1) of noncondensables. Quality Group D as defined in Regulatory Guide 1.26 (Ref. 2) and a non-seismic design classification are acceptable design criteria for this system.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from this review plan, as may be appropriate for a particular case.

ETSB reviews the system P&IDs to determine the source of sealing steam and the disposition of steam and noncondensables vented from the gland seal. Where sealing steam from primary coolant condensate is used, the review includes the radiological processing and monitoring provisions in accordance with SRPs 11.3 and 11.5.

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 turbine gland sealing system includes the equipment and instruments to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The scope of our review included the source of sealing steam, and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64. We have reviewed the applicant's system descriptions and design criteria for the components of the turbine gland sealing system and found them consistent with Regulatory Guide 1.26.
>
> "The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the turbine gland sealing system to the applicable regulations and regulatory guides referenced above, as well as to branch technical positions and industry standards. Based on our evaluation, we find the proposed turbine gland sealing system acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Material to the Environment," and Criterion 64, "Monitoring Radioactivity Releases."

2. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 2.

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SECTION 10.4.4 TURBINE BYPASS SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The turbine bypass system (TBS) provides operational flexibility so that the plant may accept certain load changes without disturbing the nuclear steam supply system. The TBS is designed to discharge a stated percentage of rated main steam flow directly to the main condensers, bypassing the turbine. This steam bypass enables the plant to take step load reduction up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure for a boiling water reactor (BWR) and steam generator pressure for a pressurized water reactor (PWR). The TBS is not required for safe shutdown as the relief and safety valves are operated under emergency conditions. The system is not required to function as a heat sink for the prevention or mitigation of postulated accidents. Failure of the TBS during a load reduction or turbine trip would result in the actuation of the relief valves and possibly the safety valves.

The APCSB reviews the system from the branch connection at the main steam system to the main condensers.

1. APCSB reviews the TBS to determine that it has sufficient capacity and reliability to minimize the necessity for relief and safety valve actuation and that a failure of the system or system components will not have an adverse effect on essential equipment.

2. The APCSB reviews the TBS functional requirements for both normal and abnormal operating conditions, and with respect to the following capabilities: (a) to isolate those portions of the system that could leak or malfunction; (b) to perform adequate operational testing and inservice inspection; and (c) to assure there are no adverse effects of postulated system piping failures on safety-related equipment.

3. The applicant's proposed technical specifications are reviewed at the operating license stage, as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results are used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows: the EICSB, upon request, provides information pertaining to controls and instrumentation for the system. The RSB determines that the appropriate seismic and quality group classifications have been established for system components and that the steam bypass capacity is consistent with reactor transient analysis.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the turbine bypass system, as described in the applicant's safety analysis report (SAR), is based on the criteria below. An additional basis for determining the acceptability of the TBS will be the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience.

The design of the TBS is acceptable if the integrated design of the system is in accordance with the following criteria:

1. Failure or malfunction of the TBS does not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).

2. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

3. The system should be capable of providing sufficient steam bypass to the main condenser so that a reactor trip will not occur as a result of load rejections.

III. REVIEW PROCEDURES

The 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 review of 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.

The procedures for OL applications include 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 as a result of the staff's review.

The reviewer selects and emphasizes material from this review plan, as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (PID's) delineate the system and components.

2. The SAR is reviewed to verify that the system design bases and an evaluation of the system capacity are provided, including the relation between the TBS capacity and relief valve capacity in terms of percentage of rated main steam flow, the maximum reactor power step change the system is designed to accommodate without a reactor or turbine trip, and the maximum electric load step change the reactor is designed to accommodate without reactor control rod motion or steam bypassing.

3. The reviewer verifies that the combined capacity of the TSB and relief valves is sufficient to preclude safety valve actuation in the event of a turbine trip or large electric load rejection.

4. The reviewer uses engineering judgment and the results of failure modes and effects analyses to determine that:

   a. Failure of the TBS to operate will not preclude operation of any essential systems. Statements in the SAR that confirm the above are acceptable.

   b. Failure of the TBS high energy piping will not have adverse effects on any safety-related systems or components that may be located close to the system.

IV. <u>EVALUATION FINDINGS</u>

The reviewer determines 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 turbine bypass system (TBS) includes all components and piping from the branch connection at the main steam system to the main condensers. The scope of review of the turbine bypass system for the ______________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the TBS and auxiliary supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the turbine bypass system and the requirements for safe operation of the TBS during normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the turbine bypass system and auxiliary supporting systems is in conformance with the 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 turbine bypass system and its supporting systems to applicable staff technical positions and industry standards.

"The staff concludes that the design of the turbine bypass system conforms to all applicable staff 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 Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 10.4.5 CIRCULATING WATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems.

1. The APCSB reviews the performance of the CWS with respect to its functional requirements and the effects of adverse environmental occurrences, abnormal operational transients, or accident conditions such as loss of offsite power.

2. The APCSB reviews the CWS to determine that a malfunction, failure of a component, or failure of a circulating water pipe do not have unacceptable adverse effects on the functional performance capabilities of safety-related systems located in the immediate area.

3. APCSB further reviews the design of the circulating water system with respect to the following:

   a. The capability to detect and control flooding of safety related areas due to circulating water system leakage.

   b. The compatibility of the methods proposed for control of water chemistry and of long-term corrosion and organic fouling with system components and piping materials.

   c. Provisions for instrumentation to permit operational testing of the system and to annunciate abnormal and unsafe operating conditions.

4. The APCSB reviews the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows:

the EICSB will, upon request, review the system instrumentation and controls as they may relate to operations that could effect safety-related systems or components.

II. ACCEPTANCE CRITERIA

Acceptability of the circulating water system, as described in the applicant's safety analysis report (SAR), is based on the criteria below and on the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. There are no general design criteria or regulatory guides that directly apply to the functional performance requirements for the CWS. Specific criteria for the CWS are as follows:

1. Means should be provided to detect and control flooding of safety related areas due to leakage from the CWS.

2. Malfunction or a failure of a component or piping of the CWS should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.

3. Agents used for the control of water chemistry, corrosion, and organic fouling should be compatible with the materials of the system.

4. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and cracks in moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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 procedures for OL reviews include 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 as a result of the staff's review.

The reviewer will select and emphasize material from this review plan as may be appropriate for a particular case.

1. Although the circulating water system is not safety-related, a failure of this system, or any of its components, may affect a safety-related component or system. Since large quantities of water flow through the CWS, a leak or break in a component or pipe could cause severe and unacceptable flooding of adjacent areas. The APCSB reviews the descriptions and drawings in the SAR and determines that provisions are incorporated in the design to prevent flooding of areas containing safety-related equipment or to mitigate the consequences of flooding.

2. The APCSB reviews the CWS to verify that the capability to detect leaks and secure the system quickly and effectively exists. The reviewer verifies that the design includes provisions to minimize hydraulic transients and their effect upon the functional capability and the integrity of system components.

3. Based on the information contained in the SAR, the reviewer verifies that the applicant's proposed methods for control of water chemistry and of long-term corrosion and organic fouling, and the chemical agents used for these purposes are compatible with the system materials.

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 circulating water system includes all components and equipment necessary to provide the main condenser with a continuous supply of cooling water. The scope of review of the cooling water system for the ________________ plant, included layout drawings, piping and instrumentation diagrams, and descriptive information for the system. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the circulating water system. (CP)] [The review has determined that the design of the circulating water system is in conformance with the 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 circulating water system to applicable staff positions and industry standards.

"The staff concludes that the design of the circulating water system conforms to all applicable staff 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 Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 10.4.6 CONDENSATE CLEANUP SYSTEM

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The condensate cleanup system (CCS) removes dissolved and suspended impurities due to corrosion from condenser or steam generator leaks that could be introduced into the CCS by carryover from the main steam system. The CCS is not required for safe shutdown or mitigation of postulated accidents, but is important in maintaining the primary coolant quality in direct cycle plants or the secondary coolant quality in indirect cycle plants.

The APCSB reviews the CCS from the supply point downstream of the condensate pumps to the discharge point upstream of the feedwater heaters, and also to the interfaces with the effluent treatment systems.

1. The APCSB reviews the CCS to determine that the system provides feedwater to the reactor for direct cycle plants or to steam generators for indirect cycle plants that meets water purity requirements. For plants with salt water-cooled condensers the design measures taken to assure that the chloride concentration is limited to allowable values until the condensate and feedwater systems can be isolated in the event of a condenser tube rupture are reviewed.

2. APCSB reviews the system to determine that the design satisfies the recommendations of Branch Technical Position APCSB 3-1 with respect to breaks and cracks in high and moderate energy system piping.

3. The applicant's proposed technical specifications are reviewed for operating license applications as they relate to areas covered in this plan.

Secondary reviews are performed by ETSB to determine the effect of the CCS on fission and corrosion product concentrations and the effect of the quantity of spent resin and regenerant solution on radwaste system requirements.

II. ACCEPTANCE CRITERIA

Acceptability of the CCS design, as described in the applicant's safety analysis report (SAR), is based on the criteria below and on the degree of similarity of the design with that of previously reviewed plants with satisfactory operating experience. There are no general design criteria that apply directly to the CCS.

1. For direct cycle plants, the design of the CCS should conform to the recommendations of Regulatory Guide 1.56, including Regulatory Positions 1 through 6, and the Appendix, as related to the design of condensate demineralizer systems to maintain the proper water purity specified for the reactor. For indirect cycle plants the design should conform with Regulatory Positions 1 through 5 and the secondary water chemistry specifications of the nuclear steam supply system vendor.

2. The design of the CCS should meet Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside 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 (PSAR) meet the acceptance criteria given in Section II of this plan. For the review of 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 (FSAR).

The procedures for OL reviews include 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 as a result of the staff's review. The reviewer will select and emphasize material from the paragraphs below, as may be appropriate for a particular case.

1. Layout drawings are reviewed to ascertain there are no safety-related components or systems located close to the high or moderate energy pipes of the CCS, or that protection from the effects of failures of these pipes is provided. The means of providing such protection will be given in Chapter 3.6 of the SAR and procedures for reviewing this information are given in the corresponding review plans.

2. The APCSB evaluates the system design information and drawings and, utilizing engineering judgement, operational experience, and performance characteristics of similar, previously approved systems, verifies that:

   a. The system meets the requirements for condensate cleanup capacity, provides effluent of the required purity, and contains adequate instrumentation to monitor the effectiveness of the system.

b. For plants with salt or brackish water-cooled condensers, the system has adequate capacity to maintain acceptable chloride concentrations in the feedwater system for a sufficient period of time to allow remedial action in the event of condenser tube failures, and the technical specifications and operational procedures adequately describe the method used to isolate the condensate and feedwater systems.

c. The system is connected to radioactive waste disposal systems to allow disposal of spent resin or regenerant solutions when required.

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 condensate cleanup system includes all components and equipment necessary for the removal of dissolved and suspended impurities which may be present in the condensate. The scope of the review of the condensate cleanup system for the _________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and for supporting systems essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the condensate cleanup system and the requirements for operation of the system. (CP).] [The review has determined that the design of the condensate cleanup system and supporting systems is in conformance with the design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's design criteria and design basis for the condensate cleanup system and supporting systems to applicable regulatory guides, staff technical positions, and industry standards.

"The staff concludes that the design of the condensate cleanup system conforms to all applicable guides, staff positions, and industry standards, and is acceptable."

V. REFERENCES

1. Regulatory Guide 1.56, "Maintenance of Purity in Boiling Water Reactors."

2. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 10.4.7 CONDENSATE AND FEEDWATER 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)
Structural Engineering Branch (SEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the reactor for boiling water reactor (BWR) plants and to the steam generators for pressurized water reactor (PWR) and high temperature gas-cooled reactor (HTGR) plants. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low pressure feedwater heaters to the feedwater pumps, and then is pumped through the high pressure feedwater heaters to the nuclear steam supply system.

APCSB reviews the CFS from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system. For indirect cycle plants, there are also interfaces with the secondary water makeup system and the auxiliary feedwater system. The CFS is used for normal shutdown. The only part of the CFS classified as safety-related, i.e., required for safe shutdown or in the event of postulated accidents, is the feedwater piping from the steam generators to, and including, the outermost containment isolation valve for indirect cycle plants.

1. The APCSB reviews the characteristics of the CFS with respect to the capability to supply adequate feedwater to the nuclear steam supply system as required for normal operation and shutdown.

2. The APCSB review determines that an acceptable design has been established for:

   a. The interfaces of the CFS with the auxiliary feedwater system (PWR), the reactor core isolation cooling system (BWR), and the condensate cleanup system.

   b. The feedwater system (PWR), including the auxiliary feedwater system piping entering the steam generator, with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

c. The detection of major system leaks that could affect the functional performance of safety-related equipment.

3. The APCSB reviews the applicant's proposed technical specifications for operating license applications as they relate to areas covered in this plan.

Secondary review evaluations are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The EICSB will, upon request, review the feedwater control system (BWR) or steam generator level control system (PWR). The RSB reviews the system for appropriate seismic and quality group classifications. Upon request, the SEB determines the acceptability of design analyses, procedures, and criteria used to establish the structural adequacy of devices or restraints as they may relate to significant water hammer forces in system piping, the MEB reviews test programs and the operability of components that may be affected by water hammer and confirms that piping and components are designed in accordance with applicable codes and standards, and the MTEB verifies that inservice inspection requirements are met for system components that may be affected by forces from water hammer.

II. ACCEPTANCE CRITERIA

Acceptability of the condensate and feedwater system, as described in the applicant's safety analysis report (SAR), is based on the criteria below and on the degree of similarity of the design to that of previously reviewed and approved plants.

1. Regulatory Guide 1.26, as related to the quality group classification of safety-related system components.

2. Regulatory Guide 1.29, as related to the seismic design classification of safety-related system components.

3. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

III. REVIEW PROCEDURES

The 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 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 procedures for OL applications include 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 as a result of the staff's review.

The reviewer will select and emphasize material from this review plan as may be appropriate for a particular case.

1. The SAR is reviewed to determine that the system description and diagrams delineate the function of the condensate and feedwater system under normal and abnormal conditions. The reviewer verifies the following:

    a. The system has been designed to function as required for all modes of operation. The results of failure modes and effects analyses presented in the SAR, if any, are used in making this determination.

    b. The system piping is designed to preclude hydraulic instabilities from occurring in the piping for all modes of operation. As appropriate, the reviewer evaluates the results of model tests and analyses that are relied on to verify that water hammer will not occur, or proposed tests of the installed system that are intended to verify design adequacy.

    c. The outermost containment isolation valves and all downstream piping to the nuclear steam supply system are designed in accordance with seismic Category I and appropriate quality group requirements, as determined by RSB.

    d. Breaks in system components or piping will not result in adverse effects on the functional performance of essential systems or components. The means for providing such protection will be given in Section 3.6 of the SAR and procedures for reviewing the information presented are given in the corresponding review plans.

    e. The CFS design is such that the plant can be safely shut down using the auxiliary feedwater system or the reactor core isolation cooling system, if required.

    f. The CFS design, or other plant systems, provide the capability to detect and control leakage from the system.

    g. Measures will be taken, as appropriate, to protect personnel from any toxic effects of chemicals used for feedwater treatment.

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 condensate and feedwater system includes all components and equipment from the condenser outlet to the connection with the nuclear steam supply system and to the heater drain system, [secondary water makeup system, and auxiliary feedwater system interfaces. (PWR's only)]. The scope of the review of the condensate and feedwater system for the ______________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and supporting systems essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and bases for the condensate and feedwater system and the requirements for system performance for all conditions of plant operation. (CP)] [The review has determined that the design of the condensate and feedwater system and supporting systems is in conformance with the design criteria and design bases. (OL)]

"The basis for acceptance in the staff review has been conformance of the applicant's design criteria and design bases for the condensate and feedwater system and supporting systems to applicable regulatory guides, staff technical positions, and industry standards.

"The staff concludes that the design of the condensate and feedwater system conforms to all applicable guides, staff positions, and industry standards, and 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 1.

2. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

3. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

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SECTION 10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the steam generator blowdown system (SGBS), as described in the applicant's safety analysis report (SAR), in the specific areas that follow. At the operating license (OL) stage, the ETSB review consists of confirming the design accepted at the CP stage.

1. ETSB reviews the SGBS design objectives in terms of expected and design flows, process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (e.g., demineralization and reverse osmosis), and process instrumentation and controls for maintaining operations within established parameter ranges.

2. ETSB reviews the seismic design and quality group classifications for equipment, piping, and components.

The liquid and gaseous waste treatment aspects of the SGBS are reviewed under Standard Review Plans (SRP) 11.2 and 11.3. Liquid and gaseous process and effluent radiological monitoring is reviewed under SRP 11.5.

APCSB evaluates, under SRP 3.6.1, the effect of system failures to assure that safety-related equipment will not be made inoperable.

II. ACCEPTANCE CRITERIA

ETSB accepts the design of the steam generator blowdown system if the following conditions are met.

1. The SGBS is sized to accommodate the design blowdown flow needed to maintain secondary coolant chemistry parameters for normal operation, including anticipated operational occurrences. Equipment capacities are based on design blowdown rates and are such that temperature limits for heat-sensitive processes are not exceeded. Instrumentation and automatic controls ensure operation within design parameters.

2. The blowdown system downstream of the outer containment isolation valves is designed according to the provisions of Branch Technical Position ETSB 11-1 (Revision 1).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from this review plan as may be appropriate for a particular case.

1. ETSB considers the pressure, temperature, flow rate, secondary coolant chemistry limits, main condenser water inleakage, and primary to secondary leakage to determine whether the SGBS design has included the effects of normal operation and anticipated operational occurrences (e.g., main condenser inleakage or primary to secondary leakage). ETSB determines that the design parameters are reasonable. If the proposed system includes processes which are heat-sensitive (e.g., demineralization or reverse osmosis), ETSB verifies that the design includes instrumentation and controls to protect the temperature sensitive elements. ETSB ensures that instrumentation and process controls are provided to control flashing, liquid levels, and process flow through the proper components for the radioactivity levels expected.

2. ETSB compares the quality group and seismic design classifications of SGBS components to the guidelines of Branch Technical Position ETSB 11-1 (Revision 1).

3. ETSB reviews the waste treatment and radiological process and effluent monitoring aspects of the SGBS in SRP's 11.2, 11.3 and 11.5. ETSB reviews the proposed piping and instrumentation diagrams (P&IDs) and process flow diagrams, the method of operation, the processing to be provided, and the interfaces between the blowdown system and other plant systems to determine: (1) whether unusual design conditions exist which could lead to safety problems, and (2) that the system is capable of performing its intended functions.

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 steam generator blowdown system (SGBS) controls the concentration of chemical impurities and radioactive materials in the secondary coolant. The scope of review of the SGBS included piping and instrumentation diagrams, seismic and quality group classifications, design process parameters, and instrumentation and process controls. The review has included the applicant's evaluation of the proposed system operation and the applicant's estimate of the controlling process parameters.

"We have reviewed the capability of the system to process blowdown flows and to maintain secondary coolant chemical and radiological limits, and the capability of the instrumentation and process controls to maintain system operation within the proposed limits.

"The basis for acceptance in our review has been conformance of the applicant's designs and design criteria to applicable regulatory guides, staff technical positions, and industry standards.

"Based on the foregoing evaluation, we conclude that the proposed steam generator blowdown system is acceptable."

V. REFERENCES

1. Branch Technical Position ETSB 11-1 (Revision 1), "Design Guidance for Radioactive Waste Management Systems installed in Light Water Cooled Nuclear Power Reactor Plants," attached to SRP 11.2.

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 10.4.9 AUXILIARY FEEDWATER SYSTEM (PWR)

REVIEW RESPONSIBILITIES

Primary - Auxiliary and Power Conversion Systems Branch (APCSB)

Secondary - Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Structural Engineering Branch (SEB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

The auxiliary feedwater system (AFS) normally operates during startup, hot standby and shutdown as the feedwater system for pressurized water reactor (PWR) plants. In conjunction with a seismic Category I water source, it also functions as an emergency heat removal system to transfer heat from the primary system when the main feedwater system is not available for emergency conditions including small LOCA cases. The AFS operates over a time period sufficient either to hold the plant at hot standby for several hours or to cool down the primary system, at a rate not to exceed limits specified in technical specifications, to temperature and pressure levels at which the low pressure decay heat removal system can operate.

The APCSB reviews the AFS from the condensate storage tank (normal operation), or the seismic Category I water supply including valving and cross connects (emergency operation), to the connections with the steam generators, which are made either through a connection to the main feedwater piping or through separate auxiliary feedwater piping directly to the steam generators. All inter-connections and cross-connections are included in the review.

The review also includes AFS components, e.g., pumps, valves, and piping, with respect to their functional performance as affected by adverse environmental occurrences, by abnormal operational requirements, and off-normal conditions, e.g., small breaks in the primary system or the loss of offsite power.

The system is reviewed to determine that a single malfunction, a failure of a component, or the loss of a cooling source does not reduce the safety-related functional performance capabilities of the system. The APCSB reviews to assure that:

1. System components and piping have sufficient physical separation or shielding to protect the essential portions of the system from the effects of internally and externally generated missiles.

2. The system satisfies the recommendations of Branch Technical Position APCSB 3-1 with respect to the effects of pipe whip and jet impingement that may result from high or moderate energy piping breaks or cracks (in this regard the AFS is considered to be a high energy system).

3. The system and components satisfy design code requirements, as appropriate for the assigned quality group and seismic classifications.

4. The failure of non-essential equipment or components does not affect essential functions of the system.

5. The system is capable of withstanding a single active failure.

6. The system possesses diversity in motive power sources such that system performance requirements may be met with either of the assigned power sources, e.g., a system with an a-c subsystem and a redundant steam/d-c subsystem.

7. The system design precludes the occurrence of fluid flow instabilities, e.g., water hammer, in system inlet piping during normal plant operation or during upset or accident conditions (see Standard Review Plan 10.4.7).

8. Functional capability is assured by suitable protection during abnormally high water levels (adequate flood protection during the probable maximum flood).

9. The capability exists to detect, collect, and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.

10. Provisions are made for operational testing.

11. Instrumentation and control features are provided to verify the system is operating in a correct mode.

12. The applicant's proposed technical specifications are such as to assure the continued reliability of the AFS during plant operation; i.e., the limiting conditions for operation and the surveillance testing requirements are specified and are consistent with those for other similar plants.

Secondary review evaluations are performed by other branches and the results used by the APCSB to complete the overall evaluation of the system. The secondary reviews are as follows. The RSB identifies any functional interfaces between essential components of the reactor coolant or emergency core cooling systems and the AFS that are required for operation during normal operations or accident conditions. The RSB establishes post-accident heat loads and the associated time intervals available for cooling various components. The RSB also determines the appropriate seismic and quality group classifications. The SEB 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 probable maximum flood (PMF), and tornado missiles. The MEB reviews the seismic qualification testing and operability of components and confirms that components, piping, and structures are designed in accordance with applicable codes and standards. The MTEB verifies that inservice inspection requirements are met for system components and, upon request, verifies the compatibility of the materials of construction with service conditions. The EICSB evaluates system controls, instrumentation, and power sources with respect to capability, capacity, and reliability during normal and emergency conditions.

II. ACCEPTANCE CRITERIA

Acceptability of the design of the auxiliary feedwater system, 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 AFS is the degree of similarity of the design with that for previously reviewed plants with satisfactory operating experience. Listed below are the specific criteria as they relate to the AFS.

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, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.

2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.

3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.

4. General Design Criterion 19, as related to the design capability of system instrumentations and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.

5. General Design Criterion 44, to assure:

    a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

6. General Design Criterion 45, as related to design provisions made to permit periodic inservice inspection of system components and equipment.

7. General Design Criterion 46, as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

8. Regulatory Guide 1.26, as related to the quality group classification of system components.

9. Regulatory Guide 1.29, as related to the seismic design classification of system components.

10. Branch Technical Positions APCSB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.

11. Branch Technical Position APCSB 10-1, as related to auxiliary feedwater pump drive and power supply diversity.

III. REVIEW PROCEDURES

The 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 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 procedures for OL applications also include 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 as a result of the staff's review.

For the purpose of this review plan, a typical system is assumed which has redundant auxiliary feedwater trains, with a 50% capacity motor-driven pump in each train feeding directly to the steam generators, and a 100% capacity steam turbine-driven pump able to supply either of the redundant trains. The pumping capacity is chosen so that the system's is able to hold the plant at hot standby and subsequently to cool down the reactor at specified cooldown rates. This requirement is also met for conditions involving a small break area loss-of-coolant accident (LOCA) or a pipe break outside containment. For cases where there are variations from the typical arrangement, the reviewer adjusts the review procedures to suit the design. However, the system design is required to meet the acceptance criteria given in Section II of this plan.

1. The SAR is reviewed to determine that the system description and piping and instrumentation diagrams (P&IDs) identify the AFS equipment and arrangement that is used for normal operation and for safe plant shutdown (essential) operation. The system P&IDs layout drawings, and component descriptions and characteristics are then reviewed to verify that:

    a. Minimum performance requirements for the system are sufficient for the various functions of the AFS.

    b. Essential portions of the AFS are isolable from non-essential portions, so that system performance is not impaired in the event of a failure of a non-essential component.

    c. Component and system descriptions in the SAR include appropriate seismic and quality group classifications, and the P&IDs indicate any points of change in piping quality group classification.

    d. Design provisions have been made that permit appropriate inservice inspection and functional testing of system components important to safety. It is acceptable if the SAR information delineates a testing and inspection program and if the system drawings show the necessary recirculation loops and around pumps or isolation valves as may be required by this program.

2. The reviewer verifies that the system safety function will be maintained as required, in the event of adverse environmental phenomena, breaks or cracks in fluid system piping outside containment, system component failures, loss of an onsite motive power source, or loss of offsite power. The reviewer uses engineering judgement and the results of failure modes and effects analyses to determine that:

    a. The failure of 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 that house, support, or are close to essential portions of the AFS, will not preclude operation of the essential portions of the AFS. Reference to SAR sections describing site features and the general arrangement and layout drawings will be necessary, as well as the SAR tabulation of seismic design classifications for structures and systems.

    b. The essential protions of the AFS are protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles. Flood protection and missile protection criteria are discussed and evaluated in detail under the standard review plans for Chapter 3 of the SAR. The location and design of the system, structures, and pump rooms (cubicles) are reviewed to determine that the degree of protection provided is adequate. A statement to the effect that the system is located in a seismic Category I structure that is tornado missile and flood protected, or the components of the system will be located in individual cubicles or rooms that will withstand the effects of both flooding and missiles is acceptable.

c. The essential portions of the system are protected from the effects of high and moderate energy line breaks in accordance with Branch Technical Position APCSB 3-1. Layout drawings are reviewed to assue that no high or moderate energy piping systems are close to essential portions of the AFS, or that protection from the effects of failure will be provided. The means of providing such protection will generally be given in Section 3.6 of the SAR.

d. Essential components and subsystems necessary for safe shutdown can function as required in the event of loss of offsite power. The SAR is reviewed to see that for each AFS component or subsystem affected by the loss of offsite power, system flow and heat transfer capability meet minimum requirements. Statements in the SAR and the results of failure modes and effects analyses are considered in assuring that the system meets these requirements. (CP)

e. The system is designed with adequate redundancy to accommodate a single active component failure without loss of function.

f. Diversity in pump motive power sources and essential instrumentation and control power sources has been provided, in accordance with guidelines of Branch Technical Position APCSB 10-1.

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 auxiliary feedwater system includes all components and equipment from the condensate storage tank (normal operation) or the seismic Category I emergency water supply (including valves and cross-connections) to the connection with the steam generators. The scope of review of the auxiliary feedwater system for the ________________ plant included layout drawings, piping and instrumentation diagrams, and descriptive information for the system and the supporting systems that are essential to its operation. [The review has determined the adequacy of the applicant's proposed design criteria and design bases for the auxiliary feedwater system, and system performance requirements for normal, abnormal, and accident conditions. (CP)] [The review has determined that the design of the auxiliary feedwater system and supporting systems is in conformance with the 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 auxiliary feedwater system 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 auxiliary feedwater system 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 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 19, "Control Room."

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. Regulatory Guide 1.26, "Quality Group Classifications and Standards for water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 1.

9. Regulatory Guide 1.29, "Seismic Design Classification," Revision 1.

10. Branch Technical Positions APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to Standard Review Plan 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to Standard Review Plan 3.6.2.

11. Branch Technical Position APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants."

BRANCH TECHNICAL POSITION APCSB 10-1

DESIGN GUIDELINES FOR
AUXILIARY FEEDWATER SYSTEM PUMP DRIVE AND
POWER SUPPLY DIVERSITY FOR PRESSURIZED WATER
REACTOR PLANTS

A. BACKGROUND

Heat removal from pressurized water reactor plants following reactor trip and a loss of offsite power is accomplished by the operation of several systems including the secondary system via the steam relief system. Similar capability is required to mitigate the consequences of certain postulated piping breaks. Such heat removal involves heat transfer from the reactor to the steam generators, resulting in the production of steam which is then released to the atmosphere. In this process it becomes necessary to supply makeup water to the steam generators. This is accomplished by the use of an auxiliary feedwater system, which generally consists of redundant components that are powered by both electrical and steam-driven sources.

The auxiliary feedwater system functions as an engineered safety system because it is the only source of makeup water to the steam generators for decay heat removal when the main feedwater system becomes inoperable. It must, therefore, be designed to operate when needed, using the principles of redundancy and diversity in order to assure that it can function under postulated accident conditions. The majority of current systems are powered by electrical or steam-driven sources. Operating experience demonstrates that each type of motive power can be subject to a failure of the driving component itself, its source of energy, or the associated control system. The effects of such failures can be minimized by the utilization of diverse systems that include energy sources of at least two different and distinct types.

The provision of several independent flow paths for the auxiliary feedwater system serves to preclude the possibility of a complete loss of function due to a single event, either occurring alone, or in conjunction with the failure of an active component. The auxiliary feedwater system is categorized as a high energy system, because either that section of line which connects to the main feedwater piping or the steam generator is pressurized during plant operation or else the entire system is pressurized when in use during startup, hot standby, and shutdown.

The staff believes that it is necessary to establish design guidelines for the auxiliary feedwater system, and in this regard has developed guidelines that may be used to select the minimum diversity acceptable for auxiliary feedwater system pump drives and power supplies.

B. BRANCH TECHNICAL POSITION

1. The auxiliary feedwater system should consist of at least two full-capacity, independent systems that include diverse power sources.

2. Other powered components of the auxiliary feedwater system should also use the concept of separate and multiple sources of motive energy. An example of the required diversity would be two separate auxiliary feedwater trains, each capable of removing the afterheat load of the reactor system, having one separate train powered from either of two a-c sources and the other train wholly powered by steam and d-c electric power.

3. The piping arrangement, both intake and discharge, for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. This arrangement should take into account pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One arrangement that would be acceptable is crossover piping containing valves that can be operated by remote manual control from the control room, using the power diversity principle for the valve operators and actuation systems.

4. The auxiliary feedwater system should be designed with suitable redundancy to offset the consequences of any single active component failure; however, each train need not contain redundant active components.

5. When considering a high energy line break, the system should be so arranged as to permit the capability of supplying necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure.

C. REFERENCES

None

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SECTION 11.1 SOURCE TERMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - None

I. AREAS OF REVIEW

ETSB reviews the sources of radioactivity that: (1) are input to the radioactive waste management systems employed for treatment of liquid, gaseous, and solid wastes, and (2) are used as the design bases for shielding and building ventilation systems.

1. Review of radioactive source terms includes consideration of parameters used to determine the concentration of each isotope in the reactor coolant; fraction of fission product activity released to the reactor coolant; concentrations of all non-fission product radioactive isotopes in the reactor coolant; leakage rates and associated fluid activity for all potentially radioactive water and steam systems; and potential sources of radioactive materials in effluents that are not considered in the applicant's safety analysis report (SAR) Section 11.2, "Liquid Waste Management Systems," and SAR Section 11.3, "Gaseous Waste Management Systems." The following release points are considered in the evaluations of effluent releases:

   a. Boiling water reactor (BWR) gaseous wastes (noble gases, radioiodine, and particulates), consisting of offgases from the main condenser vacuum system, offgases from the gland seal condenser, steam and liquid leakage to containment, radwaste, turbine, and auxiliary buildings, and ventilation air from buildings having the potential for containing radioactive materials.

   b. BWR liquid wastes, consisting of leakage to equipment and floor drains from buildings housing equipment and components that may contain radioactive fluids; contaminated liquids produced by plant operations, such as demineralizer regenerants and resin sluice water, filter backwashes, ultrasonic resin cleaning rinses, decontamination solutions, and laboratory samples and rinses; and detergent wastes.

   c. Pressurized water reactor (PWR) gaseous wastes (noble gases, radioiodine, and particulates), consisting of offgases from the steam generator blowdown flash tank; offgases from the main condenser vacuum system; leakage to containment, auxiliary, and turbine buildings; stripping of noble gases from the primary coolant during normal operation and at shutdown; and cover and vent gases from tanks and equipment containing radioactive material.

d. PWR liquid wastes, consisting of primary coolant processed to control boron concentration (shim bleed); leakage collected in equipment and floor drains from buildings housing equipment and components that may contain radioactive fluids; steam generator blowdown; condensate demineralizer regenerant solutions; contaminated liquids from anticipated plant operations such as resin sluices, filter backwashes, decontamination solutions, and sample station drains; and detergent wastes.

2. The review of the radioactive material source terms used as the design bases for shielding and building ventilation systems includes, in addition to the applicable information from 1 above, the following areas:

   a. The anticipated airborne radioactive concentrations generated during normal operation, purging, and refueling, together with the models and assumptions (leakage rates from closed systems) used to obtain these estimated concentrations.

   b. The equipment layout, equipment design, and any special design features which may influence the concentrations of airborne radioactive materials, e.g., individually ventilated equipment cubicles, closed pump and valve seal leak-off systems, covered or ventilated liquid sumps and drains.

3. The calculated releases of radioactive materials in liquid and gaseous effluents will be used in Standard Review Plans (SRP) 11.2 and 11.3 to evaluate the liquid and gaseous waste systems. ETSB will provide source terms to be used to evaluate shielding and occupational radiation exposures under the Chapter 12 standard review plans.

II. ACCEPTANCE CRITERIA

1. ETSB will accept the source terms used as the design basis for expected releases if the following criteria are met:

   a. The parameters used to calculate primary and secondary (PWR) coolant concentrations are consistent with those given in Regulatory Guides 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors," and Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors."

   b. All normal and potential release points of radioactive effluent delineated in Section I are considered.

   c. For each source of liquid and gaseous waste considered in II.1.b, the volumes and radioactivity levels given for normal operation including anticipated operational occurrences are consistent with those given in Regulatory Guides 1.BB and 1.CC.

d. Reduction factors for special design features used to reduce leakage, such as clean sealing steam for valve stems and turbine glands, are consistent with those given in Regulatory Guides 1.BB and 1.CC.

e. Decontamination factors for inplant control measures used to reduce releases to the environment, such as iodine removal systems and high efficiency particulate air (HEPA), filters for building ventilation exhaust systems, are consistent with those given in Regulatory Guides 1.BB and 1.CC.

An acceptable method for satisfying the criteria of Section II.1 consists of using the gaseous and liquid effluent (GALE) computer code and the source term parameters given in Regulatory Guides 1.BB and 1.CC. A complete Fortran listing of the PWR and BWR GALE computer code is given in these Regulatory Guides.

If the applicant's calculational technique or any source term parameter differs from that given in Regulatory Guide 1.BB or 1.CC, ETSB will review the justification for the calculations and parameters used and determine if they are reasonable and are consistent with operating experience.

2. ETSB will accept the source terms used as the design basis for shielding and ventilation exhaust systems if the following criteria are met:

   a. Concentrations of radioactive materials in components and systems are based on the design value of the fraction of the reactor power produced in fuel with cladding defects, i.e., 1% fuel cladding defects for a PWR and an offgas rate of 100 μCi/sec/MWt after 30 minutes delay for a BWR.

   b. Concentrations of airborne radioactive materials to be controlled by ventilation exhaust systems are based on design leakage values for equipment and components.

   c. Decontamination factors for internal cleanup systems are consistent with the values given in Regulatory Guides 1.BB and 1.CC.

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 review of the mathematical models and parameters given in the SAR to calculate primary coolant concentrations, and of the leakage rates to the radioactive waste management systems, ETSB compares parameters and calculations given in the SAR with the models and parameters given in Regulatory Guides 1.BB and 1.CC. If the SAR includes models or parameters to estimate primary coolant concentrations and leakage rates that differ from these guides, the parameters and calculations used need to be substantiated. The preferred method of substantiation is by presentation of operating data from similar reactors.

2. ETSB performs an independent calculation of the primary and secondary (PWR) coolant concentrations and of the release rates of radioactive materials using the GALE Computer Code and the "Principal Parameters for Source Term Calculations" given in Regulatory Guides 1.BB and 1.CC.

3. In the calculation, ETSB will use the applicant's values as given in the SAR for the following parameters: design core thermal power level, steam flow rate, coolant mass, and coolant purification rates.

4. ETSB will use the primary coolant concentrations and leakage rates calculated above as inputs for evaluation of the liquid waste system, under SRP 11.2, and the gaseous waste systems, under SRP 11.3, to determine if the radioactive waste management systems meets the dose design objectives of Appendix I to 10 CFR Part 50.

5. As design parameters for shielding and for building ventilation systems to be used in SAR Chapter 12, ETSB will base its source terms for PWRs on leakage from fuel rods producing 1% of the reactor power, and for BWRs, on an offgas rate of 100 μCi/sec/MWt at 30 minutes delay.

6. The ETSB source term calculations are used for both the review of the SAR and for the staff's Environmental Impact Statement.

IV. EVALUATION FINDINGS

The ETSB summary statement on the acceptability of source terms used as design parameters for the waste management systems will be made under SRP 11.2, "Liquid Waste Management Systems," and 11.3, "Gaseous Waste Management Systems."

V. REFERENCES

1. 10 CFR Part 20, Appendix B, "Concentrations in Air and Water Above Natural Background."

2. 10 CFR Part 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, May 5, 1975.

3. Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)."

4. Regulatory Guide 1.CC, "Calcuation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWRs)."

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SECTION 11.2 LIQUID WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Radiological Assessment Branch (RAB)
Structural Engineering Branch (SEB).

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the information in the applicant's preliminary safety analysis report (PSAR) in the specific areas that follow. During the operating license (OL) stage of review, 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 liquid radwaste treatment system design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The ETSB review will include the system piping and instrumentation diagrams (P&IDs), and process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

2. Equipment design capacities, expected flow and radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup time. The system design capacity relative to the design and expected input flows, and the period of time the system is required to be in service to process normal waste flows. The availability of standby equipment, alternate processing routes, and interconnections between subsystems. This information is used in the ETSB review to evaluate the overall system capability to meet anticipated demands imposed by major processing equipment downtime and waste volume surges due to anticipated operational occurrences.

3. The quality group classifications of piping, and equipment, and the bases governing the design criteria chosen. Provisions to prevent, control and collect releases of radioactive material in liquids due to tank overflows from all plant systems, outside reactor containment having the potential to incur such releases. Design and expected temperatures and pressures, and materials of construction of the components of the liquid waste management system.

4. Design provisions incorporated in the equipment and facility design to reduce leakage and facilitate operation and maintenance in accordance with the guidelines of Branch Technical Position, ETSB 11-1 (Revision 1).

5. Special design features that would reduce liquid input volumes or discharge of radioactive material in liquid effluents. Special design features, topical reports incorporated by reference, and data obtained from previous experience with similar systems which are submitted with the SAR.

6. The technical specifications proposed by the applicant for process and effluent control will be reviewed at the operating license stage (FSAR).

Design provisions incorporated to sample and monitor radioactive materials in liquid process and effluent streams are reviewed under Standard Review Plan (SRP) 11.5.

RAB will provide calculated doses based on the ETSB liquid source terms for inclusion in the staff's Environmental Impact Statement and Safety Evaluation Report.

SEB evaluates the applicant's proposed seismic design classification of structures housing the liquid radwaste system.

The consequences of liquid tank failures having the potential to release radioactive liquids are evaluated in SRP 15.7.3.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The liquid radwaste treatment system should have the capability to meet the requirements specified in 10 CFR Parts 20 and 50 and the dose design objectives specified in Appendix I to 10 CFR Part 50, including provisions to treat liquid radioactive waste such that:

   a. The calculated annual total quantity of all radioactive material released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

   b. In addition to a. above, the liquid radwaste treatment systems should include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor.

   c. The concentrations of radioactive materials in liquid effluents released to an unrestricted area should not exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

2. The liquid radwaste treatment system should be designed to meet the anticipated processing requirements of the station. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. ETSB will accept systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences. To meet these processing demands, ETSB will consider interconnections between subsystems, redundant equipment, and reserve storage capacity.

3. The seismic design classification of structures housing liquid radwaste systems, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1) attached to this plan.

4. ETSB will accept system designs that contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1).

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. In the ETSB review of the liquid waste treatment system, the P&IDs and system process flow diagrams are reviewed to determine all sources of liquid input volumes, the points of collection of liquid waste, the flow paths of liquids through the system including all bypasses, the treatment provided, and the points of release of liquid effluents to the environment. This information is used to calculate the quantity of radioactive materials released annually in liquid effluents during normal operation, including anticipated operational occurrences, using the parameters given, the GALE Code, and calculational techniques given in Regulatory Guides 1.BB and 1.CC. A complete Fortran listing of the GALE computer code is given in these Regulatory Guides. The results of this calculation will be used to determine whether the proposed treatment system design meets the acceptance criterion of 11.1.c. Compliance with the acceptance criteria given in Section II.1.a concerning exposures to the total body or critical organ of an individual in an unrestricted area will be determined based on RAB dose calculations using the ETSB-calculated source term.

   Compliance with the acceptance criterion given in II.1.b concerning the cost-benefit analysis will be determined based on RAB man-rem dose calculations in conjunction with ETSB cost-benefit studies.

2. The ETSB review of the liquid waste treatment system design capacity will encompass three major areas:

a. The system capability to process wastes in the event of a single major equipment item failure, e.g., an evaporator outage.

b. The system capability to accept additional wastes during operations which result in excessive liquid waste generation.

c. The system capability to process wastes at design basis fission product leakage levels, i.e., from 1% of the fuel producing power in a PWR or, in a BWR, consistent with a noble gas release of 100 μCi/sec/MWt measured after 30 minutes delay.

ETSB will compare the average input flows to the design flows to determine the fraction of time individual subsystems must be online to process normal liquid waste inputs. ETSB will review the operational flexibility designed into the system, i.e., cross connections between subsystems, redundant or reserve processing equipment, and reserve storage capacity. Based on the usage factors and operational flexibilities, ETSB will evaluate the overall system capability to process wastes in the event of (a), (b), or (c), above, by comparing the design flows to the potential process routes and equipment capacities. ETSB will assume evaporators are unavailable for 2 consecutive days per week for maintenance. If two days holdup capacity or an alternative evaporator are not available for the process stream, ETSB will assume the stream is processed by an alternate route or discharged to the environment, consistent with the guidelines of Regulatory Guides 1.BB and 1.CC.

3. ETSB compares the quality group classification for the liquid radwaste systems and the seismic design for the structures housing the systems with the guidelines of BTP ETSB 11-1 (Rev. 1). ETSB assures that the design includes provisions to prevent and collect leakage due to overflows and spillage from indoor and outdoor storage tanks, in conformance with the guidelines of BTP ETSB 11-1 (Rev. 1).

4. ETSB compares the system design, system and building layout, equipment design, method of operation, and provisions to reduce leakage and facilitate operations and maintenance with the guidelines of BTP ETSB 11-1 (Rev. 1). ETSB will evaluate special design features provided to control leakage from system components and topical reports on systems designs on a case-by-case basis.

5. ETSB reviews the technical specifications proposed by the applicant for process and effluent control (OL). The reviewer will determine that the content and intent of the technical specifications are in agreement with the requirements developed as a result of the staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 liquid radwaste treatment systems include the equipment and instrumentation to control the release of radioactive materials in liquid effluents."

In our evaluation, we have considered releases of radioactive materials in liquid effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant and have determined that for each reactor on the ____________ site the release of radioactive materials in liquid effluents will not result in an annual dose or dose commitment to any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body and 10 millirems to any organ.

We have also considered the potential effectiveness of augmenting the proposed liquid radwaste treatment systems using items of reasonably demonstrated technology and have determined that further effluent treatment will not effect reductions in the cumulative population dose reasonably expected within a 50 mile radius of the reactor at a cost of less than $1000 per man rem or man-thyroid-rem.

We have also considered the potential consequences resulting from reactor operation, and we have determined the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

We have considered the capabilities of the proposed liquid radwaste treatment system to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We have reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design classification applied to structures housing these systems. The design of the systems and structures housing these systems meet the acceptance criteria as set forth in Branch Technical Position, ETSB 11-1 (Rev. 1).

We have reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids due to inadvertent tank overflows and conclude that the measures proposed by the applicant are consistent with our acceptance criteria as set forth in Branch Technical Position, ETSB 11-1 (Rev. 1).

Based on the foregoing evaluation, we conclude that the proposed liquid radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's design, design criteria, and design bases for the liquid radioactive waste treatment systems to the Commission's Regulations and to applicable guides, as referenced above, as well as staff technical positions and industry standards.

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation," and Appendix B, "Concentration in Air and Water Above Natural Background."

2. 10 CFR § 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents - Nuclear Power Reactors."

3. 10 CFR § 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, Licensing and Regulatory Policy and Procedures For Environmental Protection.

6. 10 CFR Part 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as Practicable" for radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.

7. 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.

8. Regulatory Guide 1.BB, "Calcuation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)."

9. Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWRs)."

10. Branch Technical Position ETSB 11-1 (Rev. 1), "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," attached to this plan.

Design Guidance for Radioactive Waste Management Systems
Installed in Light-Water-Cooled Nuclear Power Reactor Plants

A. Background

An aspect of nuclear power plant operation is the control and management of liquid, gaseous and solid radioactive waste[1/] generated as a byproduct of nuclear power. We have established acceptable design guidance, seismic and quality group classifications, and quality assurance provisions for radioactive waste management systems including steam generator blowdown systems. For the purpose of this position paper, the radioactive waste management systems are considered to begin at the interface valves(s) in each line from other systems provided for collecting wastes that may contain radioactive materials and to terminate at the point of controlled discharge to the environment, at the point of recycle back to storage for reuse in the reactor, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground. The steam generator blowdown system begins at, but does not include, the outermost containment isolation valve on the blowdown line and terminates at the point of controlled discharge to the environment, at the point of interface with other liquid waste systems, or at the point of recycle back to the secondary system.

Except as noted below the positions set forth in this paper do not apply to the reactor coolant cleanup system, the condensate cleanup system, the chemical and volume control system, sumps and floor drains provided for collecting liquid wastes, the boron recovery system, building ventilation systems (heating, ventilating and air conditioning) and chemical fume hood exhaust systems. Positions set forth in this paper regarding provisions to control releases of radioactive materials in liquids due to tank overflows apply to all plant systems, outside reactor containment, having the potential to incur such releases.

The design and construction of radioactive waste management and steam generator blowdown systems should provide assurance that radiation exposures to operating personnel and to the general public are maintained at low and acceptable levels, by assuring that these systems are designed to quality standards conducive to increasing system reliability, operability, and availability. In development of this design guidance, the NRC staff has reviewed a number of designs and concepts submitted in license applications and operating system histories. The NRC staff has been guided by current industry practices and the cost of design features, taking in account the potential impact on the health and safety of operating personnel and the general public.

The design guidance given in this position paper provides reasonable assurance that equipment and components used in the radioactive waste management and blowdown systems are designed, constructed, installed and tested on a level commensurate with the health and

[1/] Radioactive waste used in this guide means liquid, gaseous, or solids containing radioactive material resulting from operation of a LWR which by design or operating practice may be or will be processed prior to final disposition.

safety of the public and plant operating personnel. Instrumentation and controls associated with the waste management and blowdown systems should be designed to a quality commensurate with their intended function.

This position paper sets forth minimum branch requirements and is not intended to prohibit the implementation of other equivalent design codes, standards, or quality assurance measures than those indicated herein.

In addition to the design guidance given for radwaste systems, recommendations are given for provisions to preclude the inadvertent release of radioactive materials in liquids due to spills or overflows from both radwaste and non-radwaste system tanks located inside or outside of plant structures.

B. Branch Technical Position

I. Systems Handling Radioactive Materials in Liquids

a. The liquid radwaste treatment system, including the steam generator blowdown system downstream of the second containment isolation valve should meet the following criteria:

(1) The systems should be designed and tested in accordance with the codes and standards listed in Table I, to include the provisions in (2) below and in Section IV of this position paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Foundations and adjacent walls of structures that house the liquid radwaste system should be designed to the seismic criteria described in Section V to a height sufficient to contain the liquid inventory in the building.

(4) Equipment and components used to collect, process, and store liquid radioactive waste need not be designed to the seismic criteria given in Section V.

b. All tanks located outside reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor storage tanks. The following design features should be included for tanks that may contain radioactive materials:

(1) All tanks, both inside and outside the plant including the condensate storage tank(s) should have provisions to monitor liquid levels and to alarm potential overflow conditions.

(2) All tanks should have overflows, drains, and sample lines should be routed to the liquid radwaste treatment system.[1/]

(3) Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system.[1/]

(4) Outdoor tanks should have a dike or retention pond capable of preventing run-off in the event of a tank overflow and have provisions for sampling collected liquids and routing them to the liquid radwaste treatment system.

II. Gaseous Radioactive Waste (Radwaste) System

a. The gaseous radwaste treatment system, including systems provided for treatment of normal offgas releases from the main condenser vacuum system for a BWR and for the treatment of gases stripped from the primary coolant for a PWR should meet the following criteria:

(1) The systems should be designed and tested in accordance with the codes and standards listed in Table 1, to include the provisions in (2) below and in Section IV of this position paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Those portions of the gaseous radwaste treatment system which by design are intended to store or delay the release of gaseous radioactive waste, including portions of structures housing these systems should be designed to the seismic design criteria given in Section V of this position paper. For systems that normally operate at pressure above 1.5 atmospheres (absolute), this should include 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., waste gas storage tanks in a PWR). For systems that operate near ambient pressure and retain gases on charcoal adsorbers, only the tank elements and the building housing the tanks are included (e.g., charcoal delay tanks in a BWR).

III. Solid Radioactive Waste (Radwaste) System

a. The solid radwaste system consists of slurry waste collection and settling tanks, spent resin storage tanks, phase separators, and tanks, equipment, and components

[1/] Retention by an intermediate sump or drain tank, designed for handling radioactive materials and having provisions for routing to the liquid radwaste system is acceptable.

used to solidify wastes prior to offsite shipment. The solid radwaste handling and treatment system should meet the following criteria:

(1) The system should be designed and tested in accordance with the codes and standards listed in Table 1 to include the provisions in (2) below and in Section IV of this paper.

(2) Materials for pressure retaining components should conform to the requirements of one of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code except that malleable, wrought, or cast iron materials and plastic pipe should not be used. Manufacturer's material certificates of conformance with material specifications may be provided in lieu of certified materials test reports.

(3) Foundations and adjacent walls of structures that house the solid radwaste system should be designed to the seismic criteria given in Section V of this position paper to a height sufficient to contain the liquid inventory in the building.

(4) Equipment and components used to collect, process or store solid radioactive waste need not be designed to seismic criteria referenced above.

IV. Additional Design, Construction, and Testing Criteria

In addition to the requirements inherent in the codes and standards listed in Table 1, the following criteria, as minimum, should be implemented for components and systems considered in this guide.

a. The Quality Assurance provisions described in VI of this guide should be applied.

b. Pressure retaining components of process systems should utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Flanged joints or suitable rapid disconnect fittings should be used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal should not be used except for instrumentation connections where welded connections are not suitable. Process lines should not be less than 3/4-inch. Screwed connections backed up by seal welding, socket welding or mechanical joints may be used on lines 3/4-inch or greater, but less than 2 1/2-inch, nominal size. For lines 2 1/2-inch nominal size and above, pipe welds should be of the butt-joint type. Backing rings should not be used in lines carrying resins or other particulate material. All welding constituting the pressure boundary of pressure retaining components should be performed in accordance with ASME Pressure and Vessel Code Section IX.

c. Completed process systems should be pressure tested to the maximum practicable extent. Piping systems should be hydrostatically tested in their entirety except at atmospheric tank connections where no isolation valves exist. Testing of piping systems should be performed in accordance with applicable ASME or ANSI codes, but in no case less than 75 psig. The test pressure should be held for a minimum of 30 minutes with no leakage indicated. Testing provisions should be incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

V. Seismic Design Requirements for Radioactive Waste Management Systems and Structures Housing Radioactive Waste Management Systems

a. Seismic Design Requirements Gaseous Radioactive Waste Management Systems[1/]

(1) For the evaluation of support elements in the gaseous waste system, a simplified seismic analysis procedure to determine seismic loads may be used. The simplified procedure consists of consideration of the system as a single degree of freedom system and picking up a seismic response value from applicable floor response spectra, once the fundamental frequency of the system is determined. The floor response spectra should be obtained analytically (Section V.b) from the application of Regulatory Guide 1.60 design response spectra normalized to OBE level maximum ground acceleration at the foundation of the building housing the gaseous radwaste system.

(2) The allowable stresses to be used for the system support elements should be those given in the AISC Manual of Steel Construction, 7th edition 1970, including the one-third allowable stress increase provision for load combinations involving earthquake loads. For design of concrete foundations of the system, where applicable, use of the ACI 318-71 code with one-third increase in allowable stress for seismic loads is acceptable.

(3) The construction and inspection requirements for the support elements should comply with those stipulated in AISC or ACI Codes as appropriate.

b. Seismic Design Requirements for Buildings Housing Radwaste Systems

(1) Define input motion at the foundation of the building housing the radwaste systems. The motion should be defined by normalizing the Regulatory Guide 1.60 spectra to the OBE maximum ground acceleration selected for the plant.

[1/] For which seismic capabilities are required in Section II(3).

A simplified analysis should be performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the systems; i.e., an analysis of the building by a "several degrees of freedom" mathematical model and the use of an approximate method to generate the floor response spectra for radwaste systems and the seismic loads for the buildings. No time history or dynamic analysis is required.

(2) The simplified method for determination of seismic loads for the building consists of (a) calculation of first several modal frequencies and participation factors for the building, (b) determination of modal seismic loads by item (1) input spectra, and (c) combination of modal seismic loads by the square root of the sum of squares (SRSS) rule.

(3) With regard to generation of floor response spectra for radwaste systems, methods such as the Biggs or other equivalent procedures which give approximate floor response spectra without need for performing a time history analysis may be used.

(4) The load factors and load combinations to be used for the building should be those given in the ACI-318-71 Code. The allowable stresses for steel components should be those given in the AISC Manual of Steel Construction, 7th edition, 1970.

(5) The construction and inspection requirements for the building elements should comply with those stipulated in the AISC or ACI Code as appropriate.

(6) The foundation media of structures housing the radwaste systems should not liquify during the Operating Basis Earthquake.

c. In lieu of the requirements and procedures defined above, optional shield structures constructed around and supporting the radwaste systems may be erected to protect the radwaste systems from effects of housing structural failure. If this option is adopted, the procedures described in Section V.b only need to be applied to the shield structures while treating the rest of the housing structures as non-seismic Category I.

VI. Quality Assurance for Radioactive Waste Management Systems

A quality assurance program should be established that is sufficient to assure that the design, construction, and testing requirements are met. The quality assurance program should include the following:

a. Design and Procurement Document Control - Measures should be established to insure that the requirements of this position paper are specified and included in design and procurement documents and that deviations therefrom are controlled.

b. Control of Purchased Material, Equipment and Services - Measures should be established to assure that purchased material, equipment and construction services conform to the procurement documents.

c. Inspection - A program for inspection of activities affecting quality should be established and executed by, or for, the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

d. Handling, Storage, and Shipping - Measures should be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

e. Inspection, Test and Operating Status - Measures should be established to provide for the identification of items which have satisfactorily passed required inspections and tests.

f. Corrective Action - Measures should be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances are promptly identified and corrected.

TABLE 1

EQUIPMENT CODES

EQUIPMENT	CODES: Design and Fabrication	Materials[2]	Welder Qualifications and Procedure	Inspection And Testing
Pressure Vessels	ASME Code Section VIII, Div. 1	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII, Div. 1
Atmospheric or 0-15 psig tanks	ASME Code[3] Section III, Class 3, or API 620 & 650, AWWA D-100	ASME Code[4] Section II	ASME Code Section IX	ASME Code[3] Section III, Class 3 or API 620; 650 AWWA D-100
Heat Exchanger	ASME Code Section VIII, Div. 1 and TEMA	ASME Code Section III	ASME Code Section IX	ASME Code Section VIII, Div. 1
Piping and Valves	ANSI 31.1	ASTM or ASME Code Section II	ASME Code Section IX	ANSI B 31.1
Pumps	Manufacturer's[1] Standards	ASME Code Section II or Manufacturer's Standard	ASME Code Section IX (as required)	ASME[3] Section III Class 3; or Hydraulic Institute

Notes:

(1) Manufacturer's standard for the intended service. Hydrotesting should be 1.5 times the design pressure.

(2) Material Manufacturer's certified test reports should be obtained whenever possible.

(3) ASME Code Stamp and material traceability not required.

(4) Fiberglass reinforced plastic tanks may be used in accordance with Part M, Section 10, ASME Boiler and Pressure Vessel Code, for applications at ambient temperature.

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Radiological Assessment Branch (RAB)
- Accident Analysis Branch (AAB)
- Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

At the construction permit (CP) stage of review, ETSB reviews the information in the applicants' safety analysis report (SAR) in the specific areas that follow. At the operating license (OL) stage of review, 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 gaseous radwaste treatment and ventilation system designs, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials (noble gases, radioiodine, and particulates) in gaseous effluents. The ETSB review will include the system piping and instrumentation diagrams (P&ID's), and the process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

2. Equipment and ventilation system design capacities, expected flows and radionuclide concentrations, expected decontamination factors for radionuclides, and available holdup time. The system design capacity relative to the design and expected input flows, the period of time the system is required to be in service to process normal waste flows, availability of standby equipment, alternate processing routes, and interconnections between subsystems. This information is used to evaluate the overall system capability to meet anticipated demands imposed by major processing equipment downtime and waste volume surges due to anticipated operational occurrences.

3. The quality group classifications of piping and equipment, and the bases governing the design criteria chosen. Design features to preclude the possibility of an explosion if the potential for explosive mixtures exist. Design and expected temperatures and pressures, and materials of construction of the components of the system.

4. Design provisions incorporated in the equipment and facility design to facilitate operation and maintenance in conformance with the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1).

5. Special design features to reduce leakage of gaseous waste or discharge of radioactive material in gaseous effluents. Special design features, topical reports incorporated by reference, and data obtained from previous experience with similar systems which are submitted with the SAR.

6. The technical specifications proposed by the applicant for process and effluent control are reviewed at the operating license stage (FSAR).

Design provisions incorporated to sample and monitor radioactive materials in gaseous process and effluent streams are reviewed under SRP 11.5.

RAB provides calculated doses based on the ETSB gaseous source term for inclusion in the staff's Environmental Impact Statement and Safety Evaluation Report.

SEB will provide an evaluation of the applicant's proposed seismic design classification of the gaseous waste processing system, of structures housing this system and the required seismic analysis for inclusion in the staff's Safety Evaluation Report.

AAB calculates the doses that result as the consequence of a gas decay tank rupture under SRP 15.7.1.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. The gaseous radwaste treatment system should have the capability to meet the requirements of 10 CFR Part 20 and the dose design objectives specified in Appendix I to 10 CFR Part 50, including provisions to treat gaseous radioactive wastes such that:

   a. The calculated annual total quantity of all radioactive material released from each reactor at the site to the atmosphere will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

   b. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form released from each reactor at the site in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

c. In addition to a. and b., above, the gaseous radwaste treatment systems should include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor.

d. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area should not exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 1.

2. The gaseous radwaste treatment system should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. ETSB will accept systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences. To meet these processing demands, ETSB will consider shared systems, redundant equipment, and reserve storage capacity.

3. The seismic design and quality group classification of components used in the gaseous waste treatment system and structures housing this system should conform to the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1) attached to Standard Review Plan 11.2. The gaseous waste handling and treatment system should be designed to withstand the effects of an explosion, if the potential for an explosive mixture exists. Instrumentation with automatic alarm and control functions should be provided to monitor the concentrations of the appropriate gas in portions of the systems having the potential for containing explosive mixtures. The design should include precautions to stop continuous leakage paths, i.e., to provide liquid seals downstream of rupture discs and to prevent permanent loss of the liquid seals in the event of an explosion.

4. ETSB will accept system designs that contain provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1) attached to this plan.

5. For HEPA filters and charcoal adsorbers installed in normal ventilation exhaust systems, ETSB will use the guidelines in Branch Technical Position ETSB 11-2 (attached to this SRP).

   Decontamination factors for iodine different from those specified in BTP 11-2 are used for design purposes. They should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The effects of aging and poisoning by airborne contaminants should also be supported by test data.

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 of the gaseous waste treatment system, the P&ID's and system process flow diagrams are reviewed to determine all sources of gaseous waste, the points of collection of gaseous wastes, the flow paths of gases through the systems, including all bypasses, the treatment provided and the points of release of gaseous effluents to the environment. This information is used to calculate the quantity of radioactive material (noble gases, radioiodine, and particulates) released annually in gaseous effluents during normal operations, including anticipated operational occurrences, using the given parameters, the GALE Code, and the calculational techniques given in Regulatory Guides 1.BB and 1.CC. A complete Fortran listing of the GALE computer code is given in these Regulatory Guides. The results of this calculation will be used to determine whether the proposed treatment system design meets the acceptance criterion of II.1.d. Compliance with the acceptance criteria of II.1.a, and b, concerning exposures of the total body, skin, and thyroid will be determined based on RAB dose calculations using the ETSB-calculated source term. Compliance with the acceptance criterion given in II.1.c concerning the cost-benefit analysis will be determined based on RAB man-rem dose calculations in conjunction with ETSB cost-benefit studies.

2. The ETSB review of the gaseous waste treatment system design capacity will encompass two major areas:

   a. The capability of the system to process gaseous wastes in the event of a single major equipment item failure. For non-redundant equipment or components, ETSB will assume a 3-week downtime every other year (10 days per year average).

   b. The capability of the system to process gaseous wastes at design basis fission product levels, i.e., from 1% of the fuel producing power in a PWR or, in a BWR, consistent with a noble gas release rate of 100 μCi/sec/MWt at 30 minutes delay.

   ETSB will review the operational flexibilities designed into the system, e.g., cross connections between subsystems, redundant or reserve processing equipment, and reserve storage capacity.

   In the evaluation of charcoal delay systems for radioactive gas decay, ETSB considers the bed dimensions, mass of charcoal, flow rate, temperatures, pressures, humidity, and dynamic adsorption coefficients to calculate the effective holdup times.

3. ETSB compares the quality group classification of the gaseous waste system, and the seismic design category of both gaseous waste systems and the structures housing these systems with the guidelines of BTP ETSB 11-1 (Rev. 1). Compliance with the acceptance criteria given in Section 11.3 concerning seismic design classification of equipment and of structures housing the gaseous radwaste treatment systems will be determined

based on SEB evaluation of the applicant's seismic analysis. If there is a potential that explosive mixtures may exist, ETSB will determine, using the system description and P&ID's whether the applicant has designed the gaseous radwaste treatment system to withstand the effects of such an explosion, or has provided redundant instrumentation to annunciate and prevent the buildup of potentially explosive mixtures. ETSB also determines if the applicant's design includes adequate provisions to stop continuous leakage paths after an explosion. The areas of concern are (1) streams where water decomposition gases (hydrogen and oxygen) exist in a BWR, (2) cover gas streams where air inleakage can occur in a PWR, and (3) where there is a possibility of liquid hydrocarbons and ozone collecting in a cryogenic distillation system.

4. ETSB will compare the system design, system layout, equipment design, method of operation, and provisions to reduce leakage and to facilitate operations and maintenance to the guidelines of BTP ETSB 11-1 (Rev. 1). ETSB will evaluate special design features provided to control leakage from system components and topical reports on system designs on a case-by-case basis.

5. ETSB will compare the design of HEPA filters and charcoal adsorbers in filtration systems with the guidelines of II.5, above.

6. At the OL stage ETSB will review the technical specifications proposed by the applicant for process and effluent control. The reviewer will determine 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 review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 gaseous radwaste treatment system includes the equipment and instruments to control the release of radioactive materials in gaseous effluents."

In our evaluation, we have considered radionuclide releases of radioactive material (noble gases, radioiodine and particulates) in gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant and have determined that for each reactor on the (______________) site, the release of radioactive materials in gaseous effients will not result in an annual external dose to any individual in an unrestricted area in excess of 5 mrem to the whole body and 15 mrem to the skin, and the release of all radioactive iodine and radioactive material in particulate form will not result in an annual dose or dose commitment to an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem to any organ.

We have also considered the potential effectiveness of augmenting the proposed gaseous radwaste treatment systems using items of reasonably demonstrated technology and have determined that "further effluent treatment will not effect reductions in the cumulative population dose within a 50 mile radius of the reactor at a cost of less than $1,000 per man-rem or $1,000 per man-thyroid-rem."

We have also considered the potential consequences resulting from reactor operation with "1% of the operating fission product inventory in the core being released to the primary coolant" or "a fission product release rate consistent with a noble gas release rate to the reactor coolant of 100 μCi/MWt-sec at 30 minutes decay" and determined that under these conditions, the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20, Appendix B, Table II, Column 1.

We have considered the capabilities of the proposed gaseous radwaste treatment systems to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We have reviewed the applicant's quality assurance provisions for the gaseous radwaste systems, the quality group classifications used for system components, the seismic classification applied to the design of the system, and of structures housing the radwaste systems. The design of the system and structures housing these systems meet the acceptance criteria as set forth in Branch Technical Position, ETSB 11-1 (Rev. 1).

We have reviewed the provisions incorporated in the applicant's design to control releases due to hydrogen explosions in the gaseous radwaste system and conclude that the measures proposed by the applicant are consistent with our acceptance criteria as set forth in Branch Technical Position ETSB 11-1 (Rev. 1).

Based on the foregoing evaluation, we conclude that the proposed gaseous radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the gaseous radwaste treatment system to the applicable regulations and guides referenced above, as well as to the staff technical positions and industry standards.

V. REFERENCES

1. 10 CFR Part 20, Appendix B, "Concentration in Air and Water Above Natural Background."

2. 10 CFR §50.34a, "Design Objective for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Reactors."

3. 10 CFR §50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

6. 10 CFR Part 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

7. 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.

8. Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWR's)."

9. Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWR's)."

10. Branch Technical Position ETSB 11-1 (Rev. 1), "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," attached to Standard Review Plan 11.2.

11. "Design, Construction, and Testing of High Efficient Air Filtration Systems for Nuclear Applications," ORNL-NSIC-65, Oak Ridge National Laboratory (1970).

12. Branch Technical Position, ETSB 11-2, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

BRANCH TECHNICAL POSITION - ETSB NO. 11-2

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER REACTOR PLANTS

A. BACKGROUND

Particulate filtration and radioiodine absorption units are included in the design of the ventilation exhaust systems of light-water-cooled nuclear power plants to reduce the quantities of gaseous radioactivity released from building or containment atmospheres during normal operation including anticipated operational occurrences.[1/] All such cleanup systems should be designed to operate continuously under the normal environmental conditions. In this position paper, cleanup systems that should operate to meet the "as low as practicable" guidelines of Appendix I to 10 CFR Part 50 inside the primary containment (recirculating units) are designated as primary systems. Primary systems generally include a containment cleanup system (kidney filtration system). Systems that operate outside primary containment are designated as secondary systems. Secondary systems generally include cleanup systems installed in the ventilation exhaust systems for the reactor building, turbine building, radwaste building, auxiliary building, mechanical vacuum pump, main condenser air ejector, and any other release points that may contain particulates and gaseous radioiodine species. In some instances, filtration equipment installed in a post-accident hydrogen purge exhaust system may be designed to the recommendations of this guide, e.g., where a removal efficiency of 90% or less for radioiodine species is sufficient for the hydrogen purge exhaust system when the sum of the calculated loss-of-coolant accident (LOCA) dose and the post-LOCA hydrogen purge dose is less than the limits of 10 CFR Part 100.

Normal environmental conditons that these atmosphere cleanup systems should withstand are inlet concentrations of radioactive iodine in the range of 10^{-6} to 10^{-13} μCi/cc, relative humidity of the influent stream up to 100%, temperatures of the influent stream up to 125°F, and atmospheric pressure. Radiation levels of airborne radioactive material and radioiodine buildup on the adsorber are not significant to the operation of the filter system or any component.

An atmosphere cleanup system installed in a normal ventilation exhaust system consists of some or all of the following components; heaters or cooling coils, used in conjunction with heaters, prefilters, high-efficiency particulate air (HEPA) filters, adsorption units, fans, and associated ductwork, dampers, and instrumentation. Heaters are designed to mix and heat the incoming stream to reduce its relative humidity before it reaches the filters and absorbers.

1/ This position paper applies only to atmosphere cleanup systems designed to collect airborne radioactive materials during normal plant operation including anticipated operational occurrences, and addresses the atmosphere cleanup systems, including the various components and ductwork in the normal operating environment. This guide does not apply to engineered safety feature (ESF) atmosphere cleanup systems that are designed to mitigate the consequences of postulated accidents.

consist of the following sequential components: (1) provisions for moisture removal (heaters or cooling coils used in conjunction with heaters), (2) HEPA filters before the adsorbers, (3) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as metal zeolites), (4) ducts and dampers, (5) fans, and (6) related instrumentation. If it is desired to reduce the particulate load on the HEPA filters and extend their service life, the installation of prefilters upstream of the initial HEPA bank is suggested. Consideration should also be given to the installation of a HEPA filter bank downstream of carbon adsorbers to retain carbon fines.

b. The volumetric air flow rate of a single cleanup train should be limited to approximately 30,000 cfm. If total system air flow in excess of this rate is required, multiple trains should be used. For ease of maintenance, a filter layout consisting of three HEPA filters high and ten wide is preferred.

c. Each atmosphere cleanup system should be locally instrumented to monitor and alarm pertinent pressure drops and flow rates in accordance with the recommendations of Section 4.8.1 of draft standard ANSI N509 (Ref. 1).

d. The power supply, instrumentation and equipment controls, and electrical distribution system for the atmosphere cleanup systems described in Section B.2.a above should be designed, qualified, and tested in accordance with ANSI CI-1971, "National Electrical Code" (Ref. 4).

e. To maintain the radiation exposure to operating personnel as low as practicable during plant maintenance, atmosphere cleanup systems should be designed to facilitate maintenance in accordance with the guidelines of Section C.3 of Regulatory Guide 8.8 (Ref. 5). The atmosphere cleanup train should be totally enclosed.

f. Outdoor air intake openings should be equipped with louvers, grills, screens, or similar protective devices to minimize the effects of high winds, rain, snow, ice, trash, and other contaminants on the operation of the system. If the atmosphere surrounding the plant has a potential for containing significant environmental contaminants, such as dusts and residues from smoke cleanup systems from adjacent coal burning power plants or industry, these contaminants should be considered in the design of the system and prevented from affecting the operation of any atmosphere cleanup system.

g. Atmosphere cleanup system housings and ductwork should be designed to exhibit on test a maximum total leakage rate as defined by Section 4.12 of draft standard ANSI N509 (Ref. 1). Duct and housing leak tests should be performed in accordance with the recommendations of Section 6 of draft standard ANSI N510 (Ref. 2).

3. Component Design Criteria and Qualification Testing

a. Adsorption units function efficiently at a relative humidity of 70% or less. If the incoming atmosphere is expected to be greater than 70% during normal reactor

HEPA filters are installed to remove particulate matter, which may be radioactive, and pass the air stream to the adsorber. The adsorber removes gaseous iodine (elemental iodine and organic iodides) from the air stream. HEPA filters downstream of the adsorber units collect carbon fines. The fan is the final item in an atmosphere cleanup system. Consideration should be given to installing prefilters upstream of the HEPA filters to reduce the particulate load and extend their service life.

The environmental history will affect the performance of the atmosphere cleanup system. Industrial contaminants, pollutants, temperature, and relative humidity contribute to the aging and weathering of filters and adsorbents and reduce their effectiveness to perform their required function. Therefore, aging, weathering and poisoning of these components, which may vary from site to site, must be considered during design and operation. Average temperature and relative humidity also vary from site to site, and the potential buildup of moisture in the adsorber warrants equal design consideration. The effects of these factors on the atmosphere cleanup system should be determined by scheduled testing during operation.

All components of the atmosphere cleanup system installed in normal ventilation exhaust systems should be designed for reliable performance under the expected operating conditions. Initial testing and proper maintenance are primary factors in assuring the reliability of the system by providing ease of maintenance. Of particular importance in the design is a layout that provides accessibility and sufficient working space to safely perform the required functions. Periodic testing during operation to verify the efficiency of the components is another important means of assuring reliability. Built-in features that will facilitate convenient in-place testing are important in system design.

B. <u>BRANCH TECHNICAL POSITION</u>

1. Environmental Design Criteria

   a. The design of each atmosphere cleanup system installed in a normal ventilation exhaust system should be based on the maximum anticipated operating parameters of temperature, pressure, and relative humidity. The cleanup system should be designed based on continuous operation for the expected life of the plant.

   b. If the atmosphere cleanup system is located in an area of high radiation during normal plant operation, adequate shielding of components from the radiation source should be provided.

   c. The operation of any atmosphere cleanup system in a normal ventilation exhaust system should not deleteriously affect the expected operation of any engineered-safety-feature system that must operate after a design basis accident.

   d. The design of the atmosphere cleanup system should consider any significant contaminants such as dusts, chemicals, or other particulate matter than could deleteriously affect the cleanup system's operation.

2. System Design Criteria

   a. Atmosphere cleanup systems installed in normal ventilation exhaust systems need not be redundant nor designed to seismic Category I classification, but should operation, heaters or cooling coils used in conjunction with heaters should be designed to reduce the relative humidity of the incoming atmosphere to 70%. Heaters should be designed in accordance with the recommendations of Section 5.5 of draft standard ANSI N509 (Ref. 1). Consideration should be given in system design to minimizing heater control malfunction. The heater should not be a potential absorbent ignition source.

b. The HEPA filters should be designed in accordance with the recommendations of Section 5.1 of draft standard ANSI N509 (Ref. 1) and should satisfy the requirements of UL-586 (Ref. 6). Each HEPA filter should be tested for penetration of dioctyl phthalate (DOP) in accordance with the recommendations of MIL-F-51068D (Ref. 8) and MIL-STD-282 (Ref. 9).[2/]

c. Component mounting frames should be designed and constructed in accordance with the recommendations of Section 5.6.3 of draft standard ANSI N509 (Ref. 1).

d. Filter and adsorber banks should be arranged in accordance with the recommendations of Section 4.4 of ORNL-NSIC-65 (Ref. 3).

e. Unit filter housings, including floors and doors, and electrical conduits, drains, and piping installed inside filter housings should be designed and constructed in accordance with the recommendations of Section 5.6 of draft standard ANSI N509 (Ref. 1), except the maximum size of any one filter housing should be limited to 30,000 cfm (see Branch Technical Position B.2.c).

f. Ductwork associated with the atmosphere cleanup system should be designed in accordance with the recommendations of Section 5.10 of draft standard ANSI N509 (Ref. 1).

g. The adsorber section of the atmosphere cleanup system may contain any adsorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodides) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this adsorbent is discussed in this paper. Each original or replacement batch of impregnated activated carbon used in the adsorber section should meet the qualification and batch test results summarized in Table 1 of this paper, "Summary Table of New Activated Carbon Physical Properties."

If an adsorbent other than impregnated activated carbon is proposed, or the mesh size distribution is different from the specifications given in Table 1, the proposed adsorbent should be demonstrated to perfrom equal to, or better than activated carbon satisfying all the specifications in Table 1.

[2/] The Energy Research and Development Administration (formerly USAEC) operates a number of Filter Test Facilities qualified to perform HEPA filter efficiency tests. The facilities are listed in the current USERDA Health and Safety Bulletin for Filter Unit Inspection and Testing (Ref. 7).

If impregnated activated carbon is used as the adsorbent, the adsorber section should be designed for an average atmosphere residence time of 0.25 sec per two inches of adsorbent bed.

h. Adsorber cells should be designed, arranged and documented in accordance with the recommendations of Section 5.2 of draft standard ANSI N509 (Ref. 1).

i. The system fan and motors, its mounting, and ductwork connections should be designed and constructed in accordance with the recommendations of Section 5.7 and 5.8 of draft standard ANSI N509 (Ref. 1).

j. The fan or blower used in the atmosphere cleanup system should be capable of operating under the environmental conditions postulated.

k. Ducts and housings should be laid out with a minimum of ledges, protusions, and crevices that can collect dust and moisture and can impede personnel or create a hazard in performance of their work. Straightening vanes should be installed to insure representative air flow measurement and uniform flow distribution through cleanup components.

l. Dampers should be designed, constructed and tested in accordance with the recommendations of Section 5.9 of draft standard ANSI N509 (Ref. 1).

4. Maintenance

a. To maintain radiation exposures to operating personnel as low as practicable, the atmosphere cleanup system should be designed to control leakage and permit maintenance in accordance with the guidelines of Section C.3 of Regulatory Guide 8.8 (Ref. 5).

b. Accessibility of components and maintenance should be considered in the design of atmosphere cleanup systems in accordance with the recommendations of Section 4.7 of draft standard ANSI N509 (Ref. 1).

c. For ease of maintenance, the system design should provide for a minimum of three linear feet from mounting frame to mounting frame between banks of components. If components are to be replaced, the dimension to be provided should be the maximum length of the component plus a minimum of three feet.

d. The system design should provide for permanent test probes with external connections in accordance with the recommendations of Section 4.11 of draft standard ANSI N509 (Ref. 1). Preferably, the test probes should be manifolded at a single convenient location with due consideration given to balancing of line lengths and diameter to produce reliable test results for refrigerant gas, resistence, flow rate, and DOP testing.

e. The cleanup components (e.g., HEPA filters, and adsorbents should be installed after construction is completed.

5. In-Place Testing Criteria

a. The atmosphere cleanup unit should be tested in-place initially and at a frequency not be exceed 18 months thereafter (during a scheduled reactor shutdown is acceptable). A visual inspection of the system and all associated components should be performed before each test in accordance with the recommendations of Section 5 of draft standard ANSI N510 (Ref. 2).

b. The air flow distribution to the HEPA filters and iodine adsorbers should be tested in-place initially, for uniformity and should be within $\pm$20% of the average flow per unit when tested in accordance with the recommendations of Section 9 of "Industrial Ventilation" (Ref. 10) and Section 8 of draft standard ANSI N510 (Ref. 2).

c. The in-place HEPA DOP test should conform to draft standard ANSI N510 (Ref. 2). HEPA filter sections should be tested in-place initially and at a frequency not to exceed 18 months thereafter (during a scheduled reactor shutdown is acceptable). The HEPA filter bank upstream of the adsorber section should also be tested following painting, fire or chemical release in any ventilation zone communicating with the system such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials. DOP penetration tests of all HEPA filter banks should confirm a penetration of less than 0.05% at rated flow. A filtration system satisfying this requirement can be considered to warrant 99% removal efficiency for particulates. HEPA filters that fail to satisfy this requirement should be replaced with filters qualified pursuant to Branch Technical Position B.3.b of this paper. If for any reason, the HEPA filter bank is entirely or only partially replaced, an in-place DOP test should be conducted.

If any welding repairs are necessary on, within, or adjacent to the ducts, housing, or mounting frames, filters and adsorbers should be removed from the housing during such repairs. These repairs should be completed prior to periodic testing, filter inspection, and in-place testing. The use of silicone sealants or any other temporary patching material on filters or mounting frames should not be allowed.

d. The activated carbon adsorber section should be leak tested with a gaseous halogenated hydrocarbon refrigerant in accordance with Section 12 of draft standard ANSI N510 (Ref. 2), to ensure that bypass leakage through the adsorber section is less than 0.05%. During the test the upstream concentration of refrigerant gas should be no greater than 20 ppm. After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm. Adsorber leak testing should be conducted whenever DOP testing is done.

6. Laboratory Testing Criteria for Activated Carbon

a. The activated carbon adsorber section of the atmosphere cleanup system should be assigned the decontamination efficiencies given in Table 2 for radioiodine if the following conditions are met:

(1) The adsorber section meets the requirements given in regulatory position B.5.d of this paper,

(2) New activated carbon meets the physical property requirements given in Table 1, and

(3) Representative samples of used activated carbon meet the laboratory test requirements given in Table 2.

If for any reason the activated carbon fails to meet any of these conditions, it should not be used in adsorption units.

b. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section. Each representative sample should be not less than two inches in both length and diameter and should have the same qualification and batch test characteristics as the system adsorbent. There should be a sufficient number of representative samples located in parallel with the adsorber section for estimating the amount of penetration of the system adsorbent throughout its service life. The design of the samples should be in accordance with the recommendations of Appendix A of draft standard ANSI N509 (Ref. 1). Where the system activated carbon is greater than two inches deep, each representative sampling station should consist of multiple two-inch samples in series whose aggregate length equals the thickness of the system adsorbent. Once representative samples are removed to laboratory test, their positions in the sampling array should be blocked off. Laboratory tests of representative samples should be conducted, as indicated in Table 2 of this paper, with the test gas flow in the same direction as flow during service conditions. Similar laboratory tests should be performed on an adsorbent sample before loading into the adsorbers to establish an initial point for comparison of future test results. The activated carbon adsorber section should be replaced with new unused activated carbon meeting the physical property requirements of Table 1 after the last representative sample has been removed and tested or if any preceding representative sample has failed to meet the testing requirement of Table 2.

TABLE 1

SUMMARY TABLE OF NEW ACTIVATED CARBON PHYSICAL PROPERTIES

BATCH TESTS TO BE PERFORMED ON FINISHED ADSORBENT

	TEST	ACCEPTABLE TEST METHOD	ACCEPTABLE RESULTS
1.	Particle Size Distribution	ASTM D2862 (Ref. 11)	Retained on #6 ASTM E11 (Ref. 12) Sieve: 0.0% Retained on #8 ASTM E11 (Ref. 12) Sieve: 5.0% Maximum Through #8, retained on #12 Sieve: 40% to 60% Through #12, retained on #16 Sieve" 40% to 60% Through #16 ASTM E11 (Ref. 12) Sieve: 5.0% max. Through #18 ASTM E11 (Ref. 12) Sieve: 1.0% max.
2.	Hardness Number	RDT M16-1T, Appendix C (Ref. 13)	95 minimum
3.	Ignition Temperature	RDT M16-1T, Appendix C (Ref. 13)	330°C minimum at 100 fpm
4.	Activity	CCl_4 Activity, RDT M16-1T, Appendix C (Ref. 13)	60 minimum
5.	Radioiodine Removal Efficiency		
	a. Elemental Iodine, 25°C and 95% Relative Humidity	RDT M16-1T (Ref. 13), para. 4.5.1, except 95% relative humidity air is required.	99.5%
	b. Methyl Iodide, 25°C and 95% Relative Humidity	RDT M16-1T (Ref. 13), para. 4.5.3, except 95% relative humidity air is required.	95%
6.	Bulky Density	ASTM D2854 (Reg. 14)	0.38 gm/ml minimum
7.	Impregnant Content	State Procedure	State type (not to exceed 5% by weight)

NOTES:

(1) A batch test is defined as a test made on a production batch of product to establish suitability for a specific application. A batch of activated carbon is defined as that quantity of the same grade, type, and series of material which has been homogenized

NOTES (Cont'd):

to exhibit, within reasonable tolerance, the same performance and physical characteristics; and for which the manufacturer can demonstrate by acceptable tests and quality control practices such uniformity. All material in the same batch shall be activated, impregnated, and otherwise treated under the same process conditions and procedures, in the same process equipment, and shall be produced under the same manufacturing release and instructions. Material produced in the same charge of batch-type equipment shall constitute a batch; material produced in different charges of the same batch-type equipment may be included in the same batch only if it can be homogenized as above. The maximum batch size shall be 350 cu. ft. of activated carbon.

(2) Test 4 should be performed on base material.

TABLE 2

LABORATORY TEST REQUIREMENTS FOR ACTIVATED CARBON

ACTIVATED CARBON[a] BED DEPTH	ASSIGNED ACTIVATED CARBON DECONTAMINATION EFFICIENCIES FOR RADIOIODINE	LABORATORY TEST REQUIREMENTS FOR A REPRESENTATIVE SAMPLE[b]
2 inches. Air filtration system designed to operate inside primary containment	90%	Test, per Test 5b in Table 1, for a methyl iodide penetration of less than 10%.
2 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled at 70%.	70%	Test, per Test 5b in Table 1, at a relative humidity of 70% for a methyl iodide penetration of less than 10%.
4 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled at 70%.	90%	Test, per Test 5b in Table 1, at a relative humidity of 70% for a methyl iodide penetration of less than 1%.
6 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.	99%	Test, per Test 5b in Table 1, at a relative humidity of 70% for a methyl iodide penetration of less than 0.1%.

[a]The activated carbon, when new, should meet the requirements stated in this paper.

[b]See regulatory position C.6.b. for definition of representative sample. Testing should be performed (1) initially, (2) at a frequency not to exceed 18 months (during a scheduled reactor shutdown is acceptable), and (3) following painting, fire or chemical release in any ventilation zone communicating with the system such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials.

[c]Multiple beds, e.g., two 2-inch beds in series, should be treated as single bed of aggregated depth.

REFERENCES

1. Draft Standard ANSI N509 (Draft 7, March 1975), "Nuclear Power Plant Air Cleaning Units and Components," American National Standards Institute.

2. Draft Standard ANSI N510-1975, "Testing of Nuclear Air Cleaning Systems," American National Standards Institute.

3. C. A. Burchsted & A. B. Fuller, "Design, Construction, and Testing of High-Efficiency Air Filtration Systems for Nuclear Application," ORNL-NSIC-65, Oak Ridge National Laboratory, January 1970.

4. ANSI C1-1971, "National Electric Code," American National Standards Institute.

5. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Practicable (Nuclear Reactors)," Office of Standards Development, USNRC.

6. Underwriters' Laboratories Standard UL-586, "High Efficiency, Particulate, Air Filter Units," (also designated ANSI B132.1-1971).

7. USERDA (formerly USAEC) Health and Safety Bulletin, "Filter Unit Inspection and Testing Services."

8. MIL-F-51068D, Military Specification, "Filter, Particulate, High-Efficiency, Fire-Resistant," 4 April 1974.

9. MIL-STD-282, Military Standard, "Filter Units, Protective Clothing, Gas Mask Components and Related Products: "Performance Test Methods," 28 May 1956.

10. "Industrial Ventilation," American Conference of Governmental Industrial Hygienists, 13th Edition, 1974.

11. ASTM D2862-70, "Test for Particle Size Distribution of Granulated Activated Carbon," American Society for Testing and Materials.

12. ASTM E11-70, "Specifications for Wire Cloth Sieves for Testing Purposes."

13. RDT Standard M16-1T, "Gas-Phase Adsorbents for Trapping Radioactive Iodine and Iodine Compounds," USAEC Division for Reactor Development and Technology, October 1973.

14. ASTM D2854-70, "Test for Apparent Density of Activated Carbon."

NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 11.4 SOLID WASTE MANAGEMENT SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

At the construction permit (CP) stage, ETSB reviews the design objectives, criteria, performance objectives, and description of the solid waste system (SWS) as given in the applicant's preliminary safety analysis report (PSAR). During the operating license (OL) stage of review, ETSB confirms the design accepted at the CP stage and evaluates the applicant's technical specifications in these areas.

1. a. The design objectives in terms of expected and design volumes of waste to be processed and handled, the types of waste to be processed (e.g., sludges, resins, evaporator bottoms, and dry material such as contaminated tools, equipment, and clothing), the radionuclide content of the waste, equipment design capacities, and the principal parameters employed in the design of the SWS are reviewed. The description of the SWS, the piping and instrumentation diagrams (P&ID's), and the process flow diagrams showing the methods of operation and factors that influence waste treatment are reviewed. The expected chemical content flows and radionuclide concentrations of liquid wastes to be processed and handled by the SWS and the expected volumes to be returned to the liquid radwaste system for further treatment are reviewed.

   b. The description of the methods for solidification (i.e., of removal of free water), the solidifying agent used, and the methods to be employed to ensure a solid matrix are reviewed.

   c. The description of the type and size of solid waste containers; the method of filling, handling, and monitoring for removable radioactive contamination; and provisions for decontamination, packaging and storage to meet applicable federal regulations are reviewed.

2. The provisions for the onsite storage of solid wastes, the expected and design volumes, the expected and design radionuclide contents, and the design bases for these values are reviewed.

3. The quality group classifications of piping, and equipment and bases governing the classification chosen are reviewed.

4. Design provisions incorporated in the equipment and facility design to reduce leakage and facilitate operation and maintenance are reviewed.

5. Special design features, referenced topical reports, and previous experience with similar equipment and methods referenced in the SAR are reviewed.

6. The technical specifications proposed by the applicant for process and effluent control are reviewed at the operating license (OL) stage (FSAR).

   SEB will provide an evaluation of the applicant's proposed seismic design classification of structures housing the solid radwaste systems and the required seismic analysis for inclusion in the staff's Safety Evaluation Report.

II. ACCEPTANCE CRITERIA

1. ETSB will accept the solid waste treatment system design if the following conditions are met:

   a. The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs and with the source terms of Section 11.1.

   b. All wet solid wastes will be solidified prior to shipment offsite and there are provisions to verify the absence of free liquid in the containers and to reprocess containers in which free liquid is detected in accordance with Branch Technical Position (BTP) ETSB 11-3.

   c. Solid waste containers, shipping casks, and methods of packaging meet applicable federal regulations, e.g., 10 CFR Part 71, and wastes are to be shipped to a licensed burial site in accordance with applicable Commission and Department of Transportation regulations.

2. ETSB will accept the design capacity of the SWS if the following conditions are met:

   a. Processing equipment is sized to handle the design SWS inputs, e.g., the solid waste generation rates reviewed under I.1 of this plan without the need to ship bulk liquids.

   b. Onsite waste storage facilities provide sufficient storage capacity to allow time for short-lived radionuclides to decay prior to shipping.

   The bases for the storage time chosen should be given in the safety analysis report.

3. ETSB will accept SWS components and piping systems, and structures housing SWS components, designed in accordance with the provisions of Branch Technical Position (BTP) ETSB 11-1 (Rev. 1) (Ref. 10).

4. ETSB will accept systems that contain provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Branch Technical Position (BTP), ETSB 11-1 (Rev. 1) (Ref. 10) and Branch Technical Position (BTP), ETSP 11-3 (Ref. 11).

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this review plan, as may be appropriate for a particular case.

1. a. ETSB reviews the P&ID's and the process flow diagrams to determine system design, methods of operation, and parameters used in the design, i.e., expected and design flow rates, radioactivity concentrations, radionuclides, and waste categories.

   The system design and design criteria will be compared with the guidelines of Branch Technical Position (BTP) ETSB 11-3 and available data from operating plants of similar design.

   b. ETSB compares the methods to be used to solidify liquids with experience gained from previous licensing reviews and with available data from operating plants employing similar methods. ETSB will review the process control programs to assure that the proposed solidification method is capable of solidifying the range of constituents expected to be present in the wastes. ETSB reviews the methods proposed to verify that all liquids have been immobilized or combined during solidification operations and will determine its acceptability considering (1) the ability of the technique to detect free, mobile, or uncombined liquids, (2) the procedure to be employed to solidify free liquids if detected, and (3) the effect of the method on operator exposures.

   c. ETSB reviews the description of procedures for the packaging and shipment of solid wastes to an approved offsite burial facility, and verifies that the applicant makes definite commitments to following appropriate federal and state regulations. ETSB compares the values given in the SAR for the volumes and radionuclide content of solid wastes to be shipped offsite with data from operating plants of similar design and information from previous license applications.

2. ETSB compares the solid waste system design capacity with the design basis input waste volumes to determine whether the applicant has provided sufficient reserve capacity for greater-than-expected waste volumes which may occur as a result of anticipated operational occurrences. The inplant storage capacity is compared to the guidelines of (BTB) ETSB 11-3. The comparison will be based on the design criteria as stated in

the SAR, on the availability of system components to handle surge flows, and on whether the storage facilities will provide onsite storage periods sufficient to permit the decay of short-lived radionuclides.

3. ETSB compares the quality group and seismic design classifications of the solid waste system and of the structures housing the system to the guidelines of BTP ETSB 11-1 (Rev. 1). The consequences of failures of tanks containing radioactive liquids are evaluated under SRP 15.7.3.

4. ETSB compares equipment layout, design features, and mode of operation of the solid waste system to the guidelines of (BTP) ETSB 11-1 (Rev. 1) and (BTB) ETSB 11-3.

5. At the OL stage ETSB reviews the technical specifications proposed by the applicant for process and effluent control. The reviewer will determine that the content and intent of the technical specifications prepared by applicant are in agreement with the requirement developed as a result of the staff's review. The review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

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 solid waste system (SWS) includes the equipment and instrumentation used for the solidification, packaging, and storage of radioactive wastes prior to shipment offsite for burial. The scope of the review of the SWS includes line diagrams of the system, piping and instrumentation diagrams)P&ID's), and descriptive information for the SWS and for those auxiliary supporting systems that are essential to the operation of the SWS. The applicant's proposed design criteria and design bases for the SWS, and the applicant's analysis of those criteria and bases have been reviewed. The capability of the proposed system to process the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with General Design Criterion 60, provisions for the handling of wastes relative to the requirements of 10 CFR Parts 20 and 71 and of applicable DOT regulations, and the applicant's quality group and seismic design classification relative to BTP ETSB 11-1, have also been reviewed. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the regulations and the guides referenced above, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed solid radwaste system is acceptable."

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation," and Appendix B, "Concentrations in Air and Water Above Natural Background."

2. 10 CFR §50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Reactors."

3. 10 CFR §50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors."

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

5. 10 CFR Part 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection."

6. 10 CFR Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Materials Under Certain Conditions."

7. 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.

8. Regulatory Guide 1.BB, "Calculations of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWR's)."

9. Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWR's)."

10. Branch Technical Position ETSB 11-1 (Rev. 1), "Design Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Reactor Power Plants," attached to Standard Review Plan 11.2.

11. Branch Technical Position ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants" attached to Standard Review Plan 11.4.

Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants

A. Background

Solid wastes may be generated as a byproduct of nuclear power either directly, as with spent air filtration media (dry wastes), or indirectly, as with concentrated evaporator bottoms which undergo a solidification process prior to shipping (wet wastes). Solidification processes may be additionally used to render wastes already in a solid form, e.g., spent demineralizer resins, into a less mobile form, thereby mitigating the consequences of potential ruptures to shipping containers.

Dry wastes normally undergo a compaction process to reduce the volume of waste shipped offsite. Special provisions are needed to assure that contaminated airborne dusts are not released to the process area during compaction.

Although there are a number of processes available which are capable of solidifying liquid wastes under controlled conditions, there is a potential for free[1] liquids to remain in containers following solidification with the widely varying chemical species encountered during power plant operations. Based on the NRC staff's judgment, it is necessary that vendors and operators implement certain measures to:

1) establish process parameters within which systems must be operated to obtain complete solidification and

2) assure systems are operated within the established process parameters, or

3) have provisions to detect free liquid in containers prior to shipment offsite.

Following packaging, wastes are normally stored for decay of short-lived radionuclides and to accumulate sufficient wastes for a shipment offsite. Insofar as the continuous operation of the solid waste system is contingent upon storage space being available for the interim period between waste packaging and shipment offsite, consideration should be given to providing ample storage capacity to accommodate wastes during periods when shipments offsite are not possible, e.g., during labor strikes.

Until a more definitive guide is published, the criteria in Section B, below, provides adequate and acceptable design solutions for the concerns outlined above.

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[1]For the purpose of this position paper, free water is defined as uncombined water not bound by the solid matrix.

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This position paper sets forth minimum branch requirements and is not intended to prohibit the implementation of more rigorous design codes, standards, or quality assurance measures than those indicated herein.

B. Branch Technical Position

I. Dry Solid Waste Compaction

a. Solid waste compaction devices should include a ventilated shroud around the waste container to control the release of airborne dusts generated during the compaction process.

II. Waste Forms Acceptable for Shipment Offsite

a. All wastes should be in a solid immobile form prior to shipment offsite.

b. Spent resins and filter sludges should be combined with a suitable binding agent (e.g., cement, urea formaldehyde) and formed into a solid matrix.

c. For normal operation, shipment of liquids offsite is unacceptable. Means should be provided to effect the complete solidification of all wastes which can be reasonably expected to be generated during normal operation including anticipated operational occurrences.

d. Adsorbants, such as vermiculite, are not acceptable substitutes for solidification.

III. Assurance of Complete Solidification

Complete solidification of wastes should be assured by the implementation of process control program or by methods to detect free liquids within container contents prior to shipment.

a. Process Control Program

1. Solidification agents and potential waste constituents should be tested and a set of process parameters established which provide boundary conditions within which reasonable assurance can be given that solidification will be complete.

2. The plant operator should provide assurance that the process is run within the parameters established under 1 above. Appropriate records should be maintained for individual batches showing conformance with the established parameters.

b. Free Liquid Detection

Each container filled with solidified wet wastes should be checked by suitable methods to verify the absence of free liquids. Visual inspection of the upper surface of the waste in the container is not alone sufficient to ensure that free water is not present in the container. Provisions to be used to verify the absence of free liquids should consider actual solidification procedures which may create a thin layer of solidification agent on top without affecting the lower portion of the container.

IV. Waste Storage

a. Tanks accumulating spent resins from reactor purification systems should be capable of accommodating at least 60 days' waste generation at normal generation rates. Tanks accumulating spent resins from other sources and tanks accumulating filter sludges should be capable of accommodating at least 30 days' waste generation at normal generation rates.

b. Storage areas for solidified wastes should be capable of accommodating at least 30 days' waste generation at normal generation rates.

c. Storage areas for dry wastes should be capable of accommodating at least one full offsite waste shipment.

V. Additional Design Features

The following additional design features should be incorporated into the design of the solid waste system.

a. Evaporator concentrate piping and tanks have heat tracing.

b. Components and piping which contain radioactive slurries have flushing connections.

c. Solidification agents are stored in low radiation areas generally less than 2.5 mr/hr with provisions for sampling.

d. Tanks or equipment which use compressed gases for transport or drying of resins or filter sludges should be directly exhausted to the plant ventilation exhaust system through HEPA filters as a minimum.

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SECTION 11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING 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

ETSB reviews the following information in the applicant's safety analysis report (SAR):

1. The design objectives and design criteria for the process and effluent radiological monitoring and sampling system are reviewed. The review includes the identification of the process and effluent streams to be monitored or sampled, the purpose of each monitoring or sampling function provided, and the parameters to be determined through monitoring or sampling (e.g., gross beta-gamma concentrations, radionuclide distribution, or quantities of specific radioisotopes).

2. The system description for the process and effluent radiological monitoring and sampling system is reviewed. The review includes (a)* description of instrumentation provided, including redundancy, range, independence, and diversity of components for normal operations, anticipated operational occurrences, and postulated accidents; (b)* location of monitors and direct readouts; (c)* location of sampling points and sampling stations; (d) calculation of radioactivity concentrations to be monitored or sampled for normal operations, anticipated operational occurrences, and postulated accidents; (e) measurements or determinations to be made (e.g., gross beta-gamma concentration or measurement of specific radionuclides); (f)* types and locations of annunciations and alarms and the actions initiated by each, (g) provisions for purging sample lines, input volumes to waste collection systems, and sampling frequency, (h) expected relationships between monitoring and sampling results and plant operations; (i)* descriptions or procedures for calibration, maintenance, and inspection of monitoring instrumentation; (j) layout drawings, piping and instruentation diagrams (P&ID's), and process flow diagrams.

3. The technical specifications proposed by the applicant for process and effuent control are reviewed at the operating license (OL) stage (FSAR).

* Final Safety Analysis Report (FSAR) only.

EICSB will review the seismic design, redundancy, and emergency power supply for process and effluent radiological monitors required for safe shutdown of the plant under Standard Review Plan (SRP) 7.6.

AAB will review specification and criteria for monitoring instrumentation and postulated accidents.

II. ACCEPTANCE CRITERIA

The applicant's design should meet the following criteria:

1. Provisions should be made to monitor or sample all normal and potential pathways for release of radioactive materials to the environment in conformance with General Design Criterion 64 and Regulatory Guide 1.21.

   a. For a boiling water reactor (BWR) the following process streams or effluent release points should be monitored or sampled continuously:

      (1) Main condenser air ejector offgases.
      (2) Main condenser offgas treatment system effluents.
      (3) Turbine gland seal condenser effluents.
      (4) Mechanical vacuum pump effluents.
      (5) Ventilation air exhausts from all buildings having the potential to contain airborne radioactivity.
      (6) Liquid waste effluent streams.
      (7) Service water effluent stream.

   b. For a pressurized water reactor (PWR) the following process streams or effluent release points should be monitored or sampled continuously:

      (1) Main condenser air ejector offgases.
      (2) Waste gas treatment system effluent.
      (3) Equipment vents routed directly to the environment (e.g., steam generator blowdown flash tank vent, liquid waste tank vents).
      (4) Ventilation air exhausts from all buildings having the potential to contain airborne radioactivity; .
      (5) Turbine building floor drain effluents.
      (6) Liquid effluents from the steam generator blowdown system.
      (7) Boron recovery system effluents.
      (8) Liquid waste effluent streams.
      (9) Service water effluent stream.
      (10) Component cooling water loop.

   c. For both BWR's and PWR's, liquid wastes should be sampled batchwise prior to release, in accordance with Regulatory Guide 1.21. For liquid discharges which cannot be practicably monitored (e.g., main condenser cooling water), an alternative method such as periodic automatic composite sampling and weekly analyses should be provided. Continuous vent monitors for open structures, such as a PWR turbine building, are not required.

2. Provisions should be made to monitor or sample radioactive waste process systems in conformance with General Design Criterion 63.
   a. Provisions should be made to assure representative samples from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating two tank volumes in approximately eight hours. For process stream samples, provisions should be made for purging sample lines and for reducing plateout in sample lines. Provisions for sampling from ducts and stacks should be in agreement with ANSI N13.1.
   b. Provisions should be made to collect samples from process waste streams at central sample stations to reduce leakage, spillage, and radiation exposures to operating personnel.
   c. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system.

3. Instrumentation, sampling and monitoring provisions should conform to the following:
   a. Sampling frequencies, required analyses, instrument sensitivities, and provisions to composite samples for low-level analyses should be in conformance with Regulatory Guide 1.21. Sampling frequencies and required analyses should be given in the plant technical specifications; these provisions will be reviewed at the OL stage.
   b. Provisions for the necessary instrumentation and facilities to perform gross beta-gamma and gross alpha measurements, isotopic analyses, and other routine analyses should be in conformance with Regulatory Guide 1.21.
   c. Provisions should be made to perform routine instrument calibration, maintenance, and inspections. The frequencies of such actions should be given in the plant technical specifications. The provisions will be reviewed at the OL stage. Provisions should also be made to replace or decontaminate monitors without opening the process system or losing the capability to isolate the effluent stream.

4. Continuous monitors on liquid effluent lines and gaseous release points should alarm when radionuclide concentrations exceed a predetermined level in the discharge line. In addition, provisions to automatically terminate the discharge of effluents at a predetermined level should be made for the following streams:
   a. BWR's: Main condenser air ejector offgas
      Drywell purge
      Containment purge
      Liquid waste discharge
   b. PWR's: Waste gas treatment system discharge
      Containment purge
      Steam generator blowdown system discharge*
      Turbine building floor drain discharge*
      Boron recovery system discharge*
      Liquid waste discharge

   Isolation valves should fail in the closed position. The release rates should be established in the plant technical specifications.

*For release routes other than through the liquid waste system.

5. The process and effluent radiological monitoring and sampling system provisions for postulated accidents will be acceptable if normal gaseous effluent paths are provided with supplemental monitoring equipment capable of monitoring postulated accident releases in accordance with ANSI Draft Standard N13/42 WG6.

6. The PERMSS provisions for monitoring liquid effluents for postulated accidents will be acceptable if both normal and postulated accident liquid effluents are discharged in the batch mode, if administrative procedures are in effect to minimize inadvertent or accidental releases of radioactive fluids, and if the normal liquid effluent monitors provide automatic termination of releases in the event that established effluent control levels are exceeded.

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 review of the process and effluent radiological monitoring and sampling system, ETSB will compare the listing of process and effluent monitors contained in the SAR with the principal release points identified in SRP 11.2 and SRP 11.3 to assure that all major process streams and release pathways are being monitored during normal operation and anticipated operational occurrences. The review includes the following:
   a. The location of probes, sample stations, and the bases for the selection of these sample points are compared with the general principles for obtaining valid samples of airborne radioactive materials, the methods and materials for gas and particulate sampling, and guides for sampling from ducts and stacks contained in ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."
   b. The equipment, piping, and description of sampling methods to assure representative sampling will be compared with the guidelines given in Regulatory Guide 1.21.
   c. ETSB will independently calculate on an audit basis the radiation levels and concentrations in the process and effluent stream using the models of Draft Regulatory Guides 1.BB and 1.CC to verify the expected levels.
   d. ETSB will compare the sampling frequencies, types of analyses required, and monitoring instrument sensitivities with those recommended in Regulatory Guide 1.21. At the OL stage, ETSB will compare the applicant's monitoring instrumentation specifications and performance criteria with that contained in ANSI N13.10-1974 (Ref. 7).
   e. In the review of the P&ID's for the waste treatment system, ETSB will verify that all major release points of radioactive material have provisions for automatic termination of releases in the event they exceed a predetermined level.
   f. ETSB will review the location of the monitors shown on the P&ID's and the location of readouts, annunciators, and alarms discussed in SAR Chapter 7 to assure that the operator will be advised of system performance and effluent releases consistent with the release limits specified in the plant technical specifications.
   g. ETSB will compare the calibration methods with the guidelines in Regulatory Guide 1.21 (FSAR)
   h. ETSB will assure that provisions are included in the design for replacing detectors or decontaminating the monitors without opening the process system or losing the capability to isolate the system or divert the effluent to a standby treatment system

i. In the review of the radiation levels and concentrations of radioactive material expected from postulated accidents, ETSB will compare the applicant's values with values contained in ANSI N13/42 WG6 (Ref. 3).

j. In the review of the design and operation of the monitoring systems for postulated accidents, ETSB will compare the applicant's design with the guidance contained in ANSI N13/42 WG6.

k. ETSB will review special features, applicable topical reports, and data referenced in the SAR on a case-by-case basis.

2. ETSB reviews the technical specifications proposed by the applicant for process and effluent radiological monitoring and sampling at the OL stage. The reviewer determines 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 review will include the evaluation or development of appropriate limiting conditions for operation and their bases consistent with the plant design.

IV. EVALUATION FINDINGS

ETSB 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 process and effluent radiological monitoring and sampling systems include the instrumentation for monitoring and sampling contaminated liquid, gaseous, and solid waste process and effluent streams. Our review included the provisions proposed to sample and monitor all station effluents in accordance with General Design Criterion 64, the provisions proposed to provide automatic termination of effluent releases and assure control over discharges in accordance with General Design Criterion 60, the provisions proposed for sampling and monitoring plant waste process streams for process control in accordance with General Design Criterion 63, the provisions for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and the provisions for monitoring process and effluent streams during postulated accidents. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems, and ventilation systems, and the location of monitoring points relative to effluent release points on the site plot diagram.

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring and sampling systems to the applicable regulations and guides, as indicated above, as well as to staff technical positions and industry standards. Based on our evaluation, we find the proposed systems to be acceptable."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Material to the Environment;" Criterion 63, "Monitoring Fuel and Waste Storage;" and Criterion 64, "Monitoring Radioactivity Releases."

2. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969).

3. ANSI N13/42 WG6, Draft, "Performance Specifications for Reactor Emergency Monitoring Instrumentation," American National Standards Institute (1974).

4. Regulatory Guide 1.21, "Measuring 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.

5. Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWR's)."

6. Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors (BWR's)."

7. ANSI N13.10-1974, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents" (1974).

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SECTION 12.1 ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) related to assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA) are reviewed:

1. POLICY CONSIDERATIONS
   a. Management policy with respect to designing and constructing the plant (preliminary safety analysis report, PSAR) and with respect to operating the plant (final safety analysis report, FSAR) and the planned organizational structure (PSAR).
   b. The applicable activities carried on by the individuals in management having responsibility for radiation protection (PSAR).
   c. Information describing the implementation of policy, organizational, training, and design review guidance or requirements provided in Regulatory Guide 8.8, Regulatory Guide 8.10, Regulatory Guide 1.8, and in 10 CFR Part 20. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

2. DESIGN CONSIDERATIONS
   a. Facility and equipment design considerations (PSAR) (but not design features, as in SAR Section 12.3).
   b. The description of the use of dose assessment (SAR Section 12.4) and the resultant man-rem doses associated with various work functions for evaluating the facility design relative to assuring that ORE will be ALARA (PSAR).
   c. The description of how experience from past designs, from various design guidelines, such as Regulatory Guide 8.8, and from operating plants is used to develop improved radiation protection designs (PSAR).
   d. Information describing the implementation of the design guidelines of Regulatory Guide 8.8 and other industry developed design guidance that includes the ALARA Criteria. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

3. OPERATIONAL CONSIDERATIONS

   a. The techniques and methods that will be used for developing detailed operational plans and procedures (PSAR).

   b. The description of how procedures for operation may impact on and interact with the plant design and on the design process (PSAR).

   c. The use of operating plant experience in planning the operational considerations for new plant designs (PSAR).

   d. The criteria and conditions under which various exposure reduction procedures and techniques are implemented (FSAR).

   e. Information describing the implementation of radiation protection programs, and operational guidance of Regulatory Guide 8.8 and Regulatory Guide 8.10. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.1 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Specific acceptance criteria for these areas of review are as follows:

1. POLICY CONSIDERATIONS

   Acceptability will be based on evidence in the PSAR that a policy for assuring that ORE will be ALARA has been formulated, described, displayed, and will be implemented in accordance with Regulatory Guide 8.8, Regulatory Guide 8.10, C.1, and Regulatory Guide 1.8. This includes designating and assigning a specific individual responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the Regulatory Guides and industry standards.

2. DESIGN CONSIDERATIONS

   Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with Regulatory Guide 8.8, including incorporation of: measures for reducing the need for maintenance is required; measures to improve the accessibility to components requiring periodic maintenance or inservice inspection; measures to reduce the production, distribution and retention of activated corrosion products throughout the primary system; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; feedback of experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a comparison with the Regulatory Guide.

3. OPERATIONAL CONSIDERATIONS

   Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with Regulatory Guide 8.8, which can incorporate the experiences obtained in facility operation into facility and equipment design and into operations planning and which will implement specific exposure control techniques. In the FSAR, acceptance of criteria and conditions under which various exposure reduction procedures and techniques are implemented will also be based on Regulatory Guides 8.8

and 8.10, C.2. Alternative proposed plans and procedures will be evaluated on the basis of a comparison with the Regulatory Guides.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2. RAB reviews the management policy and the planned organizational structure to determine how the guidance given in Regulatory Guides 8.8, 8.10 and 1.8 will be implemented or whether alternatives have been proposed. The organizational structure review includes determining if the individuals responsible for the radiation protection program are on a level in management that will assure implementation of management's commitment for assuring that ORE will be ALARA. Any problems concerning the review of organizational structure, as related to the radiation protection manager will be communicated to Quality Assurance Branch, which has primary review responsibility for this item, in Chapter 13. The reviewer uses ANSI N18.7-1972 for additional guidance on acceptable operating organizations.

The reviewer evaluates information in this section to determine if the organizational structure provides a mechanism for the radiation protection manager and his organization interacting with design groups to assure that methods and techniques for reducing ORE will be incorporated in the design of the plant. Where the future plant radiation protection manager has not been selected, design review should be accomplished in accordance with the guidance of Regulatory Guide 8.8, or a proposed alternative. For those applicants that have operating reactors the reviewer determines that a review of the proposed plant design is conducted by appropriate personnel from the operating plant. The reviewer determines from information furnished by the applicant that he has incorporated previsouly accepted design features and has used operating experience to improve on the design of the plant with regard to assuring that ORE will be ALARA. The reviewer compares the material in this section with the requirements of 10 CFR Part 20 and the guidelines of Regulatory Guides 8.8 and 8.10.

Based on the review RAB may request additional information or request the applicant to modify his submission in order to meet the acceptance criteria given in Section II.

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. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

"12.1 Assuring that ORE are ALARA.

"This section of the applicant's SAR has been reviewed to determine the policy, design, and operational considerations related to assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). The review covered the description of the applicant's management policy and organizational structure regarding radiation protection, the design considerations that are used to apply experience from past designs and operating experience toward achieving improved designs, and the methods and techniques used for developing plans and procedures for assuring that ORE will be ALARA. The review included analysis of the manner by which the applicant's policy, design, and operational considerations conform to the guidelines of Regulatory Guides 8.8, 8.10, and 1.8 or provide acceptable alternatives.

"The basis for acceptance has been conformance of the management policy, design, and operational considerations with established guidelines, criteria, and good practice at licensed nuclear plants, including those provided in Regulatory Guides 8.8, 8.10, and 1.8. The management policy and individual responsibilities are appropriately implemented both with regard to the design process and with regard to developing and considering operational procedures.

"It is concluded that the policy considerations, design considerations, and operational considerations related to assuring that ORE will be ALARA conform to the Commission's Regulations and to the appropriate Regulatory Guides and industry standards and are considered acceptable."

V. REFERENCES

1. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Practicable (Nuclear Reactors)."

2. 10 CFR Part 20, "Standards For Protection Against Radiation."

3. Regulatory Guide 8.10, "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Practicable."

4. Regulatory Guide 1.8, "Personnel Selection and Training."

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

6. ANSI N18.7-1972, "Administrative Control for Nuclear Power Plants," American National Standards Institute (1972).

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SECTION 12.2 RADIATION SOURCES

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to radiation sources are reviewed as related to in-plant radiation protection:

1. CONTAINED SOURCES

The description of the sources of radiation that are the basis for the radiation protection program, as used in the shield design calculations (SAR Chapter 11 contains the description for sources contained in equipment of the radioactive waste management systems). This should include isotopic composition, location in the plant, source strength and source geometry, and the basis for the values (preliminary safety analysis report (PSAR), and update in the final safety analysis report (FSAR). The descriptions should include any required byproduct, source, and special nuclear materials.

2. AIRBORNE RADIOACTIVE MATERIAL SOURCES

The description of the sources of airborne radioactive material considered in the design of the ventilation systems, as used for design of personnel protective measures and for dose assessment. (SAR Chapter 11 contains the description for airborne sources that have to considered for their contribution to the plant effluent releases, through equipment of the radioactive waste management systems or the plant ventilation system). The descriptions should include a tabulation of the calculated concentration of radioactive material by nuclides expected (based on some failed fuel assumptions as in 1 above) during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel and should include models and parameters for the calculations (PSAR and update in FSAR).

The ETSB will review the description of the sources and the assumptions made by the applicant in calculating the various source terms provided. ETSB will confirm in SRP Section 11.1 that the proper assumptions have been made or that the source values are correct.

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.2 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Specific acceptance criteria for these areas of review are as follows:

All sources of radiation that necessitate designed shielding, special ventilation designs, special storage locations and conditions, traffic or access control considerations, special plans and procedures, monitoring equipment, etc., shall be described to the degree needed for the shielding codes used in the design process, for establishing related facility design features, for plans and procedures development, for assessment of occupational radiation exposures, and for equipment specifications. For contained sources, the description should include plan scale drawings of each floor of the plant on which all the sources of significance are shown and identified in a manner that can be easily related to tables containing the pertinent and necessary quantitative source parameters. Their position should be accurately indicated, as well as the approximate size and shape. Airborne sources that are created by leakage, by opening formerly closed containers, by storage of leaking fuel elements, etc., shall be identified by location and magnitude, in a manner useful for designing appropriate ventilation systems and in specifying appropriate monitoring systems. The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to Chapter 11 of the SAR. Acceptance criteria for the assumptions and methods used in arriving at source term values are discussed in Chapter 11 and are judged by the ETSB. The tables of quantitative source parameters, which can be placed in Chapter 12 or referenced to Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation or in a ventilation system design. In addition, the quantities will be acceptable if the reviewer can confirm specific values given in the tables from the assumptions given along with common tables of nuclide decay schemes, gamma ray energies, etc. Specific acceptance criteria for the N-16 shielding source terms are being developed. For byproduct, source, and special nuclear materials, 10 CFR Parts 30, 40 and 70 can be used as guidance.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The reviewer determines whether source strengths, concentrations of airborne radioactivity, and quantitative source descriptions are consistent with the assumptions made and the methods used by the applicant, primarily through the review performed by ETSB. Locations of the contained sources relative to shield walls, occupied areas, traffic pathways, in-service inspection points, sampling stations, controls, etc., are examined for any potential problems in assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). Based on the review, RAB may request additional information or request the applicant to reevaluate his analysis for the purpose of modifying those areas which do not meet the acceptance criteria given in Section II.

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. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

"12.2 Radiation Sources.

"All radioactive sources that form the basis for the radiation protection design of the ____________ plant have been reviewed. The review has included contained sources and airborne radioactive material sources and the basis for the quantitative source informations tabulated in this and referenced sections of the SAR. It has included the description of byproduct, source, and special nuclear materials.

"The basis for acceptance in the review has been that all information required has been provided and that the source parameters are consistent with the staff's values. The descriptions and values are appropriate for use in shielding design calculations and for developing a plant radiation protection program. The source locations and physical descriptions have been provided in a manner which permits evaluation of the radiation protection design and procedures.

"It is concluded that all appropriate sources have been considered for the development of the radiation protection design and program."

V. REFERENCES

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

2. 10 CFR Part 70, "Special Nuclear Material."

3. 10 CFR Part 40, "Source Material."

4. 10 CFR Part 20, "Byproduct Material."

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SECTION 12.3 RADIATION PROTECTION DESIGN FEATURES

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to radiation protection design features are reviewed:

1. FACILITY DESIGN FEATURES

a. In the preliminary safety analysis report (PSAR), the description of equipment and facility design features used for assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA).

b. The radiation zone designations, including zone boundaries for both normal operational and refueling conditions (PSAR and update in the final safety analysis report, FSAR).

c. The illustrative examples of facility design features of the equipment, components, and systems lised in Sections 12.1.3 and 12.3.1 of "Standard Format and Content...." (Ref. 1) including the scaled layout and arrangement drawings of the facility showing all source locations and the other design details requested in Section 12.3.1 of the "Standard Format...." (PSAR and update in FSAR). Shield wall thicknesses for all shielded spaces should be specified on the drawings or provided in separate tables.

d. The description of facilities and equipment for handling and use of sealed and unsealed special nuclear, source, and byproduct materials (PSAR and update in FSAR).

e. Information describing implementation of Regulatory Guide 8.8 guidelines on facility and equipment design and layout. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

2. SHIELDING

a. The shielding to be provided for each of the radiation sources identified in SAR Chapter 11 and Section 12.2, including the design criteria for penetrations and the shield material used (PSAR and update in FSAR). (Note item I.1.c above)

b. The description of the methods by which the shield parameters were determined, including pertinent codes, assumptions, and techniques used or to be used in the calculations (PSAR and update in FSAR).

c. The description of any special protective features that use shielding, geometric arrangement, or remote handling to assure that ORE will be ALARA (PSAR and update in FSAR).

d. Information describing implementation of Regulatory Guides 1.69 and 8.8 (regarding special protective features). Information describing alternatives, if such are proposed (PSAR and update in FSAR).

3. VENTILATION

a. The description of the personnel protection features incorporated in the ventilation system design as called for in Section 12.3.3 of "Standard Format and Content..." (PSAR and update in FSAR).

b. Illustrative example of the air cleaning system design (PSAR and update in FSAR).

c. Information describing any application of Regulatory Guide 1.52 (particularly Section C.4 & 5) and Regulatory Guide 8.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

4. AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

a. The description of the fixed area radiation and continuous airborne radioactivity monitoring instrumentation, including in the PSAR the criteria for placement and in the FSAR additional details as called for in Section 12.3.4, "Standard Format and Content....," for normal operation, anticipated operational occurrences, and accident conditions.

b. The criteria and method for obtaining representative in-plant airborne radioactivity concentrations (PSAR and update in FSAR).

c. Information describing the implementation of Regulatory Guides 1.21, 8.2, 8.8 and ANSI N13.1-1969. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.3 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. Specific acceptance criteria for these areas of review are as follows:

1. FACILITY DESIGN FEATURES

Acceptability of the facility design features will be based on evidence that the applicant has applied the guidance in Regulatory Guide 8.8 or that alternatives have been proposed. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling, processing, etc., in-service inspection and calibration) have been considered in plant design and that potential radiation exposure from these activities will be kept ALARA by radiation protection features incorporated in the design, including ease of accessibility to work and inspection and sampling areas, the ability to reduce source intensity, design measures to reduce the production, distribution, and retention of activated corrosion products, the ability to reduce time required in radiation fields, provision for portable shielding and remote handling tools, etc. Access control will be judged for acceptability in accordance with the requirements of 10 CFR § 20.203.

The areas inside the plant structures, as well as the general plant yard, must be identified by showing radiation areas with acceptable maximum design dose rates and zones. Maximum zone dose rate should be defined for each zone, as well as anticipated occupancy and access control. Acceptance criteria are as follows: The areas that have to be occupied on a predictable basis (number of people and stay or transit times) during normal operations and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling; processing, use, storage and disposal; normal maintenance; routine operational surveillance; inservice inspection; and calibration) should be zoned such that this occupancy results in an annual dose to each of the involved individuals that is below the limits of 10 CFR Part 20 and is as low as is reasonably achievable. Based on current operating experience and on predictions made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned and sufficient radiation protection design features can be incorporated such that these individuals would receive a small fraction of the 10 CFR Part 20 limit. Whether radiation protection design and zoning is acceptable will be based partly on the actual numbers for average annual radiation exposure to these individuals, determined in the dose assessment required in Section 12.4.

2. SHIELDING

The shielding design is evaluated as to the assumptions used to calculate shield thickness, the calculational method used, and the parameters chosen. There are a number of acceptable shielding calculational codes available for use that are effective for determining the necessary shield thickness for gamma ray sources and for combination neutron-gamma sources. Most of the codes used by shield designers have been entered into the code description file of the Radiation Shielding Infomation Center at Oak Ridge National Laboratory, which means they have been tested and authenticated for operation but not for reliability and accuracy. RAB has three codes in-house for use in shielding calculations. These are SDC, a kernal integration shield design code; G^3, a general purpose gamma ray scattering program; and MORSE, a general purpose Monte Carlo multigroup neutron and gamma ray transport code. SDC can calculate gamma ray shielding requirements, handling 13 source geometries (including point, line, disk, plane, slab, and sphere) and with cross sections and materials compositions for 17 materials. As many as 12 gamma ray energy groups, covering the range from 0.1 to 10 MeV, may be used to describe the gamma ray spectrum. The staff will use these codes, as necessary, to calculate dose rates for given shield designs and source strengths, as a confirmation of the applicant's method.

The applicant's shielding design is acceptable if the methods are comparable to commonly acceptable shielding calculations and assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic. Acceptable shielding codes include but are not limited to ANISN, DOT, MORSE, SAM-CE, O5R, O6R, G^3, SDC, and many others. This listing does not imply that all these codes are equivalent, since some are much more sophisticated than others. The staff believes it is advantageous to use a good calculational procedure, since an effective shield design is essential to meeting the criteria that occupational radiation exposures will be as low as is reasonably achievable.

Two documents provide additional guidance for acceptability of the shielding design. One is "Reactor Shielding for Nuclear Engineers," Edited by N. M. Schaeffer, published by AEC-OIS, 1973." The second is the Stone & Webster Engineering Corporation topical report RP-8 entitled "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants." These documents provide useful guidance regarding radiation shielding design. Some limitations are noted for RP-8, in that the labyrinth entrance ways may not provide dose rates at the outside entrance consistent with area radiation zoning.

In addition, Regulatory Guide 1.69 provides guidance on the fabrication and installation of concrete radiation shields for nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of this document has been implemented in the facility construction, or that acceptable alternatives have been proposed. Regulatory Guide 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.

3. VENTILATION

The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will assure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, and that the concentrations of radioactive material in areas normally occupied can be maintained in accordance with the requirements of 10 CFR Part 20. The system shall have adequate capability to reduce concentrations of airborne radioactivity in areas not normally occupied where maintenance or in-service inspection has to be performed, to levels in accordance with the requirements of 10 CFR § 20.103. The system shall be designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel in normally occupied areas. Acceptability of the ventilation system, relative to radioactive gases and particulates will also be based on evidence that the applicant has applied the guidance of Regulatory Guide 8.8 or that alternatives have been prepared.

Regulatory Guide 1.52, particularly Sections C.4 and 5, provides guidance that can be used in this review, although the guide is written with regard to mitigating accidents involving airborne radioactivity. Good practice in that regard is applicable to normal operation as well, since release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases.

4. AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The area radiation monitoring instruments will be acceptable if they meet the following criteria:

a. The detectors are located in areas and normal access corridors used and occupied without restricted access which may have a potential for radiation fields in excess of the radiation zone designation given in Section 12.3.1.

b. The detectors are sensitive to dose rates that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for anticipated operational occurrences.

c. Essential instruments are provided with "auxiliary" or emergency power in the event of a power failure or postulated accidents. Specific criteria are being developed.

d. The detectors are calibrated routinely and after any maintenance work is performed on the detector. Specific criteria are being developed.

e. Each location has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.

f. There is readout and annunciation in the control room.

The continuous airborne radioactivity monitoring system will be acceptable if it meets the following criteria in addition to the above:

a. Air is sampled at normally occupied locations where airborne radioactivity is most likely to exist, such as solid waste handling areas, spent fuel pools, reactor operating floors, and BWR turbine buildings, and is detected based on sensitivity of the detection system. Monitoring air being exhausted from locations within the facility is also acceptable during normal operation, provided the monitoring system is capable of detecting one Mpc-hour (particulate or gaseous radioactivity) in any compartment which had a possibility of containing airborne radioactivity and which may be occupied by personnel.

b. Representative air concentrations are measured at the detectors, which are located as close to the sampler intakes as possible.

c. Ventilation monitors are upstream of HEPA filters.

Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants," provides useful guidance that is applicable to the acceptability of airborne radioactivity monitoring in-plant. Regulatory Guide 8.2, includes guidance on surveys to evaluate radiation hazard. American National Standard ANSI N13.1-1969 provides detailed guidance on sampling airborne radioactive materials in nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. Regulatory Guide 8.8 provides guidance on monitoring systems.

III. REVIEW PROCEDURES

The information radiation protection design features furnished in the SAR, including referenced parts of Chapters 9 and 11, is reviewed for completeness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The reviewer evaluates the SAR text and the scaled layout drawings of the facility, concentrating on the sources, shielding, and layouts for the auxiliary building, including the radwaste systems, decontamination facilities, office and access control areas, laundry, lockers and shower rooms, and laboratory facilities; the fuel handling facilities, including the spent fuel pool and related equipment; and the BWR turbine building, including location of steam lines, reheaters, and moisture separators. For the PSAR this review is particularly concerned with preliminary design features which appear to be contradictory to assuring that ORE will be ALARA. In this review, radiation protection design features are evaluated using the guidelines of Regulatory Guide 8.8. The access control plans are reviewed both to determine conformance with 10 CFR Part 20 and to determine whether they will control access properly in limited access areas and in restricted access areas (high radiation areas). The reviewer examines locations of critical controls, valve operating stations, pumps, sample collection stations, inservice inspection locations, radiation monitors, control panels, major pipes carrying radioactivity, filters for radioactive liquids and gases, and unshielded low level radioactive material storage or processing tanks. He also reviews SAR Chapters 9 and 11 to cover specific details of the fuel handling and storage systems, ventilation systems, and radwaste systems as they relate to radiation protection design. Chapter 9 will provide the major description of the mechanical features of ventilation systems with regard to the venting airborne radioactivity from the plant. Chapter 9 will also cover major features of the spent fuel pool design, the fuel handling system design and the spent fuel pool cleanup system. Chapter 11 may cover some of the design details of gaseous, liquid and solid radwaste systems that relate to radiation protection. The reviewer evaluates all aspects of the initial design plans under his areas of review, particularly to identify new arrangements, improved designs, unusual shield thicknesses, a new or modified shield thickness calculational procedure, unusual assumptions in the calculation, placement of radiation monitors, etc.

RAB evaluates the adequacy of the applicant's shielding design on the basis of acceptable radiation shielding codes. RAB makes a verifying check calculation with SDC, G^3, or MORSE, whichever is specifically applicable to the situation.

For the FSAR the reviewer considers any changes in the design that might necessitate changes in operating procedures to accommodate a changed radiation zone or a different location of equipment.

The reviewer determines whether the applicant has followed the guidance of the referenced Regulatory Guides and industry standards, both by comparison of the applicant's methods with the information in the guides and by the applicant's reference to any such guides or to alternatives that have been proposed. The reviewer evaluates whether the alternatives are equivalent to or improvements on the methods cited in the referenced Regulatory Guides. Alternatives that are neither of these are likely to be disapproved.

Based on the review, RAB may request additional information or request the applicant to reevaluate the radiation protection design features to meet the acceptance criteria of Section II.

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. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

"12.3 Radiation Protection Design Features

"This section of the applicant's SAR has been reviewed to determine that the radiation protection design features of the plant have been designed and provided in a manner that will assure that ORE will be ALARA. The scope of review covers the facility design features, the shielding, the ventilation systems, and the radiation monitoring instrumentation, as they relate to the plant radiation protection design.

"Basis for acceptance in the review has been conformance with established guidelines and criteria. The evaluation of the radiation protection design features provides reasonable assurance that it will be possible to operate the facility with ORE that are ALARA.

"The staff concludes that the protective features provided in the design of __________ nuclear plant conform to the Commission's Regulations, and to applicable Regulatory Guides, and industry standards, and are acceptable."

V. REFERENCES

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

2. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable."

3. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969).

4. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

5. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

6. "Reactor Shielding for Nuclear Engineers," N. M. Schaeffer, Editor; published by USAEC-OIS, 1973.

7. Regulatory Guide 1.52, "Design, Test, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light Water Cooled Nuclear Power Plants."

8. Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants."

9. 10 CFR Part 20, "Standards for Protection Against Radiation."

10. Stone and Webster Topical Report, "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants," RP-8, 1974.

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SECTION 12.4 DOSE ASSESSMENT

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) relating to in-plant and onsite radiation dose assessment are reviewed:

1. The expected occupancy of plant radiation areas, including numbers of personnel and times of occupancy. The estimated annual occupancy for each radiation zone and the dose rates at the points of occupancy and basis for the occupancy and the dose rate values (preliminary safety analysis report, PSAR and update in final safety analysis report, FSAR).

2. The objectives and criteria for design dose rates in-plant and at onsite areas (PSAR and update in FSAR).

3. The estimated annual man-rem doses associated with major functions such as operation, radwaste handling, normal maintenance, refueling, and in-service inspection and the average individual radiation exposure resulting from these activities (PSAR and update in FSAR).

4. The estimated annual dose at the boundary of the restricted area and to construction workers at a mult-unit plant due to radiation from onsite sources (PSAR and update in FSAR).

5. The description of any measures taken to reduce particular estimated man-rem doses for specific functions in cases where the dose would appear to result in excessive personnel costs to correct overexposures (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.4 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2. The dose limits in 10 CFR

§ 20.101 for exposure of individuals to radiation in restricted areas, and in 20.103 for exposure of individuals to concentrations of radioactive materials in restricted areas are upper limits and the plant must be designed so that they are not exceeded. The dose assessment is a key factor in determining if the plant design and proposed methods of operation assure that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). Acceptability will be based on the thoroughness with which the applicant displays and demonstrates that occupancy factors, dose rates in various occupied areas, and number of personnel needing to be involved are evaluated for the various areas of the plant, as specifically designed, and for the various functions that will be carried out. Occupancy factors will be acceptable if it is demonstrated that they are based on operating experience, coupled with the applicant's plan for operating the proposed plant, or that they are based on a thorough analysis of the future plant operation. Estimates should be included for special high dose accumulating operations such as normal maintenance, radioactive material handling, refueling, and in-service inspection. All assumptions used in the assessment should be included.

Dose rates in areas for which occupancies are given will be acceptable if they are based on generally used calculational procedures with realistic source, attenuation, and distance factors. Specific acceptance criteria are being developed.

Estimates of the number of personnel involved in various operations and various areas of the plant will be acceptable if they make realistic assumptions on use of the plant work force and include input from experience at operating reactors. The sources of this data should be cited.

All applicants are to demonstrate that the designs and the operating plans have made reasonable efforts to assure that occupational radiation exposures will be ALARA. In addition, applicants for licenses for boiling water reactor (BWR) plants are to demonstrate that they have made reasonable efforts to assure that occupational radiation exposures due to N-16 sources in turbine buildings will be ALARA. For BWR sites expected to contain more than one unit, where one or more units may be operating while others are under construction, the applicant is to provide an analysis to demonstrate that he has made reasonable efforts to assure that onsite population exposures due to such sources will be ALARA, including that to the expected construction force. The requisite analysis will be acceptable if the applicant demonstrates that he has appropriately considered the following factors and their interactions:

a. The number of reactor units, their power levels, and their expected rates of completion.
b. The relative orientations of pertinent structures and their expected completion times.
c. The overall sizes and occupational compositions of the construction force and of the plant work force.
d. The locations and orientations of components with significant inventories of N-16 with respect to structural concrete.
e. The need for additional shielding for significant source-containing components.

The staff's policy on acceptance criteria for average annual radiation exposure to plant operating personnel during predictable activities has been stated in Section 12.3, II, 1 above, in the discussion on acceptance criteria for plant radiation zoning. Also involved in the acceptance criteria for the dose assessment is the applicant's description of design and operating features for increasing accessibility to work, inspection, and sampling areas, for reducing the intensity of radiation sources that have to be worked around, for reducing the production, distribution, and retention of activated corrosion products, for reducing the time required for work in radiation fields, and for providing additional methods for reducing occupational radiation exposure. Judgment of how realistic the dose assessment is and how appropriate the ALARA operating and design features are will be used in determination of acceptability of the average annual occupational radiation exposure estimates. However, numerical acceptance criteria for total annual man-rem resulting from plant operation have not been developed. Nevertheless, the value obtained in the assessment made in this section provides a basis for judgment of the radiation protection program and a determination if ORE will be ALARA.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

The reviewer evaluates the text and scaled layout drawings of the facility to determine the manner in which the plant is zoned for radiation dose rate and to evaluate proposed occupancy. The reviewer checks the estimated occupancy factors, the calculational procedures, the dose rate values, and the in-plant airborne radioactivity concentrations. The reviewer compares the rationale and assumptions used in the dose assessment with those provided for other accepted plants and with operating experience and referenced methods. The reviewer determines whether the applicant's evaluation is thorough and realistic. The reviewer forms a judgment on whether the annual occupational radiation exposure man-rem estimate is ALARA. Under circumstances where the reviewer decided the value was not ALARA, he could request design or procedures improvements. Based on the review, RAB may also request additional information or request the applicant to modify or improve the analysis for the purpose of meeting the acceptance criteria given in Section II.

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. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

"12.4 Dose Assessment

"This section of the applicant's SAR has been reviewed to evaluate the in-plant and onsite radiation dose assessment that is an essential ingredient to determining whether the radiation protection program for ______________ nuclear plant will assure that ORE will be ALARA. The review covered the occupancy of plant radiation areas, the applicant's criteria for design dose rates in those areas, the estimated annual man-rem doses for major work functions, and the estimated annual dose at the boundary of the restricted area and to construction workers due to in-plant and onsite sources.

"The basis for acceptance of the dose assesssment includes the demonstration that consideration has been given to occupancy factors and use of various radiation zones; to radiation experience from operating plants, including doses accumulated in major work functions such as normal maintenance, in-service inspection, radwaste handling, and refueling; to the dose rate that is found, both in normally occupied areas during operation, and in areas where these functions are carried out; and to the overal annual plant man-rem value that results from the analysis of these factors.

"It is concluded that the dose assessment for _______________ nuclear plant has appropriately considered all the factors leading to a plant annual man-rem figure that demonstrates that ORE will be ALARA."

V. REFERENCES

1. ICRP Publication 22, "Implications of Commission Recommendations that Doses be Kept As Low As Readily Achievable," International Commission on Radiation Protection (1973).

2. Sixth Annual Report of the Operation of the U.S. Atomic Energy Commission's Centralized Ionizing Radiation Exposure Records and Reports System," USAEC (1974).

3. T. D. Murphy, "A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants, 1969-1973," WASH-1311, USAEC (1974).

4. Additional testimony of Dr. Morton J. Goldman on behalf of the Consolidated Utility Group (Part I), "Occupational Exposure," Docket Number RM-50-2, November 9, 1973.

5. NCRP Report 39, "Basic Radiation Protection Criteria," National Committee on Radiation Protection (1971).

6. 10 CFR Part 20, "Standards for Protection Against Radiation."

7. R. Wilson, "Man-rem Economics and Risk in the Nuclear Power Industry," NUCLEAR NEWS, February 1972.

8. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures as Low As Practicable (Nuclear Reactors)."

9. T. D. Murphy and C. H. Henson, "Occupational Radioactive Exposure at Light Water Cooled Power Reactors, 1969-1974." NUREG-75/032, USNRC (1975).

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SECTION 12.5 HEALTH PHYSICS PROGRAM

REVIEW RESPONSIBILITIES

Primary - Radiological Assessment Branch (RAB)

Secondary - None

I. AREAS OF REVIEW

The following areas of the applicant's safety analysis report (SAR) related to the health physics program are reviewed as part of the radition protection program.

1. ORGANIZATION
   a. The administrative organization of the health physics program, including the authority and responsibility of each position identified (preliminary safety analysis report, PSAR and update in the final safety analysis report, FSAR).
   b. The experience and qualifications of the personnel responsible for the health physics program and for handling and monitoring radioactive material. Reference may be made to SAR Chapter 13 as appropriate (final safety analysis report, FSAR).
   c. Information describing the implementation of Regulatory Guides 8.8, 8.10, 8.2 and 1.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

2. EQUIPMENT, INSTRUMENTATION, AND FACILITIES
   a. The criteria for selecting portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, for in-plant airborne radioactivity monitoring and sampling, for area radiation monitoring, and for personnel monitoring for normal operation, anticipated operational occurrences and accident conditions (PSAR and update in FSAR).
   b. The description of instrument storage, calibration, and maintenance facilities (PSAR and update in FSAR).
   c. The description and location of the health physics facilities (including locker and shower rooms, counting room, laboratories' decontamination facilities), protective clothing, respiratory protective equipment, and other contamination control equipment and areas (PSAR and update in FSAR).
   d. The location of items in 2a, b, and c and the description of types of detectors and monitors, sensitivity, range, and frequency and methods of calibration (FSAR).
   e. Information describing the implementation of Regulatory Guides 8.4, 8.8 and 8.9. Information describing alternatives, if such are proposed.

3. PROCEDURES

a. The description of physical and adminstrative measures for controlling access and stay time in radiation areas (FSAR).

b. The description of procedures and methods of operation for assuring that occupational radiation exposure (ORE) will be as low as is reasonably achievable (ALARA) (FSAR).

c. The description of methods, frequencies, and procedures for conducting radiation surveys (FSAR).

d. The description of the bases and methods for monitoring and control of personnel and equipment and of surface contamination including reporting practices (FSAR).

e. The description of methods and procedures for evaluating and controlling potential airborne radioactivity concentrations, for special air sampling and the issue and use of respiratory equipment, and for handling and storage of sealed and unsealed byproduct, source and special nuclear materials (FSAR).

f. The description of radiation protection training programs (FSAR).

g. Information describing the implementation of Regulatory Guides 8.8, 8.10, 8.2, 8.7, 8.9, 1.16, 1.39, and 1.8. Information describing alternatives, if such are proposed (PSAR and update in FSAR).

II. ACCEPTANCE CRITERIA

The descriptive information in the SAR is considered to be sufficient if it meets the minimum information needs set forth in Section 12.5 of the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

Apecific acceptance criteria for the areas of review given above are as follows:

1. ORGANIZATION

Acceptance will be based on a determination that the organization described, along with the duties, qualifications, and training of the individuals responsible for assuring that ORE will be ALARA are in accordance with Regulatory Guides 8.8, 8.10, 8.2, and 1.8. Alternatives will be evaluated on the basis of a comparison with the referenced Regulatory Guides.

2. EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The following shall be included in order that these items are acceptable:

a. The radiochemistry laboratory is equipped to perform routine analyses required for personnel protection, surveys, and related health physics functions.

b. The counting room (low background) is equipped and has the necessary instrumentation to perform routine counting on all plant radioactivity samples (water, air, swipe survey, etc.) in conformance with 10 CFR Part 20. Counting room equipment to normally include the following:

(1) Multichannel gamma pulse height analyzer.

(2) Low background alpha-beta proportional counter and gamma ray and alpha-beta scintillation counters. Regulatory Guide 5.9 provides specifications for Ge(Li) spectroscopy systems that may be useful for application to the gamma ray system.

(3) End window G-M type counter.

c. Portable instruments for measuring radiation or radioactivity to include:

(1) Low and high range ion chamber rate meter type instruments.

(2) Portable G-M detectors.

(3) Alpha scintillation or proportional counter rate meters.

(4) Neutron dose rate detector.

(5) Air samplers for use with particulate filters and iodine collection devices (such as charcoal cartridges) and airborne radioactivity monitors.

d. Personnel monitoring instruments to include:

(1) Friskers for detecting radioactive contamination.

(2) Self-reading low and intermediate pocket dosimeters (for early evaluation of individual doses). Performance and other requirements shall conform to Regulatory Guide 8.4, or to appropriate proposed alternatives.

(3) Count rate meters or personnel air samplers to be worn on protective clothing.

(4) Film badges and/or thermoluminescent dosimeters (TLD).

(5) Provisions for bioassay and whole body counting to meet the requirements of 10 CFR Part 20 and Regulatory Guide 8.9, or to appropriate proposed alternatives.

e. Utility-issued personnel protection equipment to include:

(1) Anti-contamination clothing.

(2) Plastic suits for liquid contamination control.

(3) Head covers, shoe covers, gloves, and safety related items.

(4) Pressure/Demand full-face-piece air line respirators.

(5) Continuous air flow two-piece plastic suits for covering whole body.

(6) Pressure demand full-face-piece self-contained breathing apparatus.

(7) Full-face mechanical filter respirators.

f. Personnel protective clothing and equipment that meet the requirements of the American Standards Institute ANSI Z88.2-1969 and the U.S. Bureau of Mines approved schedules for use in atmospheres containing radioactive materials.

g. As a minimum the following health physics support facilities or areas be provided:

(1) Portable instrument calibration and storage area. The latter should be easily accessible.

(2) Personnel decontamination area with necessary monitoring equipment. This facility should be located and designed to expedite rapid cleanup of personnel and should not be used as multiple purpose area.

(3) Facility and equipment to clean, sanitize, repair, and decontaminate personnel protective equipment, monitoring instruments, respirators, etc.

(4) A change room.

(5) Control points for entrance or exit into controlled access areas of the plant.

(6) One or more health physics stations, which may be used as the location for portable radiation survey equipment, respiratory protective equipment, personnel monitoring equipment, and contamination control supplies.

These stations and the equipment should be readily accessible and equipped to facilitate communication throughout the plant.

Acceptance will also be based on implementation of the guidance of Regulatory Guide 8.8 or the provision of acceptable alternatives.

3. PROCEDURES

Plans and procedures will be acceptable if they meet the criteria provided in Regulatory Guides 8.8, 8.10, and 8.2 or proposed appropriate alternatives. There should be provisions for a special control procedure for any zone 4 or higher area that includes a special survey of the area before entry and the development of a radiation work permit program. The work permit program should include the following: data on radiation levels in the area, allowable working time, protective clothing and respiratory protective equipment, special tools, portable shielding, and health physics and special personnel monitoring devices. For major dose accumulating functions, a post-operation review should be conducted to evaluate the effectiveness of the work permit program in assuring that occupational radiation exposures (ORE) will be as low as is reasonably achievable (ALARA). There should be provisions for supervision and control of the handling or movement of material within and from radiation or controlled access areas. Acceptance criteria for contamination control limits are being developed. There shall also be provisions for personnel monitoring procedures, bioassay, keeping records of personnel doses, and the reporting of personnel doses. 10 CFR § 20.102, .201, .401, and .407 provide the criteria for radiation surveys, personnel monitoring, bioassay, record keeping, and reporting. Guidance regarding these areas is provided by Regulatory Guide 8.2 (surveys and personnel monitoring), 8.3 (personnel monitoring equipment), 8.9 (bioassay), and 1.16, 8.2, 8.7 (record keeping and reporting), and 8.8 (decontamination, inspection, radiation protection program, and operations).

The acceptability of the health physics program will also be based on provisions for the indoctrination and personnel training and retraining programs. Regulatory Guides 8.8, 8.10, and 1.8 provide information regarding these areas. Section 19.12 of 10 CFR Part 19 requires instruction of personnel on radiation protection. There should be a regular review of the radiation protection program, which should include updating procedures, equipment, and facilities where improvements are possible. The program should include regular audits to determine where occupational radiation exposures are occurring and to review possible methods for reducing these exposures. With regard to plant cleanliness, which is critical where radioactive material is concerned, Regulatory Guide 1.39 discusses housekeeping requirements that are applicable to operation as well as construction. Finally 10 CFR Part 70 provides the guidance on special nuclear, source, and byproduct materials.

III. REVIEW PROCEDURES

The information furnished in the SAR is reviewed for completeness in accordance with the "Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants," Revision 2.

RAB reviews all the areas discussed in I., evaluating acceptability by making the referenced comparisons with Regulations, Regulatory Guides, and industry standards. These can be summarized as follows:

1. The organizational position, functional responsibilities, experience, and qualifications of persons responsible for the health physics program. The plant organization, the functional responsibilities, and the qualifications of personnel are the primary responsibility of the Quality Assurance Branch, and are reviewed as part of Chapter 13. RAB reviewers of these items for the health physics/radiation protection function and personnel qualifications will communicate their findings to the QAB in this area of review.

2. The equipment necessary to measure radioactivity and radiation fields and exposures, including the number, type, range, sensitivity, calibration method and frequency, availability, and planned use of portable, fixed, laboratory, and personnel monitoring instrumentation.

3. The health physics facilities and associated protective equipment for controlling ORE and contamination.

4. The training and indoctrination program and health physics instruction manuals, as well as the respiratory protective equipment fitting program. Plant procedures are the primary responsibility of the Operator Licensing Branch which reviews in Chapter 13 (Section 13.5), such formal procedures as the plant radiation protection procedures. RAB reviewers of health physics/radiation protection procedures should communicate any problems with specific procedures to the OLB.

5. The procedures to control storage and movement of radioactive material, to control exposures, and to control contamination. Where these procedures are part of the formal plant operating procedures, the review will include informing the OLB of any problems as in 4 above.

Based on the review, RAB may request additional information or request the applicant to modify his submission in order to meet the acceptance criteria described in Section II.

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. The report will include a summary of the applicant's coverage, the staff's basis for review and acceptance criteria, and the findings of the review. The following is a brief representation of the evaluation findings:

> "12.5 Health Physics Program
>
> "This section of the applicant's SAR has been reviewed to determine that the health physics program will assure that occupational radiation exposure will be as low as is reasonably achievable. The review covered the organization of the program, the qualification of personnel, the equipment and instrumentation related to the program, the

description of all health physics related facilities, and the procedures that are related to the control of radioactive contamination and occupational radiation exposure. The review includes the applicant's description of conformance to applicable industry standards, Regulations and Regulatory Guides, or this provision of acceptable alternatives.

"The basis for acceptance of the program organization and personnel qualifications is conformance with applicable Regulatory Guides. Acceptability of personnel monitoring, reporting and recording of information, and of radiation surveys is based on conformance to the requirements of 10 CFR Part 20, and to appropriate Regulatory Guides. The program contains an effective radiation work permit program, as well as procedures for contamination control that are consistent with assuring that occupational radiation exposures will be as low as is reasonably achievable.

"It is concluded that the health physics program, including organization, equipment and instrumentation, laboratory facilities, and methods and procedures related to personnel protection and contamination control conform to the Commission's Regulations and the applicable Guides and industry standards and is acceptable."

V. REFERENCES

1. 10 CFR Part 20, "Standards for Protection Against Radiation."

2. 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections."

3. Regulatory Guide 1.8, "Personnel Selection and Training."

4. ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," American National Standards Institute (1972).

5. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures as Low as Practiable (Nuclear Reactors)."

6. Regulatory Guide 8.6, "Standard Test Procedures for G-M Counters."

7. Regulatory Guide 8.XX, "Control of Radioactive Surface Contamination on Material, Equipment and Facilities to be Released for Uncontrolled Use." (in preparation)

8. Regulatory Guide 1.39, "Housekeeping Requirements for Water Cooled Nuclear Power Plants."

9. USBM-23, "Respiratory Protective Services for Use in Atmospheres Containing Radioactive Materials," U.S. Bureau of Mines (1973).

10. Regulatory Guide 8.7, "Occupational Radiation Exposure Records System."

11. Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters."

12. Regulatory Guide 8.3, "Film Badge Performance Criteria."

13. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

14. ANSI Z88.2-1969, "Procedures for Respiratory Protection," American National Standards Institute (1969).

15. 10 CFR Part 20, Appendix B, Table 1, "Concentrations in Air and Water Above Natural Background."

16. Regulatory Guide 1.16, "Reporting of Operating Information."

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

18. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equation, and Assumptions for a Bioassay Program."

19. 10 CFR Part 70, "Special Nuclear Material."

20. Regulatory Guide 5.9, "Specifications of Ge(Li) Spectroscopy Systems for Natural Protection Measurements - Part I: Data Acquisition."

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SECTION 13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

The applicant's management and technical support organization as described in his safety analysis report (SAR) is reviewed. These organizational units are typically located offsite, in contrast to the plant operating organization onsite, reviewed under Standard Review Plan (SRP) 13.1.2. The review of the management and technical support organization includes its responsibilities and the technical qualifications of the organization, including personnel, to engage in the activities proposed in the application. These activities may include items such as facility design, design review, design approval, construction management, testing, and operation of the facility.

In the preliminary safety analysis report (PSAR), the description of the management and technical support organization should include organization charts reflecting the applicant's current headquarters and engineering structure, and planned modifications and additions to it to reflect the added functional responsibilities associated with the addition of the nuclear plant to the applicant's power generation capacity. These added responsibilities should be identified and should include those listed in (1), (2), and (3) below. The description should show how these reponsibilities are delegated and assigned within and from the headquarters staff, identify and describe the qualifications of the working or performance level organization unit responsible for each. A schedule, relative to the construction permit (CP) and expected fuel loading date for each unit covered by the application, for implementing these responsibilities should be included along with an estimate of the number of persons expected to be assigned to the various units at each stage of the schedule.

1. Design and Construction Responsibilities (Project Phase)

These are functions that are decided and defined almost totally prior to submittal of the application to the Nuclear Regulatory Commission (NRC) and may continue until the plant is turned over to the operating organization. The extent and assignment of these functions are generally contractual in nature and determined by the applicant. (Note: QA aspects should be described in Section 17.1.)

a. Principle site-related engineering work such as meteorology, geology, seismology, hydrology, demography, and environmental effects.
b. Design of plant and ancillary systems.
c. Review and approval of plant design features.
d. Site layout in respect to environmental effects and security provisions.
e. Development of safety analysis reports.
f. Material and components specification review and approval.
g. Procurement of materials and equipment.
h. Management and review of construction activities.

2. Preoperational Responsibilites

These are functions which should be substantially accomplished before preoperational testing begins and generally before submittal of the final safety analysis report (FSAR).

a. Development of human engineering design objectives and design phase review of proposed control room layouts.
b. Development and implementation of staff recruiting and training programs.
c. Development of plans for initial testing.
d. Development of plant maintenance programs.

3. Technical Support for Operations

Technical services and backup support for the operating organization should become available prior to the initial testing program and continue throughout the life of the plant. The special capabilities that should be included are:

a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and materials, and instrumentation and controls engineering.
b. Plant chemistry.
c. Health physics.
d. Fueling and refueling operations support.
e. Maintenance support.

The PSAR should also identify general qualification requirements in terms of numbers, educational backgrounds, and experience for identified positions or classes of positions, and personnel resumes of assigned persons holding key or supervisory positions in disciplines or job functions unique to the nuclear field or this project relative to items (1) and (2) above. For identified positions or classes of positions that have functional responsibilities for other than the identified application, the expected proportion of time assigned to the other activities should be described.

In the final safety analysis report, the description should include organization charts showing the management and technical support headquarters structure, summarize the degree to which the preoperational responsibilities of (2) above have been accomplished, and describe the provisions which have been made for technical support for operations, per (3) above.

The FSAR should identify qualification requirements for headquarters staff personnel, in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations. In addition, the FSAR should include resumes of individuals already employed by the applicant to fulfill responsibilities identified in (3), above, including that individual whose job position corresponds most closely to that identified as "Engineer in Charge" at Section 4.6.1 of ANSI N18.1-1971.

II. ACCEPTANCE CRITERIA

This section of the SAR should demonstrate the applicant's recognition of the commitment to fulfill corporate responsibilities to deal with safety-related problems connected with the proposed addition of nuclear generating capacity. It should be significant contributory evidence as to the technical qualifications of the applicant, as required by 10 CFR §50.34(a)(9) and (b)(7).

Criteria for acceptability include the following:

1. Evidence of this commitment should include additions to the applicant's headquarters staff, as distinct from reliance upon personnel of vendors and consultants, and adequate personnel to support the identified responsibilities for the project.

2. Recommendations contained in the AEC document WASH-1130, Revised, Section IV-A, "Headquarters Staff," are generally acceptable for this section of the SAR.

3. Qualifications of the "Engineer in Charge" should meet or exceed those given in Section 4.6.1 of ANSI N18.1-1971.

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.

In the review and evaluation of the subject matter of this section of the SAR, the following specific points should be taken into consideration. The management and technical support headquarters structure, as demonstrated by organization charts and descriptions of functions and responsibilities, should be free of ambiguous assignments of primary responsibility. Design and construction responsibilities should be reasonably well defined in both numbers and experience of persons required to implement their responsibilities. The reviewer must recognize, however, that there are many acceptable ways to define and delegate job responsibilities. In respect to item (3) above, the applicant's plans for headquarters staffing may not yet be firm. It is acceptable, therefore, if these plans are not fully specific in terms of numbers of people, provided the commitment made is sufficiently firm to assure the responsibility can be met. It should also be noted (with respect to criterion (2) above) that some additions to staff may already have taken place prior to submittal of the PSAR. Credit should be given for such prior additions to the extent they involve job responsibilities identified in I above. Variations in staffing may also be expected between applicants who lack prior experience with nuclear power plant operation and those who have such experience. It is important that the reviewer assure himself that applicants in the former category do not underestimate the magnitude of the task. The reviewer should be alert to the possibility that excessive work loads may be placed upon too small a number of individuals.

If the application involves the addition of more than one unit, the reviewer should assure that headquarters staffing plans take this fact into account. This is particularly important if additional units are scheduled to come on line at intervals of about one year or less, since the shakedown period for the operation of a new plant can be expected to produce quite heavy workloads. In some of these cases, the applicant may plan to bolster the plant staff organization during such periods, so that it is necessary to evaluate headquarters staffing plans in conjunction with those for the plant staff organization.

The reviewer should assess the degree of participation in the "project phase" of that headquarters group that typically has plant operating (generating) responsibility. Interfaces between such a group and those with project engineering responsibilities should be examined. If project engineering groups are assigned any of the preoperational responsibilities identified in (2) above, their qualifications to represent the operator's viewpoint should be examined.

Schedules for staffing should be provided with a time scale that is fixed with regard to the expected fuel loading date. Criteria useful for judging the adequacy of such schedules are in WASH-1130. The reviewer must also be alert to the fact that some of the preoperational responsibilities identified must be discharged before designs are "frozen," e.g. design reviews for operability and maintainability.

An important element of the review of this section of the SAR is the evaluation of the qualification requirements and qualifications (FSAR) of technical support personnel. This group probably will be involved in the independent review of plant operations. Their qualifications should, therefore, be judged in the light of the review mechanism and responsibilities described in SAR section 13.4.1. Additional guidance on this point in ANSI N18.7-1972, as Section 4.2, should be used, as well as ANSI N18.1-1971, Section 4.6.1.

The review procedure for this section consists, therefore, of:

1. An examination of the information submitted to determine that all subject matter identified in I above has been addressed, and
2. A comparison of the information with the acceptance criteria of II above in the light of the specific points set forth earlier in this section.

In addition, if the applicant, as of the time the review takes place, has had experience in the operation of a previously licensed nuclear power plant, the reviewer may seek independent information relative to headquarters staffing and qualifications through the Office of Inspection and Enforcement; e.g., by discussion with inspection personnel, or review of inspection reports.

The reviewer then determines, based on the foregoing, the overall acceptability of the applicant's management and technical support organization and staffing plans.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support conclusions of the following type, to be used in the staff's safety evaluation report:

1. CP Safety Evaluation Report. "The applicant has described his reponsibilities for the design and construction of the facility and his plans for management and technical support for such activities and for utilization of this organization to support operations. These plans have been reviewed and give adequate assurance that such persons are technically qualified, that the applicant has identified preoperational responsibilities for the operation of the plant and has provided reasonable assurance that they will be satisfactorily discharged, and that the applicant is technically qualified to engage in the proposed activities."

2. OL Safety Evaluation Report. "The applicant has described the extent to which his preoperational respo ibilities have been accomplished and described his means for providing technical support for the plant staff during operation of the facility. These measures have been reviewed and provide reasonable assurance that the technical support will be provided by technically qualified persons, and provide in conjunction with the findings of Sections 13.1.2 and 13.1.3 of this Safety Evaluation Report that the applicant is technically qualified to operate the facility safely."

V. REFERENCES

I. "Utility Staffing and Training for Nuclear Power," WASH-1130, Revised, U.S. Atomic Energy Commission, June 1973.

2. ANSI N18.1-1971, "Selection and Traning of Nuclear Power Plant Personnel," American National Standards Institute (1971).

3. ANSI N18.7-1972, "Standard for Administrative Controls for Nuclear Power Plants," American National Standards Institute (1972).

4. 10 CFR §50.34(a)(9) and (b)(7).

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SECTION 13.1.2 OPERATING ORGANIZATION

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The applicant's operating organization, as described in his safety analysis report (SAR), is reviewed. This section of the SAR should describe the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant. Specific information to be included is as follows:

1. An organization chart showing the title of each position, the number of persons assigned to common or duplicated positions, the number of operating shift crews, and the positions for which reactor operator and senior reactor operator licenses are required. For multi-unit stations, the organization chart (or additional charts) should clearly reflect changes and additions as new units are added to the station.

2. The schedule, relative to fuel loading for each unit, for filling all positions should be displayed.

3. The functions, responsibilities, and authorities of plant positions corresponding to the following should be described.
   a. Overall plant management.
   b. Operations supervision.
   c. Operating shift crew supervision.
   d. Licensed operators.
   e. Non-licensed operators.
   f. Technical supervision.
   g. Radiation protection supervision.
   h. Instrumentation and controls maintenance supervision.
   i. Equipment maintenance supervision.
   j. Quality assurance and quality control supervision.

For each position, where applicable, required interfaces with offsite personnel or positions identified in SAR 13.1.1 should be described. Such interfaces include defined lines of reporting responsibilities, e.g., from the plant manager to his immediate superior, as well as functional or communication channels. In the final safety analysis report (FSAR), the following should also be described: (1) the line of succession of authority and responsibility for overall station operation through at least three persons in the event of unexpected contingencies of a temporary nature, and (2) the delegation of authority to operating supervisors and to shift supervisors, including the authority to issue standing or special orders.

If the station contains or is planned to contain power generating facilities other than those relating to the application in question and including fossil fueled units, this section should also describe interfaces with the organizations operating such other facilities. The description should include any proposed sharing of persons between the units and the proportion of their time that they will routinely and non-routinely be assigned to the other unit.

4. The position titles, applicable operator licensing requirements for each, and the total numbers of people planned to man each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown modes. Shift crew staffing plans unique to refueling operations should be described. The proposed means of assigning shift responsibility for implementing the radiation protection program on a round-the-clock basis should also be described.

RSB reviews the function, responsibility, authority, and reporting line for the radiation protection supervision, against acceptance criteria stated in Standard Review Plan 12.5.

II. ACCEPTANCE CRITERIA

This section of the SAR should demonstrate the applicant's commitment to (PSAR) and implementation of (FSAR) plans to staff the onsite operating organization and to define and delegate responsibilities to provide assurance that the plant can be operated safely. It should be significant contributory evidence as to the technical qualifications of the applicant, as required by 10 CFR §50.34(a)(9) and (b)(7).

Criteria for acceptability include the following:

1. Recommendations contained in the AEC document WASH-1130, Revised, Section IV-B, "Plant Staff," are generally acceptable guides for this section of the SAR.

2. The requirements of ANSI N18.7-1972, Section 3.3, "Operating Organization," should be met.

3. Responsibilities and authorities of operating organization personnel should conform to the requirements of ANSI N18.7-1972, Section 5.1, "Rules of Practice," and Section 4.5, "Onsite Review."

4. a. Assignments of personnel meeting ANSI N18.1-1971 qualifications, Section 4.3.1 or Section 4.5.1, should be made to onsite shift operating crews in numbers not less than the following:

For a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For a multi-unit station, each shift crew should have at least three persons per licensed unit at all times, plus one additional person per operating unit.

b. Operator license qualifications of persons assigned to operating shift crews should be as follows:

(1) A licensed senior operator who is also a member of the station supervisory staff should be onsite at all times when at least one unit is loaded with fuel.

(2) For any station with more than one reactor containing fuel, (1) the number of licensed senior operators onsite at all times should not be less than the number of control rooms from which the fueled units are monitored, and (2) the number of licensed senior operators should not be less than the number of reactors operating.

(3) For each reactor containing fuel, there should be at least one licensed operator in the control room at all times. Shift crew compositions should be specified such that this condition can be satisfied independently of licensed senior operators assigned to shift crews to meet the criteria of (1) and (2) above.

(4) For each control room from which one or more reactors are in operation, an additional operator should be onsite and available to serve as relief operator for that control room. Shift crew compositions should be specified such that this condition can be satisfied independently of (1), (2), and (3), and for each such control room.

c. Radiation protection qualifications of at least one person on each operating shift should be as follows:

The management of each station having one or more units containing fuel should either, (1) qualify and designate at least one member of each shift operating crew to implement radiation protection procedures, including routine or special radiation surveys using portable radiation detectors, use of protective barriers and signs, use of protective clothing and breathing apparatus, performance of contamination surveys, checks on radiation monitors, and limits of exposure rates and accumulated dose, or (2) assign a health physics technician to each shift, such assignment to be in addition to those assigned to shift operating crews in accordance with (a) and (b) above.

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.

In the review and evaluation of the subject matter of this section of the SAR, the following specific points should be taken into consideration.

Plant staff organizational structures are not rigidly fixed; however, experience has shown that certain components are common to and necessary for all plants. Among these are an operational group, an onsite technical support group, and a maintenance group, under the direction and supervision of a plant manager. For multi-unit sites, consideration must be given to the possibility that off-shift supervision may be stretched too thin to provide effective supervision. In particular, the operations manager function should not be stretched to cover more than two units. For on-shift persons, there should be manpower available in excess of four full operating shift crews so that excessive overtime is not routinely scheduled for these crews. For multi-unit sites, a shift supervisor should be designated in charge of the station during those periods of time when senior level supervision is not on site.

The structure of onsite technical support and maintenance groups may depend somewhat on headquarters staffing and the division of effort between onsite and offsite personnel.

With respect to shift assignments, the reviewer should determine that persons assigned to implement the radiation protection program are adequately trained and qualified for this task, and that it is a clearly defined part of the job function. Assignments to shift crews for refueling operations should be examined to assure adequate supervisory attention is given to all operations associated with fuel handling.

The review procedure for this section consists, therefore, of:

1. An examination of the information submitted to determine that all subject matter identified in I above has been addressed.
2. A comparison of the information with the acceptance criteria of II above in the light of the specific points set forth earlier in this section.

In addition, if the applicant, as of the time the review takes place, has had experience in the operation of a previously licensed nuclear power plant, the reviewer may seek independent information relative to plant staffing and qualifications through the Office of Inspection and Enforcement; e.g., by discussion with inspection personnel, or review of inspection reports.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicant's operating organizations and plant staffing plans.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support conclusions of the following type, to be used in the staff's safety evaluation report:

"The applicant has described the assignment of plant operating responsibilities; the reporting chain up through the chief executive officer of the company (applicant); the proposed size of the regular plant staff, total and by major subdivisions; the character and responsibilities of each major plant staff group; and the proposed shift crew complement for single unit or multiple unit operation. This information has been reviewed, and it is the conclusion of the staff that proposed organization is acceptable. Major elements of this organization including key positions, license requirements, and shift composition will be incorporated in the administrative controls section of the technical specifications."

V. REFERENCES

1. "Utility Staffing and Training of Nuclear Power," WASH-1130, Revised, U.S. Atomic Energy Commission, June 1973.

2. ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," American National Standards Institute (1971).

3. ANSI N18.7-1972, "Standard for Administrative Controls for Nuclear Power Plants," American National Standards Institute (1972).

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SECTION 13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - Radiological Assessment Branch (RAB)

I. AREAS OF REVIEW

The qualifications of the applicant's plant personnel, as described in his safety analysis report (SAR), are reviewed. This section of the SAR should describe the education, training, and experience requirements established by the applicant for filling each management, operating, technical, and maintenance position category in the operating organization described in SAR Section 13.1.2. At the final safety analysis report (FSAR) stage, this section should in addition provide evidence, in the form of personnel resumes, that the initial selections made for key management and supervisory positions down through the shift supervisory level, conform to those requirements. The RAB reviews the qualifications of the reactor protection supervisor against the acceptance criteria in Standard Review Plan 12.5.

II. ACCEPTANCE CRITERIA

Regulatory Guide 1.8, "Personnel Selection and Training," sets forth the staff position on plant personnel qualifications and indicates that the criteria for selection (qualifications) contained in ANSI N18.1-1971 are generally acceptable.

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.

At the construction permit (CP) stage, the applicant will generally not have made selections for plant staff positions if the application is for a new station. The review procedure, therefore, is to examine this section of the SAR for a commitment on the part of the applicant to conform to the stated acceptance criteria. This commitment should be unambiguous and should appear also in the applicant's proposed technical specifications. When such is the case, the review is completed.

Where a clear comparison cannot be made between the proposed plant staff positions and those defined at Section 4 of ANSI N18.1, the applicant should list each position on his plant staff and designate the corresponding position of Section 4 of ANSI N18.1, or describe in detail his proposed qualifications for each position on his plant staff.

The review of the FSAR, at the operating license (OL) stage, consists first of the same examination as made at the CP stage, and second of an analysis of each resume. The reviewer should make an explicit comparison of the educational and experience records obtained from each resume with the corresponding requirements set forth for the applicable position in Section 4 of ANSI N18.1 or other approved qualifications. "Applicable experience" should be judged in the light of the position responsibility. Credit for experience which is not directly applicable should be weighted to a degree commensurate with its applicability. The bases for such weighted judgments should be documented in the reviewer's notes. When the resumes for persons initially selected to fill all key management and supervisory positions from the plant superintendent down through each shift supervisor have been analyzed and these persons found to have qualifications equal to or greater than those specified in ANSI N18.1, the review is completed.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support conclusions of the following type, to be used in the staff's safety evaluation report:

"The applicant has described the qualification requirements for members of his plant staff and (at the FSAR stage) submitted personnel resumes for key management and supervisory positions.

"This information has been reviewed and we conclude that the qualifications conform to staff positions set forth in Regulatory Guide 1.8 and (at the FSAR stage) that the applicant's implementation of these requirements in the selection of initial appointees to the plant staff conforms to these requirements.

"The same qualification requirements will apply to replacement personnel as a condition of the license to be incorporated in the administrative controls section of the technical specifications."

V. REFERENCES

1. Regulatory Guide 1.8, "Personnel Selection and Training."

2. ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," American National Standards Institute (1971).

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 13.2 TRAINING

REVIEW RESPONSIBILITIES

Primary - Operator Licensing Branch (OLB)

Secondary - None

I. AREAS OF REVIEW

The applicant's plant personnel training program, as described in his safety analysis report (SAR) is reviewed. This section of the SAR should contain the description and scheduling of the training program for initial appointees to the plant staff. The program descriptions should include the following:

For the preliminary safety analysis report (PSAR):

1. The proposed subject matter of each course, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instruction, and the position titles for whom the course is given.

2. Reactor operations experience training by nuclear power plant simulator or assignment to a similar plant, including length of time (weeks), and identity of simulator and plant.

3. A commitment to conduct an onsite formal training program and on-the-job training before initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to Section 55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

   a. Individuals with no previous experience.
   b. Individuals who have had nuclear experience at facilities not subject to licensing.
   c. Individuals who hold, or have held, licenses for comparable facilities.

5. Means for evaluating the training program effectiveness for all employees. For license applicants this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to Section 55.25(b) of 10 CFR Part 55.

This program description should also include a chart to show the schedule of each part of the training program for each functional position identified in SAR Section 13.1.2. The time should be relative to expected fuel loading and should also display the preoperational test period, and the expected time for examinations for licensed operators prior to plant criticality.

In the final safety analysis report (FSAR):

1. The proposed subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instructions, and the position titles for which the course is given.

2. Reactor operations experience training by nuclear power plant simulator or asignment to a similar plant, including length of time (weeks), and identity of simulator and plant.

3. The details of the onsite training program, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks in full time attendance), the organization teaching the course or supervising instructions, and the position titles for which the course is given. The program should distinguish between classroom training and on-the-job training, before and after the initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to Section 55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

    a. Individuals with no previous experience.
    b. Individuals who have had nuclear experience at facilities not subject to licensing.
    c. Individuals who hold, or have held, licenses for comparable facilities.

5. Means for evaluating the training program effectiveness for each employee. For applicants for license examinations prior to criticality, the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to Section 55.25(b) of 10 CFR Part 55.

The program description section should also include a chart to show the schedule of each part of the training program for each functional position identified in FSAR Section 13.1.2. The time scale should be relative to expected fuel loading and should also display the preoperational test period, expected time for examinations for licensed operators prior to criticality, and expected time for examinations for licensed operators after criticality (13.2.1.2).

The description should delineate clearly the extent to which the training program has been accomplished at the approximate time of submittal of the FSAR. Contingency plans for additional training for individuals to be licensed prior to criticality should be described in the event fuel loading is subsequently delayed from the date indicated in the FSAR.

The FSAR should describe the applicant's plans for retraining of plant staff personnel including requalification training for licensed operators and senior operators (13.2.2). The detailed description of the proposed requalification training program should show how it will meet the requirements of 10 CFR Part 55, Appendix A (13.2.2.1). The FSAR should also identify the additional position categories on the plant staff for which retraining will be provided, and should describe the nature, scope, and frequency of such retraining (13.2.2.2).

II. ACCEPTANCE CRITERIA

The SAR should demonstrate that the training provided, or to be provided, for each position on the plant staff will be adequate to provide assurance that all plant staff personnel qualification requirements will be met as of the time needed, i.e., prior to operator license examinations, prior to fuel loading, or prior to appointment or reappointment to the position.

Criteria for acceptability are:

1. The training requirements and guidance set forth in the following regulations and regulatory guides should be met or acceptable alternatives should be presented.

10 CFR Part 50	All Employees
10 CFR Part 19	All employees
Regulatory Guide 1.8	All employees
10 CFR Part 55	Licensed Operators and Senior Operators
AEC Licensing Guide, "Operators' Licenses," WASH-1094	Licensed Operators and Senior Operators

2. Formal segments of the initial training program should be substantially completed when the preoperational test program begins, with the exception of a brief formal refresher just prior to operator examinations.

3. The number of persons for whom training is planned in preparation for senior operator and operator examinations prior to criticality should be sufficient to assure that applicable technical specification conditions with respect to the number of licensed operators on shift crews can be met from the time of initial fuel loading of the first unit, with due allowance given for examination contingencies and the need to avoid planned overtime for supervisory personnel during the startup phase in order to meet technical specification conditions.

4. The licensed operator requalification training program should adequately implement the requirements of 10 CFR Part 55, Appendix A.

5. Refresher training for non-licensed personnel should be periodic and not less than biannual and should include at a minimum refresher instruction on administrative, radiation protection, emergency, and security procedures.

III. REVIEW PROCEDURES

Preparation for the review of this section of the SAR should include familiarization with 10 CFR Part 50, 10 CFR Part 55, 10 CFR Part 19, Regulatory Guide 1.8, and WASH-1094, "Operators' Licenses." The reviewer may use training course descriptions obtained independently from vendors.

The review procedure for this section consists of:

1. A careful examination of the information submitted to determine that all subject matter identified in I above has been addressed, and

2. A detailed comparison of the information with the acceptance criteria of II above.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicants plant staff training plans.

IV. EVALUATIONS FINDINGS

The reviewer should verify that the information presented and his review support an evaluation findings statement of the following type, to be used in the staff's safety evaluation report:

"The training program and schedule for all staff members is acceptable for the preoperational test program, for operator licensing examinations, and for fuel loading.

"Plans for retraining and replacement training conform to the (named) regulatory position, or equivalent, and the commitment to replacement training will be incorporated in the Administrative Controls section of the applicant's Technical Specifications."

The evaluation findings should also include the following information:

1. Identity of organization(s) who will conduct or have conducted designated parts of the initial training program.

2. Identity of major training subject areas.

V. REFERENCES

1. 10 CFR Part 50, "Licensing of Production and Utilization Facilities."

2. 10 CFR Part 55, "Operators' Licenses."

3. 10 CFR Part 19, "Notices, Instructions and Reports to Workers, Inspections."

4. Regulatory Guide 1.8, "Personnel Selection and Training."

5. AEC Licensing Guide, "Operators' Licenses," WASH-1094, November 1965.

6. "Utility Staffing and Training for Nuclear Power," WASH-1130, USAEC, Revised June 1973.

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SECTION 13.3 EMERGENCY PLANNING

REVIEW RESPONSIBILITIES

Primary - Industrial Security and Emergency Planning Branch (ISEPB)

Secondary - None

I. AREAS OF REVIEW

The applicant's emergency planning, as described in his safety analysis report (SAR), is reviewed by ISEPB. The review of this section of the SAR involves evaluation of evidence of preliminary planning (in the preliminary safety analysis report, PSAR) or substantive evidence of planning (in the final safety analysis report, FSAR) for emergency preparedness directed primarily at situations involving real or potential radiological hazards.

At the PSAR stage the review covers each of the seven sub-parts A-G of 10 CFR Part 50, Appendix E, Part II. Particular attention is given to the following areas, applicable to the sub-parts indicated.

With respect to sub-part B, the designation by the Governor of the state in which the facility is to be located of an agency that has the primary responsibility for planning for radiological emergency response in the (public) environs of the plant is verified and evidence of the arrangements that have been made by the applicant with this agency for the preparation of coordinated emergency response plans in the environs of the facility is reviewed.

With respect to sub-part C, one of the protective measures considered is the evacuation of persons from the exclusion area and from potentially affected sectors of the environs. An analysis of the implications for evacuation of the most severe design basis accident postulated is reviewed to assure that it includes explicit findings or information necessary for emergency planning.

With respect to sub-part E, the review includes a determination that at least two off-site hospital facilities are identified, with evidence that preliminary contacts have established agreements and potential capability to receive and treat individuals affected by radiological emergencies.

At the FSAR stage, a comprehensive emergency plan document is reviewed. The emergency plan should demonstrate implementation of the objectives and requirements of 10 CFR Part 50, Appendix E, Parts I, III, and IV.

II. ACCEPTANCE CRITERIA

At the PSAR stage, this section is considered acceptable (1) if it conforms to the requirements of 10 CFR Part 50, Appendix E, Part II, (2) if the emergency planning information, submitted in accordance with section 13.3 of Revision 2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", is consistent with facility design features, analyses of postulated accidents, and characteristics of the proposed site location, and (3) if it provides reasonable assurance that appropriate protective measures can be taken in the event of a serious accident within and beyond the site boundary.

ISEPB considers that the last of the above is satisfied if preliminary planning and analysis shows that there is reason to expect that the emergency plans for the facility can be designed to meet, at minimum, the following objectives, based upon calculated radiological dose consequences of an airborne release following the most serious design basis accident:

1. Completion of evacuation of persons within the exclusion area within two hours from the onset of release. In this connection, ISEPB considers that the required assurance cannot be given if non-plant related activities, e.g. recreational activities, are permitted anywhere within the exclusion area where siting dose guidelines of 10 CFR Part 100 might be reached in less than two hours, as shown by calculation.

2. Completion of evacuation of persons within 45 sectors of the environs beyond the exclusion radius boundary within two hours from the onset of release, or within the times calculated as a function of distance for a potential dose to reach the upper limit of the range of protective action guide levels to be adopted as warranting evacuation as a protective measure for the general public, whichever is larger at each distance considered. ISEPB considers that the minimum range of acceptable distances within which this determination is to be made is the distance at which the referenced protective action guide level is reached in 8 hours from the onset of release.

3. Completion of initial accident assessment measures, including dose projection, and notification to offsite authorities within fifteen minutes or within the calculated time at which the dose at the exclusion radius would reach the lower limit of the range of protective action guide levels to be adopted (for evacuation), whichever is larger.

At the FSAR stage, the organization and content of a generally acceptable emergency plan for a nuclear power plant to implement the requirements of 10 CFR Part 50, Appendix E, Parts III and IV, is given in Appendix A to this standard review plan.

III. REVIEW PROCEDURES

At the PSAR stage, the review consists of an evaluation of the information submitted by comparison of this information with the foregoing Acceptance Criteria. The reviewer should determine that all of these criteria are satisfied, exercising his judgment as to the reasonableness and adequacy of the qualitative factors involved, in the light of emergency planning objectives.

The reviewer should gain familiarity with the proposed site, including the exclusion area, low population zone, demography, and land use factors, with the proposed plant design and layout, and with the calculated dose consequences of design basis accidents postulated by the applicant. To this end the reviewer should examine relevant sections of thePSAR, particularly Chapters 1.0, 2.0 and 15.0. This information may be supplemented by the use of United States Geological Survey grid maps, road maps, and a personal visit to the site by the reviewer.

With respect to the applicant's analysis and findings relative to emergency planning for evacuation, the reviewer should assess the credibility and adequacy of time factors presented by the applicant in the light of emergency operations experience and should analyze them to determine that the time estimates or allocationsfor sequential actions are consistent with the objectives and criteria set forth in II above. In addition he should assure that calculational methods and assumptions used by the applicant for dose projections are generally consistent with those found acceptable to the staff for purposes of demonstrating conformance with 10 CFR Part 100 siting criteria. Consultation with other members of the staff may be necessary to gain this assurance.

For cases in which the reviewer determines that there are site-related population, road network, or land use factors, or unique accident considerations which present potential problems for emergency planning, he may develop and recommend independent techniques to determine certain acceptable emergency plan design objectives for that site.

At the FSAR stage, the review consists of a careful examination of the applicant's emergency plan. The requirements of 10 CFR Part 50, Appendix E, Parts III and IV, and the elements of emergency planning set forth in Appendix A to this standard review plan should be used as checklists for detailed comparisons with the applicant's plan.

IV. EVALUATION FINDINGS

At the conclusion of the PSAR stage review, a finding of acceptability of the applicant's defined low population zone with respect to the definition in 10 CFR § 100.3(b), should be transmitted to the Accident Analysis Branch.

The evaluation finding for this section at the PSAR stage should be substantially equivalent to the following statement:

"The applicant has described his preliminary plans for coping with emergencies. An onsite Emergency Coordinator will direct the implementation of the Emergency Plan in accordance with detailed written emergency procedures. Initial contacts and arrangements have been made with the following agencies: (listing by name). The (identity of state agency) has been identified as having primary responsibility for radiological emergency planning in the environs of the proposed facility.

"In-plant monitors will provide the first indication of a radiological emergency. Provisions will be made for surveys by portable meters and air sampling devices on a timely basis. The plant control room has been designed for continuous occupancy and will be the principal emergency control center. One alternate center will be designated. Emergency kits will be stored at the primary assembly area. Decontamination facilities and a first aid room will be provided. Arrangements have been initiated with area hospitals to treat contaminated injury cases. All plant personnel will receive training in emergency procedures and periodic drills will be conducted.

"Analyses have been performed to confirm the practicability of taking protective measures, including evacuation, within and beyond the site boundary during the expected lifetime of the plant, and appropriate criteria have been identified for the design of an acceptable emergency plan.

"We have reviewed the applicant's preliminary plans for coping with emergencies and consider that they meet the requirements of 10 CFR Part 50, Appendix E, and are acceptable."

The evaluation finding for this section at the FSAR stage should be substantially equivalent to the following:

"The applicant has formulated and submitted an Emergency Plan which describes the program for coping with emergencies within and beyond the site boundary. The plan includes a description of the organizational control extending from the on-site emergency organization to off-site agencies, specific emergency measures to be taken as indicated by defined accident assessment techniques, including protective measures, for persons subject to potentially excessive radiological exposures, and facilities and supplies needed for coping with emergencies, including redundant communications equipment. The plan also describes arrangements made for providing necessary medical attention for persons with contaminated injuries, and provisions for maintaining an adequate emergency preparedness posture throughout the expected lifetime of the plant through training, exercises, and drills.

The plan has been determined to be acceptably coordinated with the radiological response planning of the (state name and agency identification).

We have reviewed the applicant's Emergency Plan and consider that it meets the requirements of 10 CFR Part 50, Appendix E, is responsive to the specific requirements of the staff, and provides an adequate basis for an acceptable state of emergency preparedness. Details and procedures to implement the Emergency Plan require inspection and evaluation by the Directorate of Regulatory Operations prior to the issuance of an Operating License."

Modifications or additions to this statement may be necessary to highlight features of the review of emergency planning which are unique to the plant or site in question.

V. REFERENCES

1. Appendix A, "Emergency Plans for Nuclear Power Plants", attached hereto.

2. 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities".

3. Regulatory Guide 1.70, "Standard Format and content of Safety Analysis Reports for Nuclear Power Plants," Rev. 2.

APPENDIX A

STANDARD REVIEW PLAN 13.3

EMERGENCY PLANS FOR NUCLEAR POWER PLANTS

DISCUSSION

Regulatory concern for emergency planning is directed primarily at situations involving real or potential radiological hazards. Such hazards may place the health and safety of one or more persons in jeopardy. Emergency planning should aim to diminish the degree of jeopardy by preparing for timely action on the part of individuals who constitute a coordinated emergency organization. Although it is not practicable to develop a completely detailed response procedure for every conceivable type of emergency situation, advance planning can create a high order of preparedness and assure an orderly and timely decision-making process at times of stress as well as the availability of equipment, supplies, and essential services.

An important element of emergency planning for nuclear power plants is the recognition of a need to cope with a very broad spectrum of potential consequences. Federal, state, and local agencies as well as the applicant-licensee have responsible roles to play in both the planning and the implementation of emergency preparedness procedures. Federal interagency responsibilities for nuclear incident planning have been set forth in a Federal Register notice of January 24, 1973, by the former Office of Emergency Preparedness (now the Federal Disaster Assistance Administration). To a large extent, these responsibilities are directed toward a coordination of effort to provide assistance to state and local governments in their planning. This policy is based upon the recognition that state and local governments have the necessary authority to implement emergency measures in their jurisdictions. Although federal agencies can and will respond to emergencies arising from nuclear power plant activities if necessary, such response should be regarded primarily as backup and not a substitute for responsible action by licensees and state and local governments.

In the preparation of an emergency plan for a specific nuclear power plant, the applicant should be guided by the following criteria to clarify the scope, content, and purpose of the document which describes the plan. The emergency plan should incorporate sufficient detail so that other participating organizations and agencies with related plans may review it and determine that they are coordinated effectively with one another. Detail which can reasonably be expected to change from time to time, e.g., names and telephone numbers, equipment and supplies inventory lists, or step-by-step procedures or check lists which may be altered as a result of experience or test exercises, should not be incorporated in the plan. The document itself should not be considered as a primary working document to be used during an emergency. Implementing procedures documents, keyed to the plan, should be available for this purpose. The latter documents should not be necessary for licensing review. However, they should be available for inspection by the Commission's Directorate of Regulatory Operations, and should be transmitted, if applicable, to appropriate state or local agencies.

The plan document should also clarify its scope relative to interfacing plans and procedures within the operating organization, e.g., emergency and off-normal operating procedures within the plant, radiation protection program and procedures, and security plans.

Although a part of the final safety analysis report, it is recommended that the plan be prepared as a separate document.

BRANCH RECOMMENDATIONS

A. Each applicant's emergency plan should include provisions for handling emergencies both within the site of his plant and in the environs of the site. Responsibility for planning and implementing all emergency measures for persons within the site boundaries rests with the licensee. Planning and implementation of emergency measures in the environs of the site arising from onsite activities should be coordinated with local, county, state, and federal agencies having emergency responsibilities and should be described in the applicant's emergency plan. Such planning should generally be increasingly definitive in its provisions for emergency measures as the regions of consideration get closer to the site and the plant itself.

B. The scope and content of a nuclear power plant emergency plan should be substantially equivalent to that outlined in the following section, entitled "Organization and Content of a Nuclear Power Plant Emergency Plan".

ORGANIZATION AND CONTENT OF A NUCLEAR POWER PLANT EMERGENCY PLAN

CONTENTS

ORGANIZATION AND CONTENT OF A NUCLEAR POWER PLANT EMERGENCY PLAN

(In the following, the decimal notation identifies recommended major subject matter headings for the organization of an emergency plan document. The text, including portions identified by alphabetic notation, gives specific guidance or recommendations as to the content of the section or sub-section.)

For clarity, certain terms are employed with specific definitions as follows:

Definitions:

Assessment actions - means all of those actions taken after an accident has occurred which are collectively necessary to make decisions to implement specific emergency measures.

Corrective actions - means emergency measures taken to ameliorate or terminate an emergency situation at or near the source of the problem.

Protective actions - means those emergency measures taken after an accident or an uncontrolled release of radioactive materials has occurred, for the purpose of preventing or minimizing radiological exposures to persons which would be likely to occur if the actions were not taken.

Population-at-risk - means those persons for whom protective actions are or would be taken.

Affected persons - means persons who have been radiologically exposed or physically injured as a result of an accident, to a degree requiring special attention as individuals, e.g., decontamination, first aid, or medical services.

Recovery actions - means those actions taken post-emergency to restore property to its pre-emergency condition as nearly as possible.

Protective action guides - means projected radiological dose, or dose commitment, values to individuals in the general population which warrant protective action following a contaminating event.

Emergency action levels - means radiological dose rates, specific contamination levels of airborne, waterborne, or surface-deposited concentrations of radioactivity, or specific instrument readings, which may be used to prescribe specific emergency measures.

1.0 Scope and Applicability

This section of the plan should define the unit, plant, station, or area to which the plan is applicable, and a summary of its inter-relationships with (1) its implementing procedures, (2) plant operating, radiological control, and industrial security procedures, (3) other emergency plans of the company, e.g., an overall corporate plan, and (4) emergency plans of other participating agencies, particularly the responsible state agency.

2.0 Summary of Emergency Plan

This should describe the key elements of overall emergency planning logic incorporating graded emergency classifications of increasing severity and their relationship to the participating status of onsite and offsite personnel and agencies.

3.0 Emergency Conditions

3.1 Classification System

An emergency plan should characterize several classes of emergency situations. The system of classification employed should consist of mutually exclusive groupings (to avoid ambiguity) but should cover the entire spectrum of possible situations. Each class should incorporate (1) a specific emergency organization alerting and mobilization procedure, and (2) a set of predefined preliminary actions to be taken by designated emergency organization personnel. Succinct descriptive rather than numerical or alphabetical classification designations are recommended to give better immediate clues to personnel as to the scope and character of the situation.

An acceptable classification scheme is described below in qualitative terms. This part of the emergency plan should describe the criteria for recognizing and declaring each class, including specific emergency action levels for the last three classes.

a. Personnel Emergency

Accidents or occurrences onsite may require emergency treatment of individuals. This classification applies to situations which have no potential for escalation to more severe emergency conditions. There may be no effect on the plant, nor does it necessarily involve immediate operator action to alter plant status. A personnel emergency does not activate the entire plant emergency organization but may activate teams such as first aid. It may also require special local services such as ambulance and medical.

Implementing procedures for the handling of this class of emergency may also be incorporated in the plant's radiation protection procedures and general industrial safety procedures.

Included in this class are injuries which may be complicated by contamination problems or excessive radiation exposures to onsite personnel.

The recognition of this class of emergency is primarily a judgment matter for plant staff supervisory or management personnel. Its importance as part of the classification scheme rests to some extent on its "negative" information content, viz, that the incident giving rise to an emergency is restricted in its scope of involvement. This section of the plan should designate the classification criteria, and enumerate discrete accident situations which would give rise to the use of this class.

b. Emergency Alert

Specific situations may arise that can be recognized as creating a hazards potential that was previously non-existent or latent. In and of itself the situation has not yet caused damage to the plant nor harm to personnel and does not necessarily require an immediate change in plant operating status. Inherently, then, this is a situation in which time is available to take precautionary and constructive steps to prevent the realization of an accident and to mitigate the consequences should it occur. An emergency alert situation may be brought on by either man-made or natural phenomena.

Emergency alert conditions imply a rapid transition to a state of readiness by the plant personnel, the possible cessation of certain routine functions or activities within the plant which are not immediately essential, and possible precautionary actions which the specific situation may require. Examples of situations which might be placed in this class are: threats to or breaches of plant security measures such as bomb threats or civil disturbance; severe natural phenomena in the plant environment such as floods, earthquakes, tsunamis, hurricanes, or tornadoes; emergency situations such as fires at adjacent facilities; release of a toxic or noxious gas in or near the plant; or flooding offsite caused by malfunction or failure in some part of the plant cooling water system. This section of the emergency plan should identify specific candidate situations for emergency alerts and the quantitative criteria that would guide the decision to implement each. Qualitative criteria should be added for other candidate situations to guide the decision of on-site supervisory personnel.

c. Plant (Unit) Emergency

This class incorporates physical occurrences within the plant requiring full plant staff emergency organization response. The initial information and assessment indicates that it is very unlikely that an offsite hazard will be created. However, substantial modification of plant operating status is a highly probable corrective action if this has not already taken place by the actions of automatic protective systems. Although it is judged that the emergency situation can be corrected and controlled by the plant staff, notification of corporate headquarters staff to put them on an alert status is prudent. In turn, notification of appropriate offsite agencies as to the nature and extent of the incident is advisable. Evacuation of the plant is not anticipated in this class although protective evacuations or isolations of certain plant areas may be necessary.

Examples of situations which might fall into this class are those accidents which have been analyzed in the FSAR as events which are predicted to have no radiological consequences offsite. Fires, explosions or explosive gas releases, or in-plant flooding conditions, may also fall into this class.

Activation levels for declaring plant emergencies should be based upon the recognition of an immediate need to implement in-plant emergency measures to protect or provide aid to affected persons in the plant and to mitigate the consequences of damage to plant equipment, coupled with a positive observation that (a) effluent and other radiological monitors do not indicate the possibility of a site emergency, and (b) there is no apparent breach of any fuel cladding, primary system boundary, or containment. This section should describe the alarm conditions or combinations of alarm conditions and the emergency action levels for initiating a plant emergency and their bases.

d. Site (Station) Emergency

This class involves an uncontrolled release of radioactive materials into the air, water, or ground to an extent that initial information and assessment indicates that protective actions offsite may be desirable. Mobilization and readiness of offsite emergency organizations is prudent. Protective actions are likely to include evacuation of plant areas other than control rooms and emergency stations, and should include provisions for evacuation of construction personnel during those periods when additional units are under construction on the same site. Assessment actions will include monitoring of the environment.

Situations which are likely to fall into this class include those accidents analyzed in the FSAR which are predicted to have small to moderate releases at the exclusion radius. It should be anticipated that site emergencies would not normally be preceded by a plant emergency although this evolution should not be excluded.

Emergency action levels declaring a site emergency should be defined in terms of instrument readings or alarms in the control room. To avoid false alarms or to minimize their frequency of occurrence, the levels may be defined so as to require corroborating evidence from two independent sources having input to the control room. Indications from effluent monitors should be included. Site emergencies should also be declared on the basis of evidence of apparent breaches in fuel cladding, primary system boundaries, or containment when otherwise a plant emergency would be declared. The bases and criteria used to define the instrument alarm levels should be described. Suitable criteria would be protective action guide values at a security fence, or exclusion area or site boundary and the bases would show how the effluent monitor readings relate to such values. Protective action guides selected for this purpose should be below the siting guideline values of 10 CFR Part 100 and should have the concurrence of state authorities. Federal agency guidance is available to assist in the selection of acceptable protective action guides.

e. General Emergency

This is an occurrence characterized by offsite consequences requiring protective action measures as a matter of prudence or necessity. Evacuation of the site may also be necessary under extreme circumstances. Emergency action levels for declaring a general emergency should be defined.

Two categories, short term and long term, should be recognized. The former is guided by direct radiation or inhalation hazards, while the latter is guided primarily by contamination hazards. General emergency action levels may be based upon confirmatory measurements taken in the field to the extent that it can be shown that they can be taken and evaluated rapidly enough to permit adequate time for the protective actions to be accomplished. The levels for severe short term situations require definition in terms of effluent and other onsite monitor indications. As in the previous case, the bases and criteria used to define the relevant instrument levels should be described.

3.2 Spectrum of Postulated Accidents

Accident analysis sections of safety analysis reports are primarily concerned with the design responses of a plant to postulated malfunctions or equipment failure and include estimates of the radiological consequences of discrete accidents. By contrast, emergency planning is concerned with individual and organizational responses to the continuum of potential accident situations which must include those discrete accidents which have been hypothesized. This section of the emergency plan should show that each is encompassed within the emergency characterization classes and provide a summary analysis of their implications for emergency planning. Implications to be considered include:

a. Instrumentation capability for prompt detection and continued assessment, including functional applicability, range, response time, locations of sensing and readout elements (including alarms), and backup or redundant capability.

b. Manpower requirements for assessment, including record keeping; for corrective actions; for protective actions including communications requirements; and for aid to affected persons.

c. The timing of and the time required for the implementation of each emergency measure which may be brought into play.

4.0 Organizational Control of Emergencies

Starting with the normal operating organization as a base, this section of the plan should describe the emergency organization that would be activated on the site and its augmentation and extension offsite. Authorities and responsibilities of key individuals and groups should be delineated. The communication links established for notifying, alerting, and mobilizing emergency personnel should be identified.

4.1 Normal Operating Organization

Both day and night shift operating staffs (crews) should be described, indicating clearly who is in the immediate onsite position of responsibility for the plant and station (normally a shift supervisor) and his authority and responsibility for declaring an emergency.

4.2 Onsite Emergency Organization

This section should describe the mobilization billets of plant staff personnel for controlling each class of emergency for both day and night shift situations.

4.2.1 Direction/Coordination

The position title of that person who is designated to take charge of emergency control measures onsite should be clearly identified. A specific line of succession for this function should also be given. A policy statement describing the scope of authority and responsibility vested in that role by the company (applicant) should be included. Functional responsibilities assigned to this individual should be described, and should include a summary of those preliminary assessment procedures that would be followed to prescribe or guide his decision to classify and declare an emergency.

4.2.2 Plant Staff Emergency Assignments

The plan should specify the functional areas of emergency activity to which members of the plant staff are assigned, including an indication of how the assignments are made for both day and night shifts, and for plant staff members both onsite and away from the site. Functional areas should include:

1. Plant systems operations
2. Radiological survey and monitoring
3. Fire fighting
4. Rescue operations
5. First aid
6. Decontamination
7. Security of plant and access control
8. Repair and damage control
9. Personnel accountability
10. Record keeping
11. Communications

4.3 Augmentation of Onsite Emergency Organization

This section should describe two categories of offsite supporting assistance to the plant staff emergency organization. These can be either directed, authorized, or requested by the company management to perform special emergency assistance functions.

4.3.1 Headquarters Support

Headquarters management, administrative, and technical personnel should be prepared to augment the plant staff, both in emergency planning and in the performance of certain functions required to cope with an emergency. The following special functions are considered appropriate for headquarters support and should be incorporated in the overall plan, although company policy and organizational features may dictate variations in modes of assigning responsibilities for these functions among headquarters personnel, plant staff personnel, and outside support organizations.

1. Environs monitoring.
2. Logistics support for emergency personnel, e.g., transportation, temporary quarters, food and water, sanitary facilities in the field, and special equipment and supplies procurement.
3. Technical support for planning reentry/recovery operations.
4. Notification of governmental authorities.
5. Public relations and information release, coordinated with governmental authorities, including steps taken to inform visitors to the plant or information center, and to occupants in the environs of the site, of how the emergency plans provide for notification to them and how they can expect to be advised as to what to do.

The emergency organization status of supporting headquarters personnel should be specified, relative particularly to the person directing the plant emergency organization.

In some instances, companies may provide for certain emergency supporting services to their plants by contract with private organizations. Where this is the case, the nature and scope of the support services should be characterized here. (The Commission may find it necessary to request evidence of the qualifications of such contractors.) Specific services by the contractors should be identified as such at the appropriate places in the emergency plan document.

4.3.2 Local Services Support

This section should identify the extension of the organizational capability for handling emergencies to be provided by ambulance, medical, hospital, fire, and police organizations. Evidence of the arrangements and agreements reached with such organizations should be included in an appendix and referenced here, along with references to the parts of the plan in which their functions are primarily described.

4.4 Coordination with Participating Agencies

This section should identify the principal state agency (designated state authority) and other governmental (local, county, state, and federal) agencies having planning and action responsibilities for emergencies, particularly for radiological emergencies, in the area in which the plant is located. If the boundary line between two political entities, e.g., counties or states, passes within the low population zone or approximately four miles of the site, agencies from both entities should be included. Subsections for each such agency should describe the following:

a. Identity of agency.

b. Summary of written agreement with agency which clearly defines the authority and responsibility of the agency for emergency preparedness planning, and for emergency response in the public domain, particularly relative to those of the licensee and to those of other agencies. (Copies of such agreements should be included in an appendix, along with a copy or summary of relevant parts of that agency's emergency plan.)

c. Activation of agency function, including titles and alternates of both ends of the communications links, and primary and alternate means of communication.

d. The designation and location of the emergency operations center of each agency.

e. Support of the agency function that may be provided by the company emergency organization, which may include (1) information on plant status, monitoring results, dose predictions, (2) recommendations or requests for specific actions, and (3) logistics support.

Typical agencies to be included here are: law enforcement agencies (not included above, e.g., state police/highway patrol), departments of health and environmental protection, civil defense and emergency/disaster control agencies, AEC regional operations offices, and the AEC regional office of Regulatory Operations.

5.0 Emergency Measures

Specific emergency measures should be identified in this section and related to action levels or criteria that specify when the measures are to be implemented. They should be organized with respect to each emergency classification. Preplanned action levels and criteria should be designed to assist and guide, or in some cases specify, the decision-making functions.

The planning represented by this section should lead to more detailed emergency procedures and assignments for executing tasks by appropriate members of the total emergency organization. Emergency measures begin with the activation of an emergency class and its associated emergency organization. The additional measures may be organized into assessment actions, corrective actions, protective actions, and aid to affected persons.

5.1 Activation of Emergency Organization

The emergency conditions classified in Section 3.1 involve the alerting or activation of progressively larger segments of the total emergency organization. This section should describe how the necessary communications steps are taken to alert or activate emergency personnel under each class, including, in particular, action levels for notification of offsite agencies.

5.2 Assessment Actions

Effective coordination and direction of all elements of the emergency organization require continuing assessment throughout the duration of an emergency situation. Assessment functions should be incorporated in explicit procedures for each emergency classification. They should be identified in this section and may include the following:

a. Surveillance of control room instruments and emergency control center monitors, radiological and meteorological, installed, pursuant to General Design Criteria 13 and 64 of 10 CFR Part 50, Appendix A.

b. Surveillance of containment integrity.

c. In-plant radiological surveys.

d. Site and site boundary surveys.

e. Environs surveys and monitoring.

   1. Plume and other effluent surveillance for short term assessment. Planning should consider type of data sought; instrument and equipment requirements; monitoring team transportation facilities, e.g., aircraft, boats, vehicles; methods and accuracy of plume location; and potential use of fixed off-site monitoring facilities.

   2. Contamination surveillance. Planning should consider the timing, frequency, and types of samples to be collected, such as soil, vegetation, food, milk and water supplies, and potential locations for reconcentration, e.g., in air intake filters.

f. Data reporting, reduction and analysis.

g. Interviewing evacuees or other witnesses of the accident.

h. Notification of assessment results for modification of emergency measures in progress, if necessary.

5.3 Corrective Actions

Many emergency situations involve actions which can be taken to correct or mitigate the situation at or near the source of the problem. This section should identify those actions, such as fire control, and repair and damage control, which would be implemented when necessary. Emergency exposure criteria for personnel undertaking corrective actions should be included.

5.4 Protective Actions

This section should describe the nature of protective actions which the plan contemplates, the protective action levels, the area involved, and the means of notification to the population-at-risk. Protective actions to be taken offsite by other agencies should be described.

5.4.1 Protective Cover, Evacuation, Personnel Accountability

The emergency plan should provide for timely relocation of persons to prevent or minimize exposure to direct radiation or airborne hazards. The following items should be included:

1. Plant Site

   a. Action criteria.
   b. The means and the time required to notify persons involved. These should include:
      (1) Employees not having emergency assignments.
      (2) Working and non-working visitors.
      (3) Contractor and construction personnel.
   c. Control of public access areas on or passing through site or within exclusion area.
   d. Evacuation routes, transportation of personnel, and reassembly areas, including inclement weather and high traffic density alternatives.
   e. Missing persons check.
   f. Radiological monitoring of evacuees.

2. Off-Site Areas

   a. Action criteria including inclement weather alternatives.
   b. Company emergency organization responsibilities.
   c. Agency responsibilities.
   d. The means and the time required to notify and the expected response of persons involved. These should include:
      (1) Adjacent businesses, property owners, and tenants.
      (2) Nearby schools or recreational facilities.
      (3) General public, in the environs.

5.4.2 Use of Protective Equipment and Supplies

Additional protective actions which should be considered in emergency planning include measures for minimizing the effects of radiological exposures or contamination problems through the distribution of special equipment or supplies. Measures to be considered include:

1. Individual respiratory protection.
2. Use of protective clothing.
3. Individual thyroid protection.

For each measure which might be used, a description should be given of:

1. Criteria for issuance.
2. Location(s) of items.
3. Means of distribution to onsite and offsite persons.

5.4.3 Contamination Control Measures

Provisions should be made for preventing or minimizing ingestion of or exposure to contaminated areas or materials. (Control of in-plant contamination should be described in the facility radiological protection procedures and need not be repeated here.) Measures for the protection of onsite persons outside of fenced security areas and offsite persons should include:

1. Isolation or quarantine and area access control.
2. Control of the distribution of affected commercial agricultural products.
3. Control of public water supplies.
4. Means for providing advisory information regarding the use of potentially affected home food and water supplies.
5. Criteria for permitting return to normal use.

Action levels and responsibility for execution of each measure contemplated should be described.

5.5 Aid to Affected Personnel

This section of the emergency plan should describe measures which will be used to provide necessary assistance to persons injured or radiologically exposed. The following matters should be included:

5.5.1 Emergency Personnel Exposure Criteria

Exposure limits should be specified for voluntary entry or reentry of areas to remove injured persons and limits for emergency personnel who may provide first aid, decontamination, ambulance, or medical treatment services to injured persons.

5.5.2 Decontamination and First Aid

Capabilities for decontaminating personnel for their own protection and to prevent or minimize further spread of contamination should be included, along with a brief description of first aid capabilities of appropriate members of the emergency organization.

5.5.3 Medical Transportation

Arrangements for transporting injured personnel, who may also be radiologically contaminated, to medical treatment facilities should be specified.

5.5.4 Medical Treatment

Arrangements made for local and back-up hospital and medical services, and the capability for radiation exposure and uptake evaluations should be described.

For both hospital and medical services, the plan should incorporate assurance that the required services are not only available, but also that persons providing them are prepared and qualified to handle radiological emergencies. Written agreements with respect to arrangements made by the applicant, which should be included in the appendix, would facilitate this determination.

6.0 Emergency Facilities

This section of the emergency plan should identify, describe briefly, and give the locations of the following categories of items.

6.1 Emergency Control Centers

This should include the principal and, if provided for, alternate onsite location from which effective emergency control direction is given. One alternate offsite location under the jurisdiction of the applicant should also be described. Their descriptions should also specify prevailing wind direction and evacuation routes.

6.2 Communications Systems

Brief descriptions should be given of both internal and external communications systems that would perform vital functions in transmitting and receiving information throughout the course of an emergency.

6.3 Assessment Facilities

Many of the emergency measures described in Section 5.0 will depend upon the availability of monitoring instruments and laboratory facilities. This section should list monitoring systems that are to be used to initiate emergency measures as well as those used for continuing assessment. Organization of the listing should be as follows.

6.3.1 Onsite Systems and Equipment

1. Natural phenomena monitors, e.g., meteorological, hydrologic, seismic.
2. Radiological monitors, e.g., process, area, emergency, effluent, portable monitors and sampling equipment.
3. Non-radiological monitors, e.g., reactor coolant system pressure, temperatures, containment pressure, temperature, liquid levels, flow rates, status or lineup of equipment components.
4. Fire detection devices.

6.3.2 Environs Monitoring Facilities and Equipment

1. Natural phenomena monitors.
2. Radiological monitors.
3. Laboratory facilities, fixed and mobile.

Reference may be made to the applicable part of the safety analysis report for more detailed descriptions, if applicable.

6.4 Protective Facilities

Specific facilities mentioned in Section 5.4.1 which are intended to serve a protective function should be described, emphasizing those features of the facility which assure its adequacy with respect to capacity for accommodating the number of persons expected, and with respect to shielding, ventilation, and inventory of supplies. Such facilities might include fallout shelters or similar areas, and reassembly points. If design details have been provided elsewhere in the safety analysis report, a brief summary only need be given here, along with a reference to the detail.

6.5 First Aid and Medical Facilities

A summary description of onsite facilities should be provided. Offsite medical facilities should be described in the appendix, along with the agreements providing for their use.

7.0 Maintaining Emergency Preparedness

This section of the plan should describe the means to be employed to assure that the plan continues to be effective throughout the lifetime of the nuclear facility.

7.1 Organizational Preparedness

7.1.1 Training

This section should include a description of periodic training programs to be given to all categories of emergency personnel. Specialized training for the following categories should be included:

1. Directors or coordinators of the plant emergency organization.
2. Personnel responsible for accident assessment, including control room shift personnel.
3. Radiological monitoring teams.
4. Fire, and repair and damage control teams.
5. First aid and rescue team members.
6. Local services personnel.
7. Medical support personnel.

7.1.2 Drills

Periodic (at least annual) announced drills should be incorporated in the emergency plan. These should be pre-planned simulations of accidents to test the adequacy of timing and content of specific implementing procedures and to test emergency equipment. Arrangements should be made for critiques of the drills. Coordinating drills should be made with participating agencies at least annually, testing at a minimum the communications links. An initial coordinated drill with participating agencies should be planned and carried out prior to fuel loading of the first unit at any site.

7.2 Review and Updating of the Plan and Procedures

Provision should be made for an annual review of the emergency plan and for updating and improving procedures based upon training, drills, and changes onsite or in the environs. Means for maintaining all coordinate elements of the total emergency organization informed of revisions to the plan or relevant procedures should be described.

7.3 Emergency Equipment and Supplies

The operational readiness of all items of emergency equipment and supplies should be assured. The plans and schedules for performing maintenance, surveillance testing, and inventory of emergency equipment and supplies should be described.

8.0 Recovery

This section should describe general plans, including applicable criteria, for restoring property as nearly as may be possible to its pre-emergency status.

9.0 Appendix

The appendix should include the following items:

1. Copies of agency agreement letters and copies or summaries of interfacing emergency plans.

2. Plots of calculated time-distance-dose for the most serious design basis accident as required by Revision 2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants".

3. Listings by title of written procedures which implement the plan.

4. Listings by category of protective equipment and supplies.

The written procedures themselves and detailed cataloguing of protective equipment and supplies should be available at the plant site for inspection at any time by a representative of the Commission's Directorate of Regulatory Operations.

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SECTION 13.4 REVIEW AND AUDIT

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

QAB reviews and evaluates the applicant's plan for conducting reviews and audits of operating phase activities that are important to safety, as described in the applicant's preliminary safety analysis report (PSAR) or final safety analysis report (FSAR). The primary focus of attention should be on the procedures that will be used to implement the licensee's responsibility pursuant to 10 CFR §50.59 relating to proposed changes, tests, and experiments, and on the procedures for after-the-fact review evaluation of unplanned events, such as abnormal occurrences. At the PSAR review stage, the applicants commitment to follow the recommendations of Regulatory Guide 1.33 is examined. At the FSAR stage, the applicant's proposed implementation plan for conducting reviews and audits is evaluated. Procedures for both onsite offsite (independent) review should be described, as follows:

1. Qualified members of the onsite operating organization and persons independent of the onsite organization should participate in the review of safety-related operating activities. The PSAR should provide a clear commitment for the applicant to meet Section 4 of ANSI N18.7-1972.

2. Qualified members of the onsite operating organization are expected to participate in the review of operating activities to assist the plant manager either as part of their individual job responsibilities or as members of a functional review group. The FSAR should describe how the onsite organization functions with respect to review of proposed changes to systems or procedures, tests, and experiments, and of unplanned events that have operational safety significance.

3. The FSAR should provide a detailed description of the procedure and organization employed to examine safety-related operating activities independent of the operating organization. The information should be sufficient to describe how and when such a program is to be implemented, relative to fuel loading of the first unit.

II. ACCEPTANCE CRITERIA

The staff position applicable to this section of the FSAR is the set of requirements and recommendations found in ANSI N18.7-1972 at Sections 4.1-4.5, as endorsed by Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

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 review of this section of the PSAR consists of determining that the applicant has committed to meeting the acceptance criteria of II, above.

The review of this section of the FSAR consists of an analysis of the information submitted by detailed comparison with the acceptance criteria of II, above. When the reviewer has determined that the requirements and recommendations of the referenced sections of the standard have been implemented in the applicant's plans for conducting reviews and audits, the review of this section of the SAR is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support conclusions of the following type, to be used in the staff's safety evaluation report:

> "The applicant's program for review and audit of plant operations is in conformance with staff positions described in Regulatory Guide 1.33 and applicable industry standards (ANSI N18.7-1972), and is acceptable. The major features of the review and audit program will be incorporated in the administrative controls section of the plant technical specifications."

The evaluation finding for this section should also include the following:

1. A brief description of the two-level review process.
2. A statement of the applicant's commitment to perform independent reviews and audits in accordance with a written charter.

V. REFERENCES

1. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

2. ANSI N18.7-1972, "Standard for Administrative Controls for Nuclear Power Plants," American National Standards Institute (1972).

3. 10 CFR §50.59, "Authorization of Changes, Tests, and Experiments."

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SECTION 13.5 PLANT PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Operator Licensing Branch (OLB)

Secondary - None

I. AREAS OF REVIEW

OLB reviews the plant procedures, as described in the applicant's safety analysis report (SAR). This section of the SAR should describe administrative and operating procedures that will be used by the operating organization (plant staff) to assure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, it is not expected that detailed written procedures will be included in the SAR. The preliminary safety analysis report (PSAR) should describe preliminary schedules for their preparation and the final safety analysis report (FSAR) should provide descriptions as to the nature and content of procedures as detailed below.

1. The SAR section on administrative procedures (13.5.1) should include a commitment to conduct all safety-related operations by detailed written and approved procedures, and should provide for the preparation of written administrative procedures which will govern the safety-related activities of the plant staff. The FSAR should identify the persons (by title and organization) having responsibility for the writing of procedures, and the persons who must review and approve them before they are implemented. In the FSAR, a description of administrative procedures should be provided which should include the following:

   a. Standing orders to operations shift supervisors and shift crews including:
      (1) The reactor operator's authority and responsibilities.
      (2) The senior operator's authority and responsibilities.
      (3) The responsibility to meet the requirements of 10 CFR §50.54(i), (j), (k), (l), and (m), including a diagram of the control area that indicates the area designated "at the controls."

b. Special orders of a transient or self-cancelling character.
c. Equipment control procedures.
d. Control of maintenance and modifications.
e. Master surveillance testing schedule.
f. Temporary procedures.

2. A section on operating and maintenance procedures should be included in the FSAR (13.5.2).

a. The first part should deal with procedures which are performed by licensed operators in the control room. Each such operating procedure should be identified by title and included in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

(1) System procedures.
(2) General plant procedures.
(3) Off-normal operating procedures.
(4) Emergency procedures.
(5) Alarm response procedures.
(6) Temporary procedures.

In category (5), individual alarm response procedures should not be listed. However, the system employed to classify or subclassify alarm responses, and the methods to be employed by operators to retrieve or refer to alarm response procedures should be described. Immediate action procedures required to be memorized should be identified.

b. The second part of this section should describe how other operating and maintenance procedures are classified, what group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass. The following categories should be included. If their general objectives and characters are described elsewhere in the FSAR, or application, these may be described by specific reference thereto.

(1) Plant radiation protection procedures.
(2) Emergency preparedness procedures.
(3) Instrument calibration test procedures.
(4) Chemical and radiochemical control procedures.
(5) Radioactive waste management procedures.
(6) Maintenance and modification procedures.
(7) Materials control procedures.
(8) Plant security procedures.

II. ACCEPTANCE CRITERIA

This section of the SAR should demonstrate the applicant's commitment to the conduct of operations by means of written and approved procedures. It constitues additional evidence of his technical qualifications, and forms a basis for a key part of the regulatory inspection program.

A generally acceptable target date for completion of administrative procedures and operating procedures is about six months before fuel loading, inasmuch as familiarization with these procedures is an essential part of the staff training program, including preparation for operator license examinations prior to criticality.

This section of the FSAR should comply with the guidance contained in Section 5, ANSI N18.7-1972, and Regulatory Guide 1.33, or present acceptable alternatives. In addition, this section of the FSAR should indicate the plans for meeting the requirements of §50.54(i), (j), (k), (l), and (m) of 10 CFR Part 50.

III. REVIEW PROCEDURES

The review of this section consists of a detailed conparison of the information submitted with the acceptance criteria of II above, as applicable to the PSAR or FSAR. When the reviewer has determined that each of these criteria has been satisfied, based upon the statements made by the applicant in the SAR, the review of this section is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and his review support the following type of conclusion, to be used in the staff's safety evaluation report:

> "The applicant's provisions for administrative procedures and operating procedures conform to Regulatory Guide 1.33 and applicable industry standards and are acceptable.
>
> "The applicant's provisions meet the requirements of §50.54(i), (j), (k), (l), and (m) of 10 CFR Part 50.
>
> "The significant provisions will be included in the administration controls section of the plant's technical specifications."

The evaluation findings for this section should also include the following:

1. A statement that plant operations are to be performed in accordance with written and approved procedures.

2. A brief description of the categories of procedures to be included.

3. A brief description of the review and approval mechanism for procedures and changes, thereto.

V. <u>REFERENCES</u>

1. ANSI N18.7-1972, "Standard for Administrative Controls for Nuclear Power Plants," American National Standards Institute (1972).

2. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

3. 10 CFR Part 50, §50.54, "Conditions of Licenses."

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

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Section 13.6 INDUSTRIAL SECURITY

REVIEW RESPONSIBILITIES

Primary - Industrial Security and Emergency Planning Branch (ISEPB)

Secondary - None

I. AREAS OF REVIEW

At the preliminary safety analysis report (PSAR) stage, the review of this section covers plans for implementing security measures relating to (1) the screening of personnel employed to work at the proposed plant and (2) the layout of the plant and other design features and equipment arrangements intended to provide protection of vital equipment against acts of industrial sabotage.

At the final safety analysis report (FSAR) stage, the review involves the evaluation of the industrial security plan, which describes a comprehensive physical security program for the plant site. The review encompasses the physical security organization, access controls to the plant including physical barriers and means of detecting unauthorized intrusions, provisions for monitoring the status of vital equipment, selection and training of personnel for security purposes, communications systems for security, and arrangements with law enforcement authorities for assistance in responding to security threats. The implementation schedule for the physical security program is reviewed, including phases for multi-unit plants where applicable.

Specific information to be reviewed, referenced to applicable sections of ANSI N18.17-1973, include the following:

1. Clear diagrams, to approximate scale, displaying the following:
   a. Designated security areas of the plant site, including physical barriers.
   b. The locations of alarm stations.
   c. The locations of access control points to protected and to vital areas.
   d. The location of parking lots relative to the clear areas adjacent to the physical barriers surrounding protected areas.
   e. Special features of the terrain which may present special vulnerability problems.

f. The location of relevant law enforcement agencies and their geographical jurisdictions.

2. If the policy of the owner organization permits use of any part of the owner-controlled area by members of the general public, details of how the requirements of Section 3.2 will be met.

3. The response capabilities of local law enforcement agencies (Section 4.4.7), including estimates of the number of officers that can arrive at the plant site in the event of a security threat, within five to fifteen minutes, fifteen to thirty minutes, and thirty minutes to one hour, after receipt of a call for assistance. (This response capability bears upon the adequacy of the size of the onsite guard force.)

II. ACCEPTANCE CRITERIA

At the PSAR stage, preliminary planning for industrial security should show how conformance to the applicable provisions of Regulatory Guide 1.17 are expected to be achieved, including:

1. ANSI N18.17-1973, Section 2, "Definitions;" Section 3, "Designated Security Areas;" Section 4.3, "Employee Screening"; and Section 5, "Plant Design."

2. Regulatory Guide 1.17, Revision 1, Section C.1.b, "Security Alarms," and Section C.3, "Protection of Vital Equipment."

This planning should include a commitment to design phase review for physical security and should show how this responsibility is to be implemented by the applicant.

At the FSAR stage, the applicant's security plan must conform to the requirements of 10 CFR 50.34(c), and to applicable requirements of 10 CFR Part 73. In addition, the provisions of Regulatory Guide 1.17, Revision 1, including the requirements and recommendations of ANSI N18.17-1973, Sections 3 and 4, establish the basis for an adequate security plan for the protection of nuclear power plants against industrial sabotage.

Specific acceptance criteria, including staff interpretations of some of the more general requirements of the ANSI Standard, are as follows: (Section references are to sections of ANSI N18.17-1973.)

1. Surveillance of a protected area (Section 3.3.3) should be by a system which can provide for continuous monitoring of the entire perimeter of a protected area so as to allow response to be initiated at the time of penetration of a protected area.

2. Central alarm stations should be regarded as vital areas and meet the qualifications required thereof.

3. For each plant, the onsite security force should include not less than two guards on each shift.

4. "Armed guards" means guards physically carrying firearms. Persons assigned to control access points to protected areas should not be armed if their work post is exterior to the protected area.

5. If search procedures of individuals and packages they may be carrying are not stipulated for all persons and hand-carried packages entering the protected area, then selection of individuals and hand-carried packages for search should be on the basis of a random process which is exercised each time an individual is about to enter the protected area.

6. Essential vehicles (Section 3.3.1) allowed access to protected areas include those designated strictly for security or emergency purposes, or vehicles not used primarily for conveyance of people that must be allowed within the protected area to serve a required function.

7. Picture badge identification should be used to satisfy Sections 3.3.2.1 and 3.3.2.2, with special color coding or symbols to satisfy 3.4.1, when inside vital areas.

8. Casual visitor groups, such as tour groups do not constitute "persons having a need to enter such (vital) areas", Section 3.4.1.

Implementation of the physical security program should be accomplished one to two months before fuel loading. Security features required for new fuel in storage prior to loading of the first unit should be implemented as of the time fuel is onsite.

III. REVIEW PROCEDURES

At the PSAR stage, the review consists of a careful examination of the information submitted and comparison with the acceptance criteria set forth in II above. The general plant description in Chapter 1 and site-related information in Chapter 2 of the PSAR should be examined to determine if there are unique features that should be considered in establishing the physical protection program. It may be desirable at this stage to discuss the formulation of this program with the applicant.

At the FSAR stage, the physical security plan is reviewed to determine its conformance with the regulations, the information requirements of I above, and the acceptance criteria of II above. Applicable regulations, the position statements in Regulatory Guide 1.17, and the requirements and recommendations of ANSI N18.17-1973 are used as check lists for this review. The reviewer may also use appropriate Division 5 Regulatory Guides to the extent they are applicable to physical protection programs at nuclear power plants. Those having potential applicability are listed in the references. It is particularly important that the reviewer assure himself that all items of vital equipment are contained within vital areas. A site visit by the reviewer may be necessary, during the construction phase, before the evaluation of the plan can be completed.

IV. EVALUATION FINDINGS

The evaluation finding at the PSAR stage should be substantially equivalent to the following statement:

"The applicant has provided a general description of plans for protecting the plant against potential acts of industrial sabotage. Provisions for the screening of employees at the plant, and for design phase review of plant layout and protection of vital equipment have been described and conform to Regulatory Guide 1.17. We conclude that the applicant's arrangements for protection of the plant against acts of industrial sabotage are satisfactory for this stage of the licensing process."

The evaluation finding at the FSAR stage should be substantially equivalent to the following statement:

"The applicant has submitted a comprehensive physical security plan for the protection of the plant against potential acts of industrial sabotage. This plan has been withheld from public disclosure pursuant to 10 CFR 2.790(d).

"This plan has been reviewed and found to contain features considered essential for such a program by the staff. In particular, it has been found to comply with the Commission's regulations including 10 CFR 50.34(c) and applicable sections of 10 CFR Part 73, and conforms to the positions set forth in Regulatory Guide 1.17."

V. REFERENCES

1. Regulatory Guide 1.17, Revision 1, "Protection of Nuclear Power Plants Against Industrial Sabotage."

2. ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants," American National Standards Institute (1973).

3. Regulatory Guide 5.7, "Control of Personnel Access to Protected Areas, Vital Areas, and Material Access Areas."

4. Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials."

5. Regulatory Guide 5.20, "Training, Equipping, and Qualifying of Guards and Watchmen."

6. 10 CFR 50.34(c), "Physical Security Plan."

7. 10 CFR Part 73, "Physical Protection of Plants and Materials."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 14.1 INITIAL PLANT TEST PROGRAMS - PSAR

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

The QAB reviews the following areas, relating to initial plant test programs, described in Chapter 14 of the preliminary safety analysis report (PSAR) submitted by the applicant as part of his construction permit (CP) application:

1. Scope of Test Program

The initial plant test program is normally divided into four major phases: preoperational tests; initial fuel loading and precritical tests; low power tests; and power ascension tests. The following descriptive information provided for each phase is reviewed: (1) the definition of each phase; (2) the general testing objectives for each phase and the general prerequisites to be completed before each phase is begun; (3) the extent to which the test program will be used to verify the adequacy of construction and design of both the nuclear portion of the facility and the balance-of-plant; (4) the organizations, including those of the applicant, nuclear steam system (NSS) supplier, and architect-engineer, that will participate in the development and execution of the test program along with the general responsibilities of these organizations; (5) the applicant's planned involvement in the development and approval of test procedures, conduct of the tests, and review and approval of test results; and (6) the extent that the applicant will use his NSS supplier, architect-engineer, his own engineering units, or other system designers to provide scoping documents containing testing objectives and acceptance criteria for use in developing detailed test procedures.

2. Plant Design Features That are Special, Unique, or First-of-a-Kind

The applicant's description of the preoperational and startup tests planned for principal plant design features that are special, unique, or first-of-a-kind is reviewed. The applicant's plans relative to special prototype or in-plant functional testing requirements for such features are reviewed to establish the extent that such test requirements will verify design performance objectives.

3. Regulatory Guides and Industry Standards

The applicant's plans for utilizing applicable regulatory guides and industry standards in the development of his test program are reviewed. The extent to which the applicant intends to conform to the general recommendations of Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," in the development of his test program is reviewed. Exception to Regulatory Guide 1.68 are reviewed on a case-by-case basis, considering the justification provided.

4. Utilization of Plant Operating and Testing Experience at Other Reactor Facilities
   The applicant's plans relative to the review and evaluation of plant operating experiences at other reactor facilities and his plans appropriately to factor the results of this study into his test program are reviewed. The applicant's schedule for conducting such studies, the organizational units that will conduct the studies, and the scope of the study are reviewed.

5. Test Program Schedule
   The applicant's schedule for developing and conducting the different phases of the test program is reviewed and is compared with the expected fuel loading date to determine if adequate time has been allotted to develop and conduct a comprehensive test program. The schedule is examined to determine the availability of the plant operating and technical staff to participate in the test program and to determine if interference exists with the schedules for hiring or training of the plant operating and technical staff. The schedule is also examined to establish the availability of plant operating and emergency procedures that will be use-tested during the test program.

6. Trial Use of Plant Operating and Emergency Procedures
   The applicant's plans for trial use of plant operating and emergency procedures during the test program is reviewed to determine the extent such procedures will be trial tested.

7. Augmenting of the Applicant's Staff During the Test Program
   The descriptive information pertaining to the applicant's plans for augmenting his staff during the development and conduct of each phase of the test program is reviewed considering the prior nuclear experience of the applicant's staff and the degree of planned augmentation. The schedule for providing augmenting personnel is also reviewed.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described above are as follows:

1. Scope of Test Program
   The primary objectives of the review of information on the scope of test programs in the PSAR are to assure that the applicant recognizes the need to develop and conduct comprehensive test programs and that the applicant has performed the necessary early planning for successful achievement of these goals. Particular criteria that should be satisfied are as follows:

a. The description of the test program should identify and define the major phases of preoperational testing, fuel loading and precritical testing, low power testing, and power ascension testing. The definitions provided should be consistent with those included in Regulatory Guide 1.68 or justification provided for exceptions.

b. The applicant's stated test program objectives should provide assurance that the adequacy of construction, as well as design, will be verified for safety-related structures, systems, and components.

c. The applicant should designate the responsible organizations that will participate in the test program and there should be reasonable assurance that such designated organizations can collectively provide the necessary skills and experience to develop and conduct the test program.

d. The applicant should plan to utilize the plant operating and technical staff in the development and conduct of the test program and in the review of test results.

e. The applicant should establish plans for the designers of structures, systems, and components, which may include the NSS supplier, architect-engineer, or other design groups, to provide test objectives and acceptance criteria to be used in the development of detailed test procedures.

f. The applicant's program should provide assurance that construction of structures, systems and components will be essentially complete prior to beginning preoperational testing.

g. The applicant's program should provide assurance that construction tests and inspections will be essentially completed before preoperational testing is begun. Such testing or inspection may include system flushing and cleaning, wiring checks, leak tightness tests, initial calibration of instrumentation, and subsystem and component functional tests.

h. The applicant should describe the administrative controls to assure that preoperational test prerequisites will be satisfied.

2. Plant Design Features That Are Special, Unique, or First-of-a-Kind

a. The applicant should identify all principal safety-related plant design features or systems that are special, unique, or first-of-a-kind.

b. The applicant's description of planned tests should establish that such design features and systems will be functionally tested to demonstrate performance requirements.

3. Regulatory Guides and Industry Standards

The applicant should commit to following the guidelines of Regulatory Guide 1.68 in developing his test program or should justify any exceptions. Exceptions to these guidelines will require evaluation on a case-by-case basis. The applicant should also commit to reviewing other regulatory guides at the time the detailed test procedures are being developed to establish which guides have applicability to his test program.

4. Utilization of Plant Operating and Testing Experiences at Other Reactor Facilities

The applicant should plan to conduct a study of reactor operating experiences and to factor the results into his test programs as appropriate. The schedule for conducting the study should be consistent with the schedule for development of the detailed test procedures.

5. Test Program Schedule

   a. The applicant should provide a schedule, relative to the expected fuel loading date, for the beginning and end of each major phase of the test program. The schedule for conducting preoperational testing should provide for at least nine months of actual testing.
   b. The preoperational testing phase should be scheduled such that most of the plant operating and technical staff can participate in the development, conduct, and review of test results. The hiring and training schedules for these personnel, as presented in Chapter 13 of the PSAR, must be compatible with the preoperational testing schedule. In general, the schedule for hiring and training of such personnel should be completed at least one year prior to the expected fuel loading date.
   c. The schedule for development of plant operating and emergency procedures should establish that they will be available for trial use during the test program.

6. Trial Use of Plant Operating and Emergency Procedures

The applicant's plans and commitments in this area should establish that plant operating and emergency procedures will be tested, to the extent practical, during the test program.

7. Augmenting of the Applicant's Staff During the Test Program

   a. The applicant should plan to augment the plant staff throughout the entire test program.
   b. The organizations designated by the applicant to be utilized to augment the plant staff should have applicable previous nuclear power plant testing or operating experience.

III. REVIEW PROCEDURES

Preparations for the review of Chapter 14 of the PSAR should include familiarization with the applicant's commitments in Chapter 13 pertaining to organization and staffing, plant procedures, and training, and commitments in Chapter 17 pertaining to transfer of plant structures, systems, and components from the vendor or constructors to the applicant when construction or installation is completed. The reviewer should also be familiar with the contents of Regulatory Guide 1.68 and ANSI N18.7-1972. Although the specific tests discussed in Regulatory Guide 1.68 are applicable to water-cooled power reactors, the general test program requirements contained therein should be applicable to test programs for gas-cooled power reactors and can be used as a basis for evaluation of such programs.

The review consists of an analysis of the information submitted in Chapter 14 of the PSAR and a detailed comparison with each of the acceptance criteria contained in II above. Coordination of the review with the assigned reviewer in the Operator Licensing Branch may be necessary for items 5, 6, and 7 listed in I above. Coordination of the review with the assigned licensing project manager may be necessary for item 2 listed in I above.

When the reviewer has determined that each of the acceptance criteria listed in II above has been satisfied, based upon the information presented in the PSAR, the review is complete. Any unjustified deviations from the acceptance criteria should be identified and this information transmitted to the assigned project manager for resolution with the applicant.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been presented in the PSAR to support conclusions of the following type, to be included in the staff's safety evaluation report:

"The applicant has committed to conduct a comprehensive initial test program for the facility. The program described by the applicant is considered to be acceptable and should, when implemented, provide for further verification of the functional adequacy of the facility."

V. REFERENCES

1. ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," American National Standards Institute.

2. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 14.2 INITIAL PLANT TEST PROGRAMS - FSAR

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - Core Performance Branch (CPB)
Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Mechanical Engineering Branch (MEB)
Materials Engineering Branch (MTEB)
Structural Engineering Branch (SEB)

I. AREAS OF REVIEW

The QAB reviews the following areas, relating to initial plant test programs, described in Chapter 14 of the final safety analysis report (FSAR) submitted by the applicant as part of his operating license (OL) application:

1. Summary of Test Program and Objectives

   The summary descriptions for each major phase of the test program and the specific objectives for each major phase are reviewed.

2. Organization and Staffing

   The information provided on the applicant's organizational units and any augmenting organizations or other personnel that will manage or execute any phase of the test program is reviewed. The information provided on the responsibilities, authorities, and qualifications of principal participants is reviewed. The information describing the extent and nature of the participation of the plant operating and technical staff in the test program is reviewed.

3. Test Procedures

   The system the applicant will use to develop, review, and approve individual test procedures is reveiwed. The responsibilities of the organizational units that will perform these activities, the designated functions of each organizational unit, and the general steps to be following in conducting these activities are reviewed. The type and source of design performance information that will be, or is being used in the development of detailed test procedures is reviewed. The format for the test procedures is also reviewed.

4. Conduct of Test Program

The administrative controls that will govern the conduct of each major phase of the test program are reviewed. The specific methods to be used to assure that prerequisites are satisfied for individual tests are reviewed. The procedures to be followed in initiating plant modifications, or repairs that are determined to be required by the test program are reviewed. The controls that will be in effect to require adherence to approved test procedures and the controls for changing test procedures are reviewed. The general prerequisites to be completed before beginning each major test phase are reviewed.

5. Review, Evaluation, and Approval of Test Results

The controls that will govern the review, evaluation, and initial approval of test results for each phase of the test program are reviewed, including the specific controls to be used to assure notification of affected and responsible organizations when test acceptance criteria are not met and the specific controls established to resolve such problems. The applicant's controls relating to the methods and schedules for approval of test data for each major phase are reviewed.

6. Test Records

The applicant's plans pertaining to the disposition of test documents and data obtained during the test program are reviewed.

7. Test Programs' Conformance with Regulatory Guides

The applicant's plans pertaining to conformance with all regulatory guides applicable to initial test programs are reviewed to establish the extent of conformance. Exceptions from applicable regulatory guides are reviewed, along with the justification provided for each guide that is appropriate at the time the FSAR was tendered for review.

8. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

The status of the applicant's program for review of operating experiences at other reactor facilities is reviewed. The degree of conformance to commitments in the preliminary safety analysis report (PSAR) is reviewed. The organizations involved in the study, the summary of qualifications of these organizations, the time span the applicant included in the study, the reactor types (BWRs, PWRs, etc.) or specific reactor facilities included in the study and the sources of information utilized in the study are reviewed. The conclusions or findings of the study and the effect on the test programs are reviewed.

9. Trial Use of Plant Operating and Emergency Procedures

The information pertaining to how, and to what extent, the plant operating and emergency procedures will be use-tested during the test programs is reviewed.

10. Initial Fuel Loading and Initial Criticality

The procedures that will guide initial fuel loading and initial criticality, including the prerequisites and precautionary measures to be used to assure safety, are reviewed.

11. Test Program Schedule

The schedule for conducting each phase of the testing program, relative to the fuel loading date, is reviewed. Information pertaining to anticipated schedule overlap of the test program with test program schedules for other reactor facilities at the site is reviewed. The sequential test schedule for testing individual plant structures, systems, and components for each test phase is reviewed. Also reviewed is the schedule for final approval of testing procedures.

12. Individual Test Descriptions

The listing of individual tests for each test phase is reviewed to establish the degree of conformance with tests identified in Regulatory Guide 1.68, other applicable regulatory guides, and other special testing requirements for the facility, including those identified in Section 14.1.2 of the PSAR. The prerequisites for each test, summary of the test method and objectives, and the acceptance criteria for each test are reviewed to establish that the functional adequacy of those structures, systems or components involved will be verified.

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review described above are as follows:

1. Summary of Test Program and Objectives

The applicant's description should establish that the major phases of the program and the objectives for each phase are consistent with the general guidelines and regulatory positions contained in Regulatory Guide 1.68 or the justification provided for any exceptions should be found to be acceptable by the reviewer.

2. Organization and Staffing

The information provided should conform to the general guidelines and regulatory positions contained in Regulatory Guide 1.68 and ANSI N18.7-1972. The experience and qualification of each principal participant in the test program is found to be acceptable based on the reviewer's judgment. Factors to consider in this judgment are the assigned responsibilities of augmenting personnel and the size and qualifications of the applicant's staff. The applicant's commitments, pertaining to how and to what extent the plant operating and technical staff will be utilized, should provide assurance that such personnel will be used to the maximum extent practicable.

3. Test Procedures

The applicant's system that will be used to develop, review, and approve individual test procedures should provide for appropriate levels of review prior to final approval. The different organizational units or personnel involved in these reviews should possess, by virtue of prior training and experience, the necessary qualifications to provide for a meaningful review. The format for the test procedures should be similar to the format contained in Regulatory Guide 1.68 or the justification for exceptions should be found to be acceptable by the reviewer. The applicant should utilize system designers to provide the functional requirements and performance objectives to be used in developing detailed test procedures. The system designers should include the nuclear steam system (NSS) supplier, architect-engineer, and other major contractors, subcontractors, and vendors, as appropriate.

4. Conduct of Test Program

a. The test program should be conducted using detailed procedures approved by designated management positions within the applicant's organization.

b. The controls used by the applicant to assure that test prerequisites are met should include requirements for inspections, checks, etc., and require sign-off by designated personnel.

c. The controls provided for plant modifications and repairs are found to be acceptable if: (1) the controls are sufficient to assure the required repairs or modifications will be made; (2) the controls will assure retesting is conducted following such modifications or repairs; and (3) the controls will assure a review of any proposed facility modifications by the original design organization or other designated design organizations. The applicant's requirements for documentation associated with such controls should permit audits to be made to assure proper implementation of controls.

d. The controls pertaining to adherence to test procedures and the methods for changing test procedures are found to be acceptable based on the reviewer's judgment.

e. The prerequisites to be completed before each major test phase is begun should be similar to those contained in the general guidelines and regulatory positions stated in Regulatory Guide 1.68 or the justification provided for exceptions should be found to be acceptable by the reviewer.

5. Review, Evaluation, and Approval of Test Results

a. The controls that will govern the review, evaluation, and approval of test results should provide for a technical evaluation of test results by qualified personnel and approval of test results by designated management positions in the applicant's organization.

b. Design organizations should be notified and should participate in the resolution of problems involving design problems that result in a failure to meet test acceptance criteria.

c. The applicant should establish requirements for review and approval of the test data for each major test phase, prior to beginning the next phase.

d. The applicant should establish requirements for review and approval of the test data at each major power test plateau before raising power to the next test plateau during the power ascension test phase.

6. Test Records

The applicant should establish requirements for retention of test procedures and test data throughout the life of the facility.

7. Test Programs' Conformance with Regulatory Guides

The applicant should establish initial test program requirements consistent with regulatory positions established in all applicable regulatory guides or the justification provided for exceptions should be found to be acceptable by the reviewer.

8. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

a. The study conducted by the applicant of operating experiences should be completed in sufficient time to permit the findings to be incorporated in the detailed test procedures and conduct of the test program.

b. The organizations conducting the study for the applicant should be technically qualified by virtue of previous experience or training.

c. The reactor facilities included in the study should include plants similar in design to the applicant's plant.

d. The time frame included in the applicant's study should be sufficient to achieve a recent survey of plant operating problems. In general, this would involve the previous two-year period.

e. The conclusions reached, and the effect on the test program should recognize and account for categories of repeated abnormal occurrences of the same type being experienced and current operating experiences of safety concern.

9. Trial Use of Plant Operating and Emergency Procedures

The applicant should incorporate the plant operating and emergency procedures into the test procedures or otherwise use-test these procedures to the extent practicable during the test program.

10. Initial Fuel Loading and Initial Criticality

The procedures that will guide initial fuel loading and initial criticality should include precautions, prerequisites, and measures consistent with the general guidelines and regulatory positions contained in Regulatory Guide 1.68 or exceptions should be found to be acceptable by the reviewer.

11. Test Program Schedule

a. At least nine months should be allowed for conducting preoperational testing.

b. At least three months should be allowed for conducting startup testing including fuel loading, low power tests and power ascension tests.

c. Overlapping test program schedules should not result in significant divisions of responsibilities or dilutions in the staff provided to implement the test program.

d. The sequential schedule for individual tests should establish test requirements for all plant structures, systems, and components that are relied upon to prevent or to limit or mitigate the consequences of postulated accidents prior to exceeding 25% power level. The schedule should also establish that, insofar as practicable, testing will be accomplished as early in the test program as feasible and that the safety of the plant will not be dependent on the performance of an untested system or feature.

e. Test procedures should be available for review by regulatory inspectors no later than 30 days prior to their intended use.

12. Individual Test Descriptions

The applicant's commitments pertaining to the structures, systems, and components to be covered by the test program should assure the testing of those that: (1) are relied on for the safe shutdown of the facility under normal and faulted conditions; (2) are relied on for establishing conformance with limits or limiting conditions for operation that will be established by the technical specifications; (3) are relied on to prevent or to limit or mitigate the consequences of anticipated transients and postulated accidents. The prerequisites, the test method and objectives, and the acceptance criteria for each test should establish that the functional adequacy of the structures, systems and components involved will be demonstrated.

III. REVIEW PROCEDURES

Preparations for the review of Chapter 14 of the FSAR should include familiarization with the applicant's commitments in Chapter 13 of the PSAR and FSAR pertaining to organization, staffing, plant procedures, and training and commitments in Chapter 17 of the PSAR and FSAR pertaining to the transfer of plant structures, systems, and components to the applicant when construction or installation is completed. The reviewer should also be familiar with the contents of Regulatory Guide 1.68, ANSI N18.7-1972, and other regulatory guides that have applicability to initial test programs. Although the specific tests contained in Regulatory Guide 1.68 are applicable to water-cooled power reactors, the general administrative test program requirements contained therein are also applicable to initial test programs for gas-cooled power reactors and can be used as a basis for evaluation of test programs for these facilities.

The review consists of an analysis of the information submitted in Chapter 14, and of other chapters, as applicable, of the FSAR. A detailed comparison is made with each of the acceptance criteria contained in II above. Coordination of the review with the assigned reviewers for Chapter 13 of the FSAR may be necessary for items 2 and 9 and with the assigned licensing project manager and other branches for item 12, as deemed necessary by the reviewer. In general, the secondary review branches will be requested by the QAB to review the applicant's proposed testing requirements for special, unique, or first-of-a-kind design features to establish the adequacy and validity of test requirements.

The reviewer should be familiar with abnormal occurrences currently being experienced at operating reactors and other problems of safety concern at operating reactors to evaluate item 8 above. Computerized information on abnormal occurrences can be obtained through the Office of Operations Evaluation.

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Summary of Test Program and Objectives

A comparison of the information provided is made with the general guidelines and Regulatory positions in Regulatory Guide 1.68. A comparison should also be made with the information on the initial test program provided in the PSAR and significant differences in commitments evaluated, as applicable.

2. Organization and Staffing

The information provided should be compared with the general guidelines and regulatory positions contained in Regulatory Guide 1.68. The information should also be compared to the applicable guidelines contained in ANSI N18.7-1972. An evaluation of the experience and qualifications of the principal participants in the test program and the organizational structure is made utilizing the general guidance provided in the above referenced documents.

3. Test Procedures

The information pertaining to the development, review, and approval of individual test procedures is examined to determine if adequate controls and organizational arrangements have been established and that key individuals in these activities possess the necessary qualifications and experience. The methods used to develop detailed test procedures are examined to establish that appropriate input from design organizations will be made. The information pertaining to the format of the test procedures is compared with the general guidelines and regulatory positions contained in Regulatory Guide 1.68.

4. Conduct of Test Program

Administrative controls that will govern the initial test program are examined including those relating to plant modifications and repairs, those that require adherence to approved test procedures, and those that specify the methods for changing test procedures. The information provided on prerequisites to be completed before each phase of the test program is begun is compared with the general guidelines and positions contained in Regulatory Guide 1.68 and the applicable guidelines contained in ANSI N18.7-1972.

5. Review, Evaluation, and Approval of Test Results

The controls that will govern the review, evaluation, and initial approval of test results for each phase of the test programs are reviewed. Organizations and individuals participating in these activities are evaluated based on the information provided on qualifications and experience normally contained in Chapters 13 or 14 of the FSAR and on the information on interfaces and organizational arrangements provided in Chapter 14 of the FSAR. The applicant's plans for review and approval of test data for each test phase and at each major power test plateau (or test condition) during the power ascension test phase are compared with the guidelines and regulatory positions contained in Regulatory Guide 1.68.

6. Test Records

Information provided on the applicant's plans for the disposition of initial test program procedures and data from such tests is reviewed to establish that they will be available throughout the anticipated 40-year life of the plant. The administrative controls established for the retention of test documents are reviewed to assure that adequate requirements have been established by the applicant.

7. Test Programs' Conformance with Regulatory Guides

The administrative and technical aspects of the initial test program are compared with all regulatory positions and guidelines contained in regulatory guides applicable at the time the FSAR was tendered for review.

8. Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program

The information provided on the applicant's program for review of operating and testing experiences at other reactor facilities is examined. The adequacy of the applicant's program is evaluated based on the qualifications of the key personnel involved in the program, the relevancy to the applicant's facility of the specific reactors or reactor types included in the review, and the sources and timeliness of information utilized by the applicant. The applicant's conclusions and their effects on the test programs are examined to establish that any abnormal occurrences that have occurred several times and other current safety concerns have been adequately factored into the applicant's test program.

9. Trial Use of Plant Operating and Emergency Procedures

The applicant's plans for trial use of operating and emergency procedures are reviewed to assure that a trial use of procedures will be accomplished during the initial test program to the maximum extent practicable.

10. Initial Fuel Loading and Initial Criticality

The information provided is compared with the general guidelines and regulatory positions contained in Regulatory Guide 1.68. Particular emphasis is placed on the prerequisites, precautions, and sequence of steps to be followed. Justifications for exceptions to regulatory positions contained in Regulatory Guide 1.68 are examined on a case-by-case basis.

11. Test Program Schedule

The information provided is checked against the list in Section II.11 of this plan. The reviewer assures himself, by review of previously licensed applications and by review of applicable chapters of the FSAR and Section 14.1.2 of the PSAR, that the scheduled time periods for each test phase and the sequential method of testing of plant structures, systems, and components are adequate.

12. Individual Test Descriptions

The reviewer examines individual test abstracts to assure that the prerequisites, the test method and objectives, and the acceptance criteria provide reasonable assurance that performance characteristics of systems, structures, and design features will be demonstrated by adequate testing.

IV. EVALUATION FINDINGS

When the review of the information in the FSAR is complete and the reviewer has determined that is satisfactory and in accordance with the acceptance criteria in II above, a statement of the following type should be provided for the staff's safety evaluation report:

"The staff has reviewed the information provided in the final safety analysis report on the applicant's initial test program. This review included an evaluation of: (1) the applicant's organization and staffing for the development, conduct, and evaluation of the test program; (2) the qualifications and experience of the principal participants managing and supervising the test program; (3) the administrative controls that will govern the development, conduct, and evaluation of the test program; (4) the participation of the plant operating and technical staff in the test program; (5) the applicant's requirements pertaining to the trial-use of plant operating and emergency procedures during the test program; (6) the schedule for conducting the test program; (7) the sequence of testing to be followed; and (8) the methods for conducting individual tests and the acceptance criteria to be used in evaluating the test results for plant structures, systems, and components. The review also included an evaluation of the applicant's study of reactor plant operating experiences, conducted to determine where improvement or emphasis was warranted in his initial test program and the applicant's conclusions and actions resulting from this study. The staff has concluded that the information provided in the application describes an acceptable initial test program that will demonstrate the functional adequacy of plant structures, systems, and components."

V. REFERENCES

1. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

2. ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants," American National Standards Institute.

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SECTION 15.0 INTRODUCTION

The Division of Technical Review branches that have responsibility for review of the postulated transients and accidents listed in this chapter are the Reactor Systems Branch (RSB), the Core Performance Branch (CPB), the Accident Analysis Branch (AAB), and the Effluent Treatment Systems Branch (ETSB). RSB reviews analyses of the transients and accidents from the viewpoint of systems operation and performance. CPB reviews analyses of core, fuel, and thermal-hydraulic behavior during postulated accidents. ETSB reviews analyses of postulated spills of radioactive material outside containment. AAB evaluates possible radiological consequences of the transients and accidents.

Events such as fires, floods, storms, or earthquakes are not explicitly considered in the review of postulated transients and accidents in Chapter 15. Rather, the consequences and effects of these postulated events are reviewed under other standard review plans (SRP's), as listed in Table 15-1 (attached).

The reviewers are responsible for the selection and emphasis of aspects of the reviews of transients and accidents in this chapter. Judgment on the areas to be given attention during each review is based on an inspection of the information provided, the similarity of this material to that recently reviewed on other plants, and whether items of special or unique safety significance are involved.

The Reactor Systems Branch reviews nearly all the postulated transients and accidents. Various anticipated process disturbances, equipment malfunctions, and postulated component failures are examined to evaluate the plant capability to control or accommodate such failures and malfunctions. The RSB review includes an analysis of the pressure to which the reactor and steam systems are subjected and the need for engineered safety features to mitigate the consequences of accidents. The impact of various single failures on the course of a transient or accident is also considered. The containment and subcompartment response to postulated accidents is reviewed by the Containment Systems Branch under SRP 6.2.1.

The Core Performance Branch is responsible for the review of all physics data presented in this chapter. This includes power levels, power distributions, Doppler coefficients, moderator temperature coefficients, void coefficients, reactor kinetics parameters, and control rod worths. The CPB review includes the evaluation of possible damage to the fuel. CPB also reviews the operating bands and uncertainty bands associated with these variables and assists RSB as requested.

The review of transients and accidents by RSB requires an evaluation of results, presented in the application, of analytical methods which frequently are not documented in the application. In such cases, the applicant may refer to a vendor topical report. The methods include DNB (departure from nucleate boiling) correlation development, subchannel analysis, system transient analysis, analysis of RIA (reactivity-initiated accidents), and LOCA (loss-of-coolant accident) analysis. For those cases where applicants use techniques previously considered and approved by the staff, additional review of methods may not be required. However, if new methods are involved, the CPB performs a review of topical reports and other information which describe the method of analysis. Such a review generally includes vendor model description, data correlations and empirical relationships, solution techniques, summary of computer codes if involved, sample problems, experimental verification, and comparative calculations.

In its review of transients and accidents, RSB may request from CPB, from time to time, an independent check of the results submitted by the applicant. In such cases, CPB obtains input data from the applicant for use in auditing-codes available to the staff.

Upon request of RSB, the Electrical, Instrumentation and Control Systems Branch provides assistance in evaluating the sequence of postulated events, protective and safeguards systems actuation and potential bypass modes, and manual control. EICSB determines whether reactor protection and safeguards controls and instrumentation will function as assumed in the transient analysis with regard to manual or automatic actuation; remote sensing, indication, and control; onsite emergency power systems; and interlocks with auxiliary or shared systems.

The Accident Analysis Branch review is concentrated on those more severe accidents that could result in the release of radioactive materials and could have significant radiological consequences involving the general public. AAB determines the potential doses resulting from the accidents and compares these doses to established dose criteria and guidelines. Based on the results of these analyses, AAB determines the adequacy of equipment designed to mitigate radiological consequences. In addition, radiological analyses are made to determine certain technical specification limits for safety-related equipment and structures.

TABLE 15-1

Postulated Event		Standard Review Plans	Branches Having Primary Review Responsibility
Accidents at Nearby Locations	2.2.3	Evaluation of Potential Accidents	Accident Analysis Branch
Storms	2.3	Meteorology	Site Analysis Branch
	3.3	Wind and Tornado Loadings	Structural Engineering Branch
Floods	2.4	Hydrologic Engineering	Site Analysis Branch
	3.4	Water Level (Flood) Design	Auxiliary and Power Conversion Systems Branch
Earthquakes	2.5	Geology and Seismology	Site Analysis Branch
	3.2	Classification of Structures, Components and Systems	Reactor Systems Branch
	3.7	Seismic Design of Structures	Structural Engineering Branch
	3.8	Design of Category I Structures	Structural Engineering Branch
	3.10	Seismic Design of Category I Instrumentation and Electrical Equipment	Electrical, Instrumentation and Control Systems Branch
Missiles	3.5	Missile Protection	Accident Analysis Branch Reactor Systems Branch Auxiliary and Power Conversion Systems Branch Structural Engineering Branch
Fires	9.5.1	Fire Protection System	Auxiliary and Power Conversion Systems Branch

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.1.1, 15.1.2, 15.1.3, 15.1.4 — DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency, and which involve an unplanned increase in heat removal by the secondary system, are covered by this review plan. Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. The power level increase will lead to a reactor trip. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure.

Each of the transients covered by this plan should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated include:

1. Pressurized Water Reactors (PWR's) and Boiling Water Reactors (BWR's)
   a. Feedwater system malfunctions that result in a decrease in feedwater temperature.
   b. Feedwater system malfunctions that result in an increase in feedwater flow.
   c. Steam pressure regulator malfunctions or failures that result in increased steam flow.
2. PWR's Only
   a. Inadvertent opening of a steam generator relief or safety valve.

The topics covered in the primary review include: postulated initial core and reactor conditions which are pertinent to feedwater system malfunctions, pressure regulator or safety or relief valve malfunctions, methods of thermal and hydraulic analysis, postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor system components, functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power, vessel pressure, steam line pressure (for BWR's), steam line flow (for BWR's), feedwater flow (for BWR's), and reactivity.

The sequence of events described in the SAR for these transients is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrument and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients which result from an increase in heat removal are:

   a. To identify which of the moderate-frequency* transients that result in increased heat removal are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

*The term "moderate-frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

a. Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used for both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

RSB reviews the applicant's description of the transients caused by excessive heat removal with specific attention to the occurrences that lead to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that a particular transient involving an increase in heat removal is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the increase-in-heat-removal transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the SRP for Chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the values of reactivity coefficients and control rod worths used by the applicant in his analysis, and the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the justification provided by the applicant to show that the core burnup selected yields the minimum margins. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of parameters listed in Sections 15.x.x.3(c) and 15.x.x.4(c) of the Standard Format (Ref. 1) is reviewed. The values of the more important of these parameters, as listed in Section I of this SRP, are compared to those predicted for other similar plants to see that they are within the range expected.

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:

"A number of plant transients can result in an unplanned increase in heat removal by the secondary system. Those that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulator malfunctions or the inadvertent opening of a steam generator safety or relief valve (PWR only). All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _________ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the _________ transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)* did not decrease below _________ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with respect to transients resulting in an unplanned increase in heat removal by the secondary system that are expected to occur with moderate frequency."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*Minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR) for a BWR.

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.1.5 SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system and results in a reduction of coolant temperature and pressure. Due to the negative moderator temperature coefficient this cooldown causes an increase in core reactivity. The core reactivity increase causes a power level increase and a decrease in shutdown margin. If the plant is at power, the reactor is automatically tripped and the main steam and feedwater line isolation valves are automatically closed. Decay heat is removed through the unaffected steam generators by venting steam from the secondary system safety and relief valves. The auxiliary feedwater system supplies makeup water to the unaffected steam generators.

The transient following a steam line break is sensitive to the discharge rate so that a range of break sizes must be evaluated both inside and outside containment to determine the acceptability of the system response. The course the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (i.e., hot shutdown, full power, one-, two-, or three-loop operation). Analyses with various assumed initial conditions are required to verify that the condition leading to the severest consequences has been identified.

The topics reviewed include: postulated initial core and reactor conditions pertinent to the steam line break accident, methods of thermal and hydraulic analyses, postulated sequence of events including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system, assumed responses of the reactor coolant and auxiliary systems, functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events, and operator actions required to secure and maintain the reactor in a safe shutdown condition.

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include reactor coolant system (RCS) pressure, steam generator pressure, fluid temperatures, fuel and clad temperatures, discharge flow rate, steam line and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

APCSB reviews the auxiliary feedwater system to see that it can function following a steam line break given a single active component failure with either onsite or offsite power. This is done as described in Standard Review Plan (SRP) 10.4.9. RSB reviews the auxiliary feedwater system to see that the flow provided is acceptable for controlling the transient following a steam line break.

MEB evaluates potential water-hammer effects on safety valve integrity.

AAB evaluates the fission product release and verifies that the radiological consequences resulting from a steam line break are within acceptable limits. This evaluation is performed for the design basis case as described in the appendix to this review plan.

CSB, as described in SRP 6.2.1, evaluates the response of the containment to ruptures of steam lines with regard to the effects of pressure and temperature on the containment functional capabilities.

II. ACCEPTANCE CRITERIA

1. The general objective of the review of steam line rupture events is to confirm that the primary reactor coolant system is maintained in a safe status for a range of steam line ruptures up to and including a break equivalent in area to the double-ended rupture of the largest steam line.

2. The specific criteria against which the consequences of these breaks are to be evaluated are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above 1.30 or 1.32, as appropriate, based on correlations given in References 3 and 4. If the DNBR falls below these values, fuel damage (rod perforation) must be assumed unless it can be shown, based on an acceptable fuel damage model, that fuel failure has not occurred. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

   c. The radiological criteria used in the evaluation of steam system pipe break accidents (PWR's only) appear in the appendix to this plan.

3. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

   a. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system (NSSS) design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR.

   b. Assumptions as to the loss of offsite power and the time of loss should be made so as to maximize the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break, or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feed-water pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents.

c. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Positions APCSB 3-1 and MEB 3-1 (Ref. 8).

d. The worst single active component failure should be assumed to occur. The assumed single failure may cause more than one steam generator to blow down, or may be in any of the systems required to control the transient.

e. The maximum-worth rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results, but for the analysis of steam line break accidents this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

1. The reviewer determines the acceptability of the analytical models and assumptions as follows:

   a. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparing with an acceptable analysis performed on another plant of similar design, or by comparing with staff calculations for typical plants done by CPB on request.

b. The acceptability of the equations, sensitivity studies, and models proposed by the applicant are evaluated. For situations where new generic methods are proposed, the reviewer will request an evaluation by CPB. Acceptable equations, sensitivity studies, and models are described in References 5, 6, and 7.

c. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

(1) Reactor trip signal: credit taken for any reactor trip signal is reviewed by EICSB to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

(2) Emergency core cooling system (ECCS): credit taken for actuation of the ECCS is reviewed by EICSB to verify the ability of the instrumentation and control systems to respond as assumed.

(3) Auxiliary feedwater system: the availability of the auxiliary feedwater system to supply adequate auxiliary feedwater flow to the intact steam generators during the accident and the subsequent shutdown condition is evaluated. This is done by APCSB as to availability of the system and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion.

d. The variations with time during the transient of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the steam line break accident (as listed in Section I of this SRP) are compared with those predicted for other similar plants to see that they are within the range expected.

2. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. This phase of the review is done using the system review procedures described in the SRP for Chapters 5, 6, 7, 8, and 10 of the SAR. The reviewer also considers single failures that may cause more than one steam generator to blow down, thus increasing the reactivity addition to the core.

3. The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam

isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for auxiliary feedwater system actuation, safety injection signal delay time, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe shutdown condition.

4. Based on the above information, AAB evaluates the radiological consequences of the design basis steam line break accident as described in the appendix to this plan.

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 analyses and effects of steam line break accidents inside and outside containment, during various modes of operation and with or without offsite power, have been reviewed. The accident which resulted in the most severe consequences was determined and evaluated using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the spectrum of steam line break accidents showed that fuel damage was minimal and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was _______, resulting in __% of the rods experiencing clad perforation. The maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _________between the water and steam phases and a X/Q value of ______sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _______. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike."

"Based on the above, the staff concludes that the plant design is acceptable with regard to steam line break accidents."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

4. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox Power Generation Systems, BAW-10000, Supplement 1, March 1971.

5. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

6. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

7. "Standard Nuclear Steam Supply System B-SAR-241," Babcock and Wilcox Company, February 1974 (under review).

8. 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.

APPENDIX

STANDARD REVIEW PLAN 15.1.5

RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURES OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The AAB review under this appendix covers the following areas:

1. The plant response to a main steam line failure outside containment with and without a concurrent loss of offsite power.

2. The calculation of whole body and thyroid doses at the site boundary due to the releases resulting from these accidents.

The purposes of the review are to assure that the plant procedures for recovery from a main steam line failure, with and without offsite power available, are properly taken into account in computing the whole body and thyroid doses at the nearest exclusion area boundary, and that postulated releases of radioactive gases due to the failure are adequately limited by the coolant concentration technical specifications.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a main steam line failure outside containment, and the primary and secondary coolant activities appropriately limited, if calculations show that the resulting doses at the nearest exclusion area boundary are small fractions of the 10 CFR Part 100 exposure guidelines, and are within 10 CFR Part 100 guidelines for the case of a coincident iodine spike or for the case of one rod held out of the core.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix 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 detailed review of radiological consequences of the main steam line failure accidents is done at the operating license stage when system parameters are fully developed. At the construction permit stage, there is generally insufficient information available to make meaningful radiological consequence calculations for these accidents. At this stage, the review is limited to a brief review of the applicant's discussion of the main steam line failure accidents to determine that there are no unusual design features that would preclude the limitation of radiological consequences by appropriate limits on coolant concentrations and primary-secondary leakage.

The AAB review of main steam line failure accidents at the operating license stage consists of the following steps:

1. Review of the applicant's step-by-step descriptions of the steam line failure accidents (with and without offsite power). This includes a review of the time steps used in the descriptions, the bases for their selection, and assurance of an adequate degree of conservatism.

2. Review of the signals available to the reactor operator that indicate the occurrence of the accident and the state of the system throughout the recovery procedure. Automatic and required manual operations by the operator as a function of time must also be determined.

3. Preparation of input data required to run a digital computer code (Ref. 2) based on the above information. For this purpose the reviewer sets up a series of time intervals similar to those described by the applicant or modified, if necessary, in order to obtain an adequate degree of conservatism (Ref. 3). The values of parameters describing the primary and secondary system operating conditions are obtained from the summary table that the applicant provides in this section of the safety analysis report (SAR). These data may also be found elsewhere in the SAR, mainly in Chapters 4 and 10.

4. Determination of the meteorological parameters for the dose calculation. The SAB provides the reviewer with the distance to the nearest exclusion area boundary and the accident condition (5 percentile) wind speed. The X/Q value for the calculation of the two-hour doses is obtained from Regulatory Guide 1.5 (Ref. 4) and corrected for winds differing from 1 m/sec (inverse ratio).

5. Determination of parameters for the thyroid dose calculation. An appropriate value for the iodine decontamination factor is used in the calculation of the thyroid doses. A decontamination factor of 10 is currently used between the water and steam phases for most plants unless the applicant presents reasonable evidence that the use of another value is justified, and the staff concurs after review of the evidence. A breathing rate of 3.47×10^{-4} m^3/sec is used in calculating the thyroid doses.

6. Determination of coolant activity concentrations. The reviewer assumes the primary and secondary coolant activity concentrations allowed by the technical specifications (SAR Chapter 16) as equilibrium conditions prior to the accidents. Additional inventory may become available for release due to fuel failures resulting from the accident.

The RSB, on a generic basis, reviews the effect of a main steam line failure on core thermal margins. If this event is predicted to cause fuel failures, RSB notifies AAB so that the predicted magnitude and extent of fuel failures can be properly considered in evaluating the radiological consequences.

7. Determination of iodine spiking effects. The effect of iodine spiking following the accident (Ref. 5) can be accounted for by increasing the iodine source term in the primary system upon depressurization. At the present time the I-131 equivalent source term (release rate from the fuel) is increased by a factor of 500 at the time of reactor trip. A case with an iodine spike which already exists (due to a previous power transient) is also considered assuming the I-131 equivalent coolant concentration technical specification limit for an iodine spike.

8. Determination of the leakage into the steam system. Normal operating primary-to-secondary leakage is assumed to exist in the steam generators. The leakage rate should be the maximum allowed by the technical specifications (SAR Chapter 16). Currently this value is about 1 gpm but may be lower if required because of fuel densification, rod ejection accident consequences, or consequences of an anticipated transient without scram (ATWS).

9. Calculation of the nearest exclusion area boundary doses from the steam line failure. The reviewer uses a digital computer code (Ref. 2), with the input data and assumptions as developed in the steps above, to determine nearest exclusion area boundary doses for the steam line failure accident. Dose calculations are made with and without coincident iodine spiking.

10. Review of the results of the dose calculations. The reviewer compares the nearest exclusion area boundary doses, calculated without coincident iodine spiking, to the 10 CFR Part 100 guidelines. If the doses are a small fraction of the guideline values, the design is accepted. If not, the coolant concentration and primary-secondary leakage limits allowed by the technical specificaitons are reduced accordingly. The doses calculated with coincident iodine spiking are also compared with the 10 CFR Part 100 guidelines, and should be within the guidelines to be acceptable. If they are not, appropriate reductions of the technical specificaiton limits on coolant concentrations and primary-secondary leakage are made.

11. Review of the effects of possibel fuel damage in the accident on nearest exclusion area boundary doses. The reviewer should assume that the applicant's calculations of fuel damage are correct for the case of a control rod held at the fully withdrawn position unless informed otherwise by the RSB. If fuel damage does occur, calculations should be performed in order to assure that 10 CFR Part 100 guidelines are not exceeded (without a coincident spike).

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 with the RSB findings in the staff's evaluation report at the operating license stage:

"The radioactivity release has been evaluated using the computer code ______ and a conservative description of the plant response to the accident. A decontamination factor of __________ between the water and steam phases and a X/Q value of ______ sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table __________. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike."

The following paragraph is added if fuel damage is found to be a possible consequence of the accident:

"The evaluation of the main steam line failure outside containment accident has also been evaluated with ________% fuel damage in the core (as a result of the most reactive control rod remaining fully withdrawn). The resulting doses are within the guidelines of 10 CFR Part 100 provided the normal operating primary-to-secondary secondary leakage is limited to ________ gpm."

At the construction permit stage the following paragraph is included with the RSB findings in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam line and steam generator tube failure accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications."

V. REFERENCES

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

2. Computer codes are currently under development. Documentation will be published in a NUREG report.

3. H. M. Fontecilla, "Analysis of Accidental Iodine Releases from the Secondary Coolant System," ANS Transactions, Vol. 17, November 1973.

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

5. W. F. Pasedag, "Effects of Iodine Spiking on Light-Water Reactor Accident Analysis," ANS Transactions, Vol. 17, November 1973.

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STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.2.1
15.2.2
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15.2.5

LOSS OF EXTERNAL LOAD, TURBINE TRIP, LOSS OF CONDENSER VACUUM, CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR), AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency result in an unplanned decrease in heat removal by the secondary system. These are covered in this review plan. Each of the transients should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated are:

1. Loss of External Load

In a loss of external load event an electrical disturbance causes loss of a significant portion of the generator load. This loss of load situation is different from the loss of a-c power condition considered in Standard Review Plan (SRP) 15.2.6 in that offsite a-c power remains available to operate the station auxiliaries (such as reactor coolant pumps). The onsite emergency diesels are therefore not required for the loss of external load transient. Immediate fast closure of the turbine control valves (TCV) and intercept valves is initiated whenever a loss of generator load takes place. For a boiling water reactor (BWR), a fast TCV closure (0.150-0.2 sec) causes a sudden reduction in steam flow and results in a reactor pressure surge. For a BWR without select rod insert (SRI), reactor scram occurs. For a pressurized water reactor (PWR) there is also a sudden reduction in steam flow, and this causes the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. For a PWR with an integrated control system, reactor power can be run back to a lower level on TCV closure.

In all light water-cooled reactors, sensible and decay heat can be removed through actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling systems, and auxiliary feedwater system (PWR).

2. Turbine Trip

In a turbine trip event a malfunction of a turbine or reactor system causes the turbine to be tripped off the line by abruptly stopping steam flow to the turbine. This is different from the loss of electrical load condition described above in that fast closure of the turbine stop valves (TSV) is initiated. The TSV have faster (0.1 sec) closure times than the turbine control valves, resulting in more severe transients. For typical BWR and PWR plants, position switches on the TSV sense the trip and initiate reactor scram. The remainder of this transient is similar to the previously discussed loss of electrical load.

3. Loss of Condenser Vacuum

A loss of condenser vacuum event is one of the malfunctions that can cause a turbine trip. The remarks in 2, above, thus apply to this transient.

4. Main Steam Isolation Valve Closure

The main steam isolation valve (MSIV) transient for BWR's can be initiated by various steam line or reactor system malfunctions and by various operator actions. As the MSIV's close, position switches initiate a reactor scram when the valves in three or more of the steam lines are less than 90% open, the reactor pressure is above 600 psi, and the reactor mode switch is in the RUN position. The effect of MSIV closure is to limit steam flow to the turbine. The results are similar to those discussed in 1, above, but tend to be less severe since the MSIV closure time is much longer than that of the TCV.

5. Steam Pressure Regulator Failure

Steam pressure regulator failure in a closed position yields a transient similar to the transients discussed above. Generally, because the rate of change of system parameters is slower for a steam pressure regulator failure, a less severe transient results.

The review of the transients described above includes the sequence of events, the analytical models, the values of parameters used in the analytical models, and the predicted consequences of the transients.

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient analyses are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the analyses are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients listed in Section I are:

   a. To identify which of the moderate-frequency transients that result in an unplanned decrease in secondary system heat removal are the most limiting. (The term "moderate frequency" is used in this review plan in the same sense as in the definitions of design and plant process conditions in References 8 and 9.)

   b. To verify that, for the most limiting transients, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant should analyze these transients using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are

acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of these transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that any one of these transients is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the reduction-of-heat-removal transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the limiting transient are compared to those predicted for other similar plants to verify that they are within expected range.

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 (SER):

"A number of plant transients can result in an unplanned decrease in heat removal by the secondary system. Those that might be expected to occur with moderate frequency are turbine trip, loss of external load, steam pressure regulator malfunctions, main steam isolation valve closure (in BWR's), loss of condenser vacuum, loss of non-emergency a-c power to the station auxiliaries, and loss of normal feedwater flow.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _______ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the ___________ transient showed that cladding integrity was maintained by ensuring that the maximum departure from nucleate boiling ratio (or minimum critical heat ratio for a BWR) did not decrease below ______ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

"The staff concludes that the plant design is acceptable with regard to transients resulting in an unplanned decrease in heat removal by the secondary system that are expected to occur with moderate frequency."

*The SER should present one statement for moderate frequency transients involving an unplanned decrease in heat removal by the secondary system. Thus, the results of the reviews under SRP 15.2.6 and 15.2.7 are included in this statement.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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SECTION 15.2.6 LOSS OF NON-EMERGENCY A-C POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The loss of non-emergency a-c power is assumed to result in the loss of all power to the station auxiliaries. This situation could result either from a complete loss of the external grid (offsite) or a loss of the onsite a-c distribution system. It is different from the loss of load condition considered in Standard Review Plan (SRP) 15.2.2 because, in the latter case, a-c power remains available to operate the station auxiliaries. The major difference is that in the loss of a-c power transient all the reactor coolant circulation pumps are simultaneously tripped by the initiating event. This causes a flow coastdown as well as a decrease in heat removal by the secondary system.

Within a few seconds the turbine trips and the reactor coolant system is isolated, causing the pressure and temperature of the coolant to increase. A reactor trip is initiated by the turbine trip. For this transient, the diesel generators are automatically started and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, reactor core isolation cooling system in a boiling water reactor (BWR), emergency core cooling system, and auxiliary feedwater system in a pressurized water reactor (PWR).

The review of the loss of a-c power transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient analysis are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objective in reviewing the loss of a-c power transient is to confirm that either of the following criteria are met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of heat removal by the secondary system and has the same anticipated frequency classification.

   b. The plant responds to the loss of a-c power transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

*The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

3. The applicant's analysis of the loss of a-c power transient should be based on an acceptable model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If the applicant proposes other analytical methods, these are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. Use of the following parameter values is considered acceptable:

a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during the review of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of a-c power transient presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. The operation of standby diesel generators that is required.

If the SAR states that the loss of a-c power transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of a-c power transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, standby diesel generator and other systems needed to limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This aspect of the review uses the procedures described in standard review plans for chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of parameters listed in 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of a-c power transient are compared to those predicted for other similar plants to verify that they are within the expected range.

IV. EVALUATION FINDINGS

The evaluation findings under this plan are incorporated in a statement covering all transients of moderate frequency involving a decrease in heat removal by the secondary system. See the findings statement in SRP 15.2.1-5 for a typical statement.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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SECTION 15.2.7 LOSS OF NORMAL FEEDWATER FLOW

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power. The sequence of events for the loss of feedwater transient differs between a boiling water reactor (BWR) and a pressurized water reactor (PWR). A PWR has a backup (auxiliary and emergency) feedwater system while a BWR relies on the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system for backup core cooling. In either case, loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage.

For both PWR's and BWR's, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow. This can be accomplished by actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling system, and auxiliary feedwater system (PWR).

The review of the loss of feedwater transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the predicted results of the transient are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objective in the review of the loss of normal feedwater transient is to confirm that one of the following criteria are met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of heat removal by the secondary system, and has the same anticipated frequency classification.

   b. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

*The term "moderate frequency" is used in this review plan in the same sense as in the definitions of design and plant process conditions in References 8 and 9.

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If the applicant proposes to use other analytical methods, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

   a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty).

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during the reviews of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of normal feedwater flow transient presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. The operation of auxiliary systems that is required.

If the SAR states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the loss of feedwater transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. This part of the review uses the procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of parameters listed in Section 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of normal feedwater transient are compared to those predicted for other similar plants to see that they are within the range expected.

IV. EVALUATION FINDINGS

The evaluation findings under this plan are incorporated in a statement covering all transients of moderate frequency involving an unplanned decrease in heat removal by the secondary system. SRP 15.2.1-5 contains a general statement covering the findings of this plan.

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Over-pressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

8. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

9. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

The transient that results from a postulated feedwater line break is sensitive to the discharge rate; consequently, a range of break sizes should be evaluated both inside and outside containment to determine the acceptability of the response. Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup (by reducing feedwater flow to the affected steam generator). Therefore, analyses of various postulated break sizes and locations are needed to identify the particular situation that is most limiting with respect to system effects.

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may be such to prevent addition of any feedwater to the affected steam generator. An auxiliary feedwater system is therefore provided to assure that feedwater is available to provide decay heat removal.

The review includes evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break, the methods of thermal and hydraulic analysis, the postulated sequence of events including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system actions, the assumed response of the reactor coolant and auxiliary systems, the functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition. The results of the analyses are reviewed to ensure that the values of pertinent system parameters, discussed in Section II below, are within expected ranges. The parameters of importance for these transients include reactor coolant system pressure, steam generator pressure, fluid temperatures, fuel and clad temperatures, discharge flow rate, steam line and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, mass and energy transfer within the containment (for breaks inside containment), reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

APCSB reviews the auxiliary feedwater system to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power. This review is performed as described in Standard Review Plan (SRP) 10.4.9.

RSB reviews the auxiliary feedwater system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break.

MEB evaluates potential water-hammer effects on safety valve integrity.

AAB evaluates the fission product release assumptions used in determining any offsite releases. AAB verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits. This analysis is done independently of the RSB review.

CSB, under SRP 6.2.1, evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities.

II. ACCEPTANCE CRITERIA

1. The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for a range of feedwater line breaks up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

2. The specific criteria used in evaluating the consequences of these breaks are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 3).

   b. The potential for core damage should be evaluated on the basis that it is acceptable if the minimum DNBR remains above 1.30 or 1.32, as appropriate, based on correlations given in References 4 and 5. If the DNBR falls below these values, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

   c. Any activity release must be such that the calculated doses at the site boundary are well within the guidelines of 10 CFR Part 100.

3. There are certain assumptions which should be used in the analysis regarding important parameters that describe initial plant conditions and postulated system failures. These are listed below.

   a. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced if considered applicable.

   b. The assumptions as to whether offsite power is lost and the time of loss should be made conservatively. Offsite power may be lost simultaneously with the occurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should be made to determine the most conservative assumption appropriate to the plant design being reviewed. The study should take account of the effects that loss of offsite power has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater, and the resulting modification of the sequence of events.

   c. The effects of the postulated feedwater line breaks on other systems (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) should be considered in a manner consistent with the intent of Branch Technical Positions APCSB 3-1 and MEB 3-1 (Ref. 9).

d. The worst single active component failure should be assumed to have occurred in the systems required to control the transient.

e. The maximum rod worth should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be assumed.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results for the case of a feedwater line rupture inside containment. However, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

III. REVIEW PROCEDURES

The procedures below are used during reviews of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB and are compared to the initial conditions listed in Section II of this plan. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of reactivity parameters used in the applicant's analysis.

Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The equations, sensitivity studies, and models proposed by the applicant are reviewed by RSB to determine if these have been previously reviewed and found acceptable by the staff. For situations where new generic methods are proposed, the reviewer will request an evaluation by CPB. Acceptable equations, sensitivity studies, and models are described in References 6, 7, and 8.

Credit taken for a reactor trip signal or for actuation of engineered safety features should be reviewed by EICSB to determine the ability of the instrumentation and control systems to respond as assumed under accident conditions.

The ability of the auxiliary feedwater system to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated by APCSB as to availability and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to determine the maximum times permitted for their completion.

To the extent considered necessary, the RSB reviewer evaluates the effect of single active failures of systems and components that may alter the course of the accident. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, 8, and 10 of the SAR. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 2) are reviewed. The more important of these parameters for the feedwater line break accident (as listed in Section I of this SRP) are compared to those predicted for other similar plants to see that they are within the expected range.

The reviewer confirms that the amount of secondary coolant expelled from the system has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparison with an acceptable analysis performed on another plant of similar design, or by comparison with staff calculations for typical plants which will be available from CPB on request.

The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests to verify that valve discharge rates and response times including, for example, opening and closing times (delay times) for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves, have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for actuation of the auxiliary feedwater system, safety injection signal delay time, and delay times for delivery of any high concentration boron injection required to bring the plant to a safe shutdown condition.

Using the information developed in the review, the AAB reviewer evaluates the radiological consequences of the design basis feedwater line break. This evaluation based on a qualitative comparison with the results of the design basis steam line break, or on a detailed analysis using the approach described in the appendix to Standard Review Plan 15.1.5.

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 analyses and effects of a spectrum of feedwater line breaks inside and outside containment, during various modes of operation, and with or without offsite power have been reviewed. The accident which resulted in the most severe consequences was determined and evaluated using a mathematical model that had been previously reviewed and found acceptable by the staff. The values of the parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analyses of the spectrum of feedwater line break accidents showed that fuel damage was minimal and that no loss of core cooling capability would result. The minimum departure from nucleate boiling ratio experienced by any fuel rod was ________, resulting in ______ of the rods experiencing cladding perforation. The maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The radiological consequences of the design basis feedwater line break have been evaluated. Technical specification limits on primary and secondary coolant activities limit potential doses to small fractions of 10 CFR Part 100 exposure guidelines.

"Based on the above, the staff concludes that the plant design is acceptable with regard to feedwater line break accidents."

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.

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

5. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox Power Generation Systems, BAW-10000, Supplement 1, March 1971.

6. "Reference Safety Analysis Report RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973 (under review).

7. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

8. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

9. 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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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.3.1 LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING
15.3.2 TRIP OF PUMP AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. The resulting increase in clad temperature could result in fuel damage. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are covered by this review plan. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref. 1).

Core thermal and hydraulic transients associated with partial and complete loss of reactor coolant flow are evaluated. These include:

1. For boiling water reactors (BWR's), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller to cause decreasing flow.

2. For pressurized water reactors (PWR's), partial and complete reactor coolant pump trips.

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump, a fault in the power supply to the pump, a pump trip caused by such anomalies as over-current or phase imbalance, or a failure within the recirculation flow control network (BWR) resulting in decreasing flow. A complete loss of forced coolant flow may result from the simultaneous loss of electrical power to all pumps.

The review includes the postulated initial core and reactor conditions which are pertinent to the loss of flow transient, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor systems components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's flow transient analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. These parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), steam line flow (BWR), and feed-water flow (BWR).

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety system, and operator action to secure and maintain the reactor in a safe condition. The EICSB review, as described in Standard Review Plans (SRP) 7.2 and 7.3, concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. CPB, as described in the appendix to SRP 4.4, performs generic reviews of the thermal-hydraulic computer models used for this transient. CPB also performs, upon request, additional analyses related to these accidents for selected reactor types.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB engineer to contact his counterpart in CPB to ensure that the relevant physics data have been used in any staff calculations.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of loss of forced reactor coolant flow transients are:

   a. To identify which of the transients are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the loss of flow transients in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency* are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures. (Ref. 2).

* The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 7 and 8.

b. Fuel cladding integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 is satisfied throughout the transient.

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of the loss of reactor coolant flow transients should use an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of parameters used in the analytical model should be suitably conservative. The use of the following values is considered acceptable:

a. The reactor is initially at rated output (licensed core thermal power) for the number of loops assumed operating, plus 2% to account for power measurement uncertainty.

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

II. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of each of the loss of reactor coolant flow transients presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that are required.

5. The extent to which operator actions are required.

If the SAR states that a particular loss of flow transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The reviewer confirms that all types of flow loss transients are considered, e.g., pump trips during two-, three- and four-loop operation. The applicant is to present a quantitative analysis in the SAR of the loss of flow transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the loss of flow. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operation action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by the RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the applicant's proposed model.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for the loss of reactor coolant flow transients are compared to those predicted for other similar plants to verify that they are within the expected range.

The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests to verity flow coastdown calculations.

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 (SER):

"Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. The ones expected to occur during the life of the plant are those caused by reactor coolant (or recirculation) pump trips or a flow controller malfunction.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ________ transient. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The values of the parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the ________ transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below ________ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with regard to transients that are expected to occur during plant life and result in a decrease in reactor coolant flow rate."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review.)

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December, 1973 (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

*The SER should present one statement for all similar transients.
**Minimum critical heat flux ratio (MCHFR) or minimum critical power ratio (MCPR) for a BWR.

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

8. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.3.3 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT
15.3.4 PUMP SHAFT BREAK

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time.

This review plan is intended to cover both of these infrequent transients. Each of these transients should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1).

The review is concerned with the postulated initial core and reactor conditions that are pertinent to the rotor seizure or broken shaft events, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, the assumed reactions of reactor system components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The parameters include: peak clad temperature, peak fuel temperature, core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio or critical power ratio, departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), steam line flow (BWR), and feedwater flow (BWR).

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer, as described in Standard Review Plans (SRP) 7.2 and 7.3, concentrates on the instrumentation and controls aspects of the sequence described in the SAR and evaluates whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. CPB, as described in SRP 4.4, performs generic reviews of the thermal-hydraulic computer models used for this transient. CPB also performs, upon request, additional analyses related to these accidents for selected reactor types.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB engineer to contact his counterpart in CPB to ensure that the relevant physics data have been used in any staff calculations.

AAB is notified regarding the extent of fuel failures that are predicted by the analysis. AAB then evaluates the radiological consequences.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients resulting from a rotor seizure or shaft break in a reactor coolant pump are:

    a. To identify which of these infrequent transients is the more limiting.

    b. To verify that, for the more limiting transient, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure (discussed in the following paragraphs) are met.

2. The specific criteria for the rotor seizure and shaft break transients are:

    a. For infrequent incidents*, such as the rotor seizure or shaft break in a reactor coolant pump, the plant should be designed to limit the release of radioactive

*The term "infrequent incidents" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 7 and 8.

material to assure that doses to persons offsite are kept to values which are a small fraction of 10 CFR Part 100 guidelines.

b. The reactor coolant pump rotor seizure or shaft break event should be accommodated with the failure of only a small fraction of the fuel rods in the reactor, and the core geometry should remain intact so there is no loss of core cooling capability. Safety functions should be accomplished assuming the worst single failure of a safety system active component.

c. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

3. The applicant's analyses should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 2 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable:

a. Initial power level is taken as the rated output (licensed core thermal power) for the number of loops assumed operating, plus an allowance of 2% to account for power measurement uncertainty. An analysis to determine the effects of rotor seizure or pump shaft break should be made for each allowed mode of operation (e.g., one-, two-, three-, and four-loop operation) or the effects referenced to a limiting case.

b. Core coolant inlet subcooling is at the minimum of the operating range in the technical specifications so as to maximize the calculated coolant enthalpy in the core.

c. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of core.

d. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicants' analysis of the rotor seizure and shaft break events are reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that one of the transients is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the transient that is determined to be more limiting. For the transient that is found more limiting, the reviewer confirms that the effects of the transient are determined for each mode of operation (e.g., one-, two-, three-, or four-loop) allowed by the technical specifications. Either a separate analysis should be presented for each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1), are reviewed. The more important of these parameters (as listed in Section I, above) are compared to those predicted for other similar plants to confirm that they are within the expected range. In particular, the peak cladding temperature and percentage of fuel rods that experience a departure from nucleate boiling condition are compared.

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 analyses and effects of an instantaneous seizure of a rotor and an instantaneous break of a shaft of a reactor coolant pump* during any allowed mode of operation have been reviewed. It was found that the more limiting of these events was the ________________. This event was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis showed that ___% of the fuel rods experienced departure from nucleate boiling (DNB) and that the peak clad temperature reached was ___°F. This assures that the fuel damage will be minimal and that no loss of core cooling capability will result. The analysis showed that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

> "The staff concludes that the plant design is acceptable with regard to a possible seizure of a rotor or break of a shaft of a reactor coolant pump."

V. REFERENCES

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

2. F.M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

3. C.D. Morgan, H.C. Cheatwood, and J.R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

*Recirculation pump shaft for a BWR.

4. R.H. Stoudt and J.E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. R. Linford, "Analytical Methods of Transient Evaluations in the General Electric Boiling Water Reactor," NEDP-10802, General Electric Company, April 1973.

7. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

8. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; a single rod, with current control modes, for a boiling water reactor) from a subcritical or low power (e.g., startup range) condition are evaluated by RSB. CPB reviews the reactivity coefficients and control rod worths under Standard Review Plan 4.3. The review under this plan covers the description of the causes of the transient and the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transient as compared with the acceptance criteria.

II. ACCEPTANCE CRITERIA

1. The following general design criteria (Ref. 1) apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. Examples of limits used previously to satisfy this criterion are:

      (1) In boiling water reactors (BWR's), the minimum critical heat flux ratio (MCHFR) calculated with the Hench-Levy correlation (Ref. 2) should exceed 1.0 at all times. The value of the minimum critical power ratio (CPR) calculated with the GETAB analysis (Ref. 3) will vary for different plants and product lines. Typically, the value will exceed 1.06.

(2) In pressurized water reactors (PWR's), the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.30 using the W-3 correlation (Ref. 4), or less than 1.32, using the B&W-2 correlation (Ref. 5).

If the application under review does not use one of these limits, then the correlation used by the applicant is reviewed by RSB under Standard Review Plan 4.4 and a criterion for critical heat flux is established that is acceptable here.

b. Fuel temperature and clad strain limits consistent with the acceptance criteria of Standard Review Plan (SRP) 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate (usually expressed in kW/ft). Examples of this criterion are:

(1) For BWR's, a linear heat generation rate of 24-28 kW/ft, which would result in limited UO_2 melting, but is not sufficient to cause 1% clad strain and potential clad failure.

(2) For PWR's, a linear heat generation rate of 20-22 kW/ft, which would result in a centerline fuel temperature equal to or less than the melting point of UO_2.

The specific value of the linear heat generation rate for this criterion is established during each review in a manner consistent with the acceptance criteria of SRP 4.2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

The procedures below are used for both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety settings included in the proposed technical specifications.

The reviewer, in determining whether the acceptance criteria are met, considers the following:

1. Peak conditions for the transient are maximized by low initial power; thus, the power level of the reactor should be at the lowest possible value compatible with the control rod configuration used for the accident. The postulated minimum core pressure and maximum temperature (i.e., the extremes of postulated conditions) should be consistent with the rod and power configuration to give minimum DNBR or MCHFR conditions.

2. Peak conditions for the transient are maximized by large reactivity addition rates near prompt critical; thus, the control rod configurations for the assumed withdrawal must be examined to confirm that such a maximized state has been included in the calculations. For a PWR, control bank withdrawal should be used. For a BWR, with the present control rod withdrawal procedures, a single rod of maximum worth available

in a normal configuration should be used. In many cases this will be a rod at the 50 percent rod density configuration. (More recent modes of BWR control such as group withdrawal may require that other configurations be examined.) The withdrawal rate should be the maximum available to the system.

3. The exact analysis of the transient would normally involve a three-dimensional, coupled neutron kinetics-thermal hydraulics calculation. However, acceptable results may be obtained with a neutron point-kinetics analysis and a coupled or separate hot fuel rod thermal analysis, if conservative input data are used. The reviewer determines whether the applicant's analytical methods are acceptable by using one or more of the following procedures:

   a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports and reports prepared in response to technical assistance requests (TARs).

   b. Perform a de novo review of the method (usually described in a separate licensing topical report, and frequently handled outside the scope of the review for a particular facility).

   c. Perform auditing-type calculations with methods available to the staff.

   d. Require additional, bounding calculations by the applicant to cover portions of the applicant's analytical methods that have not been fully reviewed or approved.

4. The input to the kinetics analysis model should be examined to assure that the input is appropriately conservative both for the state of the reactor and for the particular way it is used in the analysis. The power distribution or peaking factors used in the kinetics and hot pin thermal calculations must provide a conservative representation of the control rod configuration under consideration. The Doppler feedback coefficient should be related conservatively to the values accepted in the review under SRP 4.3, considering the time in cycle and temperature conditions of the fuel. The use of beginning of lifetime (BOL) Doppler coefficients is the most conservative; they should be used unless other values are specifically justified. Non-weighting of the coefficients is conservative, but weighting factors for the particular flux distribution shapes involved in the transients may be used if fully explored and justified. The moderator coefficients used should also be conservatively related to the values accepted in the review under SRP 4.3. The most positive or least negative values should be used and for a PWR this occurs at BOL. If the coefficient is negative, it may be conservatively taken as zero, as is generally done in BWR analyses.

5. The analysis should consider the relationships between the particular spatial flux shapes for the transient and the nuclear instrument response to assure that scrams occur at the times used in the analysis, that valid scram power levels are assumed, and that conservative scram delays and reactivity functions are used.

6. The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, reactor temperatures and pressures, maximum heat flux levels, and the related fuel temperatures, DNBR or MCHFR, and maximum clad strain. The latter are compared to the acceptance criteria.

IV. EVALUATION FINDINGS

If the staff, on completion of the review finds the applicant's analysis acceptable, conclusions of the following type should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods under low power startup conditions have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worths, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized).

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert criteria from SRP 4.2 and 4.4). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from a subcritical or low power condition have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such control rod assembly accidents."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants".

2. J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186 (proprietary) and APED-5286 (nonproprietary), General Electric Company (1966).

3. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

5. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969).

6. Standard Review Plan 4.2, "Fuel System Design."

7. Standard Review Plan 4.3, "Nuclear Design."

8. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; a single rod, with current control modes, for a boiling water reactor) at power are evaluated by RSB. CPB reviews the reactivity coefficients and control rod assembly worths involved under Standard Review Plan (SRP) 4.3. The review under this plan covers the description of the causes of the transient and of the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transients as compared with the acceptance criteria.

II. ACCEPTANCE CRITERIA

1. The following general design criteria (Ref. 1) apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. Examples of limits used previously to satisfy this criterion are:

      (1) In boiling water reactors (BWR's), the minimum critical heat flux ratio (MCHFR) should not be less than 1.0 using the Hench-Levy correlation (Ref. 2), or typically the minimum critical power ratio (MCPR) should not be less than 1.06 using GETAB analysis (Ref. 3).

(2) In pressurized water reactors (PWR's), the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.3, using the W-3 correlation (Ref.4), or less than 1.32, using the B&W-2 correlation (Ref. 5). The use of these correlations is limited to 14 x 14 and 15 x 15 rod arrays.

If the application under review does not use one of these limits, then the correlation used by the applicant is reviewed by RSB under SRP 4.4, and a criterion for critical heat flux is established that is acceptable here.

b. Fuel temperature and fuel clad strain limits consistent with the acceptance criteria of SRP 4.2 (Ref. 6) should not be exceeded. For steady-state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate (usually expressed in kW/ft). Examples of this criterion are:

(1) For BWR's, a linear heat generation rate of 24-28 kW/ft, which would result in limited UO_2 melting, but is not sufficient to cause 1% clad strain and potential clad failure.

(2) For PWR's, a linear heat generation rate of 20-22 kW/ft, which would result in a centerline fuel temperature equal to or less than the melting point of UO_2.

The specific value of linear heat generation rate for this criterion is established during each review in a manner consistent with the acceptance criteria of SRP 4.2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

1. The review process and the areas examined differ somewhat depending on whether a BWR or PWR is being reviewed. For both systems, the review covers the entire power range from low to full power and the allowed extreme range of reactor conditions during the operating (fuel) cycle including rod configurations, power distribution, and associated reactivity feedback components. The continuous withdrawal of normal configurations of rods should be assumed for the initial conditions in the transient calculation. For a PWR, this is one or two control banks; for a BWR, with current modes of control, it is a single control rod (future modifications under consideration may change this to group movement). The review covers a full range of rod or bank withdrawals up to maximum rod or bank worths and rates of reactivity addition.

   The exact analysis of the transient would normally involve a three-dimensional, coupled neutron kinetics, thermal-hydraulics calculation. However, acceptable results may be obtained with suitable approximate calculations. The problem examined and the approximations used differ for a PWR and a BWR.

2. For a BWR, past analyses and reviews have shown that at maximum rod worths and rates of reactivity addition, the reactor power increases slowly and the total increase is

is relatively small, so that the transient may be approximated by steady-state analyses. Because of changes in local power distribution attributable to rod motion and strong void feedback effects on the power distribution, three-dimensional, steady-state, coupled neutron distribution, thermal-hydraulics calculations that take account of these effects are required. The transient is halted by action of the rod block monitor (RBM) system, which should block rod withdrawal before fuel safety limits are reached.

The review process for a BWR, while recognizing the inherent transient nature of the problem, is concentrated on the steady-state aspects of the transient to assure that initial and subsequent power distributions are maximized, that the reactor conditions produce minimum CHFR, and that the response of the RBM system is conservatively calculated considering minimum operation of the associated local power range monitoring system.

3. A PWR analysis, on the other hand, generally involves larger power changes and requires transient calculations. Because power distributions in the course of the transient can frequently be predicted conservatively using design-limit peaking factors, point kinetics may be used for the nuclear transient. The nuclear transient is coupled, however, to core and system thermal-hydraulic response to the power changes (fuel and moderator thermal feedback and system instrumentation response).

   For a PWR, the reviewer ascertains that a full range of transient conditions are explored, that the transient calculation models are adequate, and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated. The range of parameters to be considered includes:

   a. Initial power levels from low to full power.

   b. Reactivity insertion rates from very low to maximum possible for the control system, including allowance for uncertainties.

   c. Fuel and moderator feedback reactivity coefficients covering the range expected throughout the cycle, including allowance for uncertainties.

   d. Power peaking factors at design limits for the initial power level conditions.

4. For both types of reactors, the reviewer determines whether the applicant's analytical methods and models are acceptable, including steady-state, transient, system response, and fuel response models. This may be done by using one or more of the following procedures:

   a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports (SER's) and reports prepared in response to technical assistance requests (TAR's).

b. Perform a de novo review of the method (usually described in a separate licensing topical report and frequently handled outside the scope of the review for a particular facility).

c. Perform auditing-type calculations with methods available to the staff.

d. Require additional, bounding calculations by the applicant to cover portions of the applicant's analytical methods that are not fully reviewed or approved.

5. The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod; scram or rod block actions that occur; reactor temperatures and pressures; maximum heat flux levels; and the related fuel temperatures, DNBR or MCHFR, and maximum clad strain. The latter are compared to the acceptance criteria.

IV. EVALUATION FINDINGS

If the staff, on completion of the review finds the applicant's analysis acceptable, conclusions of the following type should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transient and the instrumentation response to the transient. The methods used to determine the peak fuel rod response, and the input that analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized.)

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert acceptance criteria from SRP's 4.2 and 4.4, as appropriate). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single error control rod malfunctions have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both input assumptions and models to assure that fuel damage will not result from control rod withdrawal transients."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5286, General Electric Company, July 1966.

3. "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

5. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969).

6. Standard Review Plans 4.2, "Fuel System Design," and 4.4, "Thermal and Hydraulic Design."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

CPB reviews the following:

1. The types of control rod misoperations that are assumed to occur. For a pressurized water reactor (PWR), this may include one or more rods moving or displaced from normal or allowed control bank positions (such as dropped rods and rods left behind when inserting or withdrawing banks, or single rod withdrawal) and may include the automatic control system attempting to maintain full power. For a boiling water reactor (BWR) with current modes of control rod operation, limiting anomalies have been reviewed in Standard Review Plans (SRP) 15.4.1 and 15.4.2, and no additional areas are considered here.

2. Descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations.

   EICSB in SRP 7.2 and 7.7 reviews those safety systems required to prevent misoperations, as required by General Design Criterion 25 (Ref. 1), as well as the control rod system. The purpose of the review is to determine what events are to be included as single error malfunctions (e.g., single rod withdrawal).

3. The course of transients involved, e.g., rod drop followed by automatic return to full power with possible power overshoot.

4. Descriptions of the calculational models used and justification of their validity and adequacy.

5. The input to the calculations, including rod worths, power distributions, and feedback coefficients, and evidence of the conservatism of the input.

6. Descriptions of the sequence of events occurring during each transient, including the effects of important feedback mechanisms and trips.

7. Results of the analyses including, for each of the transients considered, plots of the time history of reactor power, reactor vessel pressure, critical heat flux for the limiting fuel rod, and maximum fuel centerline temperature.

II. ACCEPTANCE CRITERIA

1. The following general design criteria (Ref. 1) apply:

   a. Criterion 25, which requires that the reactor protection system be designed to assure that specified fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. An example of limits used previously to satisfy this criterion is that the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.3, using the W-3 correlation (Ref. 2), or less than 1.32, using the B&W-2 correlation (Ref. 3). If the application under review does not use one of these limits, then the reviewer must assure that an acceptable correlation is established under SRP 4.4 and is used here.

   b. Fuel temperature and fuel clad strain limits consistent with the acceptance criteria of SRP 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be referred to a linear heat generation rate (usually expressed in kW/ft). An example of this criterion is a linear heat generation rate of 20 to 22 kw/ft, which would result in a centerline fuel temperature equal or less than the melting point of UO_2. The specific value of linear heat generation rate for this criterion is established in a manner consistent with the acceptance criteria of SRP 4.2 (Ref. 4). For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. REVIEW PROCEDURES

The reviewer, in determining whether the criteria are met, must determine the transients that should be considered for this event. Generally, the list of errors should include: inadvertently withdrawing one or several rods; leaving one or several rods behind during bank withdrawal; and inserting one or several rods with power compensation in other portions of the core. In addition to these events, the reviewer must also decide, by postulating single failures in equipment or errors in operation, whether additional single rod malfunctions can be created. Once the list of transients has been established, the reviewer must determine acceptability in accordance with the criteria of Section II of this SRP.

1. For each failure event, the limiting (i.e., worst) fuel rod condition are determined by consideration of sensitivity calculations.

2. For each event, the analytical methods used by the applicant are reviewed. Those steady-state and transient methods that are primarily based on reactor physics considerations are the responsibility of CPB. Where thermal-hydraulic methods are involved, assistance of RSB may be required. In either case, the reviewer must determine whether the applicant's evaluation methods are acceptable. This may be done by using one or more of the following procedures:

    a. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports (SER's) and reports prepared in response to specific technical assistance requests (TAR's).

    b. Perform a _de novo_ review of the method (usually described in a separate licensing topical report, and often handled outside the scope of the review for a particular facility).

    c. Perform auditing-type calculations with methods available to the staff.

    d. Require additional bounding calculations by the applicant to confirm the validity of those portions of the applicant's analytical method that have not already been fully reviewed and approved.

3. For each event, the results are evaluated. In addition to verifying conformance to the acceptance criteria of Section II above, the reviewer determines that:

    a. Input conditions (e.g., pressure, temperature, flow rate) are at the adverse end of the range of values specified as the operating range.

    b. Initial power is 102% of licensed core thermal power, unless a lower power level is justified by the applicant.

    c. Output signals (power, temperature, flux perturbation) provided adequate alarm or scram signals.

    d. Nuclear conditions that interact with this event (e.g., Doppler coefficient, void coefficient) have been calculated as described in SRP 4.3.

IV. EVALUATION FINDINGS

If the reviewer's evaluation shows that the applicant's analyses are acceptable, the following kinds of statements should be included in the staff's safety evaluation report:

"The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects due to moderator and fuel temperature changes, have been examined. (If check calculations have been done, they should be summarized).

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling (DNB) have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert criteria from SRP 4.2 and 4.4). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that maximum configurations and transients for single error control rod malfunctions have been analyzed, that the analysis methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, in both input assumptions and models, to assure that fuel damage will not result from control rod malposition transients."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

3. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969).

4. Standard Review Plan 4.2, "Fuel System Design."

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SECTION 15.4.4 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW
15.4.5 CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch

Secondary - Electrical, Instrumentation and Control Systems Branch (EICSB)
Core Performance Branch (CPB)

I. AREAS OF REVIEW

A number of transients that may occur with moderate frequency cause either increased core flow or introduction of cooler or de-borated water into the core. These transients result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction. This review plan is intended to be applicable to all such transients. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref. 1).

The specific transients (Table 15-1 of Ref. 1) evaluated are:

1. Boiling water reactor (BWR): startup of an idle recirculation pump.

2. BWR: flow controller malfunction causing increased recirculation flow.

3. Pressurized water reactor (PWR) with loop isolation valves; startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which the isolation valves open.

4. PWR without loop isolation valves: startup of a pump in an inactive loop.

The review of the core flow increase transients considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests CPB to initiate a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transients are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The basic objectives of the review of the transients described above are:

   a. To identify which of the transients are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier, other than the fuel cladding, to the release of radioactive materials. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of the most limiting transients should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, they are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:

   a. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty. An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two, or three loops initially operating) or the effects referenced to a limiting case.

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analyses, and the reviewer should compare these to the limiting safety settings included in the proposed technical specifications.

The description of the core flow increase transients presented in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that a particular core flow increase transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR of the increase of flow transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7 and 8 of the SAR. The reviewer considers and evaluates the possibility of a single failure that would permit the loop isolation valves to open prior to startup of a pump in an idle loop (for those plants with loop isolation valves). If this could occur, the core flow rate increase would not be limited by the rate at which the valve opens, and the resulting rate of reactivity insertion could be greater than for other transients of this group.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that the selected core burnup yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the maximum pressure in the reactor coolant and main steam systems. For each transient the variations with time during the transient of the core and barrier performance parameters listed in the "Event Evaluation" section of Chapter 15 of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the core flow increase transients are compared to those predicted for other similar plants to see that they are within the range expected.

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 (SER):

"A number of plant transients can result in a core flow increase. Those that might be expected to occur with moderate frequency are the startup of an idle recirculation pump (BWR); flow controller malfunction causing increasing core flow (BWR); startup of a pump in an inactive reactor coolant loop (PWR); and startup of a pump in an initially isolated inactive reactor coolant loop.* All these postulated transients have been reviewed. It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _________ transient. This transient was evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (MDNBR)** did not decrease below _________ and that the maximum pressure within the reactor coolant and main steam system pressures did not exceed 110% of the design values.

"The staff concludes that the plant design is acceptable in regard to transients that result in an increase in coolant flow through the reactor core."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

4. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

5. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

6. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*The SER should present one statement for all similar transients.
**The minimum critical heat flux ratio or critical power ratio (MCHFR or MCPR) for a BWR.

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SECTION 15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

Unborated water can be added to the reactor coolant system, via the chemical volume and control system (CVCS), to increase core reactivity. This may happen inadvertently, because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated. Since the sequences of events that may occur depend on plant conditions at the time of the unplanned moderator dilution, the review includes conditions at the time of the unplanned dilution, such as refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown.

The review of postulated moderator dilution events considers causes, initiating events, the sequence of events, the analytical model, the values of parameters used in the analytical model, and predicted consequences of the event.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both the RSB and EISCSB. The RSB reviewer concentrates on the need for the reactor protection system and the operator action required to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests CPB to initiate a generic evaluation of the new analytical model. APCSB reviews the functional and operational characteristics and potential failure modes of the CVCS, as described in Standard Review Plan (SRP) 9.3.4. The RSB reviewer makes use of this review to evaluate initiating causes and the expected sequence of events.

The predicted results of moderator dilution events are reviewed to assure that the consequences meet the acceptance criteria given in Section II of this SRP. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objective of the review of moderator dilution events is to confirm that either of the following conditions are met:

   a. The consequences of these events are less severe than the consequences of another transient that results in an uncontrolled increase in reactivity and has the same anticipated frequency classification.

   b. The plant responds to the events in such a way that the criteria regarding fuel damage and system pressure are met and adequate time is allowed the operator to terminate the dilution before the shutdown margin is eliminated.

2. The specific criteria for moderator dilution events are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 1).

   b. Fuel clad integrity should be maintained by ensuring that Section II, Acceptance Criterion 1 of SRP 4.4 is satisfied throughout the transient.

   c. Such incidents will not generate a more serious plant condition without other faults occurring independently.

   d. From the time an alarm makes the operator aware of unplanned moderator dilution, the following minimum time intervals must be available before a loss of shutdown margin occurs:

      (1) During refueling: 30 minutes.

      (2) During startup, cold shutdown, hot standby, and power operation: 15 minutes.

3. The applicant's analysis of moderator dilution events should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 2 through 4 are acceptable. Should other analytical methods be proposed, these methods must be evaluated by the staff. For new generic methods, the reviewer requests an evaluation by CPB.

   All of the following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown.

   The parameters and assumptions used in the analytical model should be suitably conservative. The following values and assumptions are considered acceptable:

   a. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2% to account for power measurement uncertainty.

   b. The boron dilution is assumed to occur at the maximum possible rate.

   c. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. This will usually be the beginning-of-life (BOL) condition.

   d. All fuel assemblies are installed in the core.

   e. A conservatively low value is assumed for the reactor coolant volume.

   f. For analyses during refueling, all control rods are withdrawn from the core.

   g. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1%) is assumed to exist prior to the initiation of boron dilution.

   h. For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

   i. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

III. <u>REVIEW PROCEDURES</u>

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and set points used in the analysis will be preliminary in nature and subject to change. At the OL review, final values will be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The descriptions of moderator dilution transients presented in the SAR are reviewed by RSB regarding the occurrences leading to the initiating events. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function. Particularly important are the alarms which alert the operator to the unplanned boron dilution.

2. The extent to which the plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The RSB reviewer confirms that analyses are included for a boron dilution incident occurring during each of the following plant initial conditions: refueling, startup, power operation (automatic control and manual modes), hot standby, and cold shutdown. For each such incident reviewed, all possible causes must have been considered by the applicant and justification presented that the cause selected for analysis is the one that allows the operator the least time to take corrective action.

With the aid of the EICSB reviewer, the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of each boron dilution incident to acceptable levels is reviewed. The RSB reviewer compares the predicted variations of system parameters with various trip and system initiation set points. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation where the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components that may affect the course of the transient. This phase of the review uses the system review procedures described in the standard review plans for Chapters 5, 6, 7, 8, and 9 of the SAR. In particular, the redundancy of alarms that alert the operator to the unplanned dilution is confirmed.

The mathematical models used by the applicant to evaluate core performance and reactivity status are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant. The justification provided by the applicant to show that the selected core burnup condition, boron concentration, and rod worths yield the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis. These values are reviewed by CPB under SRP 4.2. The value of core reactivity as a function of time following each incident analyzed is confirmed by comparison with an acceptable analysis performed for another plant, by comparison with staff calculations for typical plants done by CPB on request, or by independent calculations by the RSB reviewer.

The assumed dilution flow rates are reviewed, taking into consideration the system parameters which act to limit the flow. The reviewer examines the flow-limiting equipment characteristics provided by the applicant to justify his flow rate assumptions; e.g., if the flow is limited by the charging pump capacity, the assumed flow is compared with the flow for all charging pumps acting at full capacity. If some lesser value of flow is assumed, such as not all pumps operating, or flow limited by a valve, justification must be provided. EICSB is consulted concerning any interlocks for which credit is taken.

The results of the analyses are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding the time available for the operator to take corrective action. The variations with time during the transient of important parameters are compared to those predicted for other similar plants to see that they are within the range expected. Parameters of particular importance are core reactivity, boron concentration, rate of addition of unborated water, power level, core pressure, and minimum departure from nucleate boiling ratio (DNBR).

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:

> "Various chemical and volume control system (CVCS) malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The malfunctions that allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling. These events were evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analyses of these events showed that the operator has ___ minutes to take corrective action if a boron dilution incident occurs during refueling and ___ minutes if at power. In the latter case,the most severe transient results in a minimum departure from nucleate boiling ratio (DNBR) of ____ and reactor coolant and main steam system pressures of less than 110% of design.

"Based on these results, the staff concludes that the plant design is acceptable with regard to boron dilution incidents caused by malfunctions of the chemical and volume control system."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

3. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

4. "Standard Nuclear Steam System B-SAR-241," Babcock & Wilcox Company, February 1974 (under review).

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

6. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

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SECTION 15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Analysis Branch (AAB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The review of fuel loading errors considers:

1. The spectrum of misloading events analyzed. A sufficient number of fuel loading errors must be studied by the applicant and presented to show that the worst situation undetectable by incore instrumentation has been identified. The kinds of errors considered should include loading of one or more fuel assemblies into improper locations and, where physically possible, with incorrect orientation. For those reactors in which burnable poison or fuel rods are added to or removed from assemblies at the plant, errors in these processes must be considered.

2. Changes in the power distribution and increased local power density.

3. The provisions made to search for loading errors at the beginning of each fuel cycle.

CPB also reviews the effect of misloaded fuel on nuclear design parameters, the detection of fuel loading errors, and any operational restrictions that would assist in staying within safety limits.

AAB reviews the radiological implications of misloaded fuel on request.

On request, QAB reviews the measures provided to minimize the probability of fuel misloading.

II. ACCEPTANCE CRITERIA

The acceptance criteria are based on General Design Criterion 13 (Ref. 1) and on 10 CFR Part 100 (Ref. 2). Criterion 13 is concerned with instrumentation and control. One intent of this criterion is to require monitoring of variables over anticipated ranges for normal operating and accident conditions. The consequences of misloaded fuel that cannot be detected are covered by 10 CFR Part 100, which is concerned with reactor site criteria.

The primary safeguards against fuel loading errors are procedures and design features to minimize the likelihood of the event. Additional safeguards include incore instrumentation systems which would detect errors. However, should an error be made and go undetected, it is possible in some reactor designs for fuel safety limits to be exceeded. Therefore, additional acceptance criteria are required to cover the event of operation with misloaded fuel.

The acceptance criteria for fuel loading errors are:

1. The plant technical specifications should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

2. In the event the error is not detectable by the instrumentation system and fuel safety limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

III. REVIEW PROCEDURES

The review procedures for fuel loading errors are as follows:

1. The reviewer verifies that the various cases of misloaded fuel assemblies outlined in Section I have been analyzed by the applicant and the worst case determined. For each case the effect on the reactor power distribution should be given.

2. The reviewer determines that the effect each postulated error has on reactor instrumentation has been ascertained. For limiting events (where safety limits are exceeded) the reviewer verifies that acceptable techniques (Ref. 4) have been used to calculate the fuel temperature conditions.

3. The reviewer assures compliance with Criterion 1 of Section II above by reviewing the plant technical specifications to verify that they contain provisions requiring that incore instrumentation be used to search for misloaded fuel after each fueling operation. Since low-power mapping is typically done, searching for misloading can be accomplished by the usual low-power maps.

4. When it is determined that fuel limits can be exceeded, AAB is requested to perform dose calculations to assure that Criterion 2 of Section II above is met.

IV. EVALUATION FINDINGS

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

"The staff has evaluated the consequences of a spectrum of postulated fuel loading errors. We conclude that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the offsite consequences of any core damage are a small fraction of 10 CFR Part 100 guidelines. The applicant affirms that the available incore instrumentation will be used before the start of a fuel cycle to search for fuel loading errors."

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 4.2, "Fuel System Design."

4. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

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SECTION 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Reactor Systems Branch (RSB)
Accident Analysis Branch (AAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

CPB evaluates the consequences of a control rod ejection accident in the area of physics. RSB, under Standard Review Plan (SRP) 4.4, reviews the relevant thermal-hydraulic analyses. The CPB review covers the possible initial conditions, rod patterns and worths, scram worth as a function of time, adequacy of the various reactivity coefficients, adequacy of the calculational methods, and any core parameters which affect the peak reactor pressure or the probability of fuel pin failure.

AAB reviews the radiological consequences of a rod ejection accident by using a source term for dose calculations based on the amount of failed fuel as obtained by CPB from the physics and thermal-hydraulic analyses.

EICSB in SRP 7.2 and 7.3 reviews the applicant's determination of the reactor trip delay time, i.e., the time elapsed between the instant the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion.

II. ACCEPTANCE CRITERIA

Regulatory Guide 1.77 (Ref. 1) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident. Two criteria are used by CPB in evaluating the rod ejection accident:

1. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the emergency condition stress limits as defined in the ASME Code (Ref. 2).

The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation by AAB. The radiological criteria used in the evaluation of rod ejection accidents (PWR's) are given in Appendix B of Regulatory Guide 1.77 (Ref. 1).

III. REVIEW PROCEDURES

1. Review of the applicant's analyses, showing that the first of the acceptance criteria above is met, proceeds as follows:

   a. A spectrum of initial conditions is considered, which must include both zero-power and full-power conditions, at beginning and end of fuel lifetime (BOL and EOL), to assure examination of upper bounds on possible fuel damage.

   b. From the initial conditions of (a) and from control rod patterns (Ref. 3) the limiting rod worth is determined. Where confirmation is considered necessary the reviewer may calculate, as an audit, the worth of limiting rods.

   c. Reactivity coefficient values corresponding to the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed under SRP 4.3 (Ref. 3). The two coefficients of most interest are the Doppler and moderator coefficients. If no three-dimensional space-time calculation is performed, the reactivity feedback must be conservatively weighted to account for the variation in the missing dimension.

   d. The reviewer inspects the control rod insertion assumptions which include: trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are covered under SRP 7.2 by EICSB. Rod worth is checked by the reviewer for consistency with SRP 4.3.

   e. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work, if the analytical methods have been previously reviewed and approved by the staff. Otherwise he must perform a complete review on this case. Alternatively an audit of several calculations, using methods considered acceptable to the staff, may be done by the reviewer (or consultants to the staff). The primary concern of the reviewer is how well the analytical model elements represent the true three-dimensional problem. Other items checked by the reviewer include feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

   f. Results of the calculations done by procedures described in steps a-e are expressed as values of the radially-averaged fuel rod enthalpy (in units of cal/gm). The reviewer determines that the maximum value does not exceed 280 cal/gm.

2. Verification of compliance with the second acceptance criterion is accomplished as follows:

   a. The same procedures considered in steps a-f above are followed.

   b. For each accident, the transient primary system pressure should have been calculated by an analytical method acceptable to the staff or, as before, an independent audit calculation is made by the staff. The reviewer checks the results (as obtained by the applicant or the staff) for compliance with the second criterion.

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedure:

   a. The reviewer determines that an acceptable procedure for calculating a departure from nucleate boiling condition during the reactivity excursion has been used. This may be done by referring to previous cases for the same nuclear steam supply system (NSSS) vendor. If no approved technique is available, as might be the case for the first project using a new or substantially revised model, the reviewer must perform a separate detailed review (which is usually documented separately in a topical report).

   b. The reviewer must determine that the number of rods used in the radiological evaluation is the number of rods calculated to have a departure from nucleate boiling ratio (DNBR) less than 1.30 when a DNB correlation such as W-3 (Ref. 4) is used, or 1.32 when a DNB correlation such as B&W-2 (Ref. 5) is used.

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 staff has evaluated the applicant's analysis of the assumed rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below the emergency condition stress limit (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

"The consequences of the rod ejection accident have been evaluated, and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the recommendation of Regulatory Guide 1.77, the computer code ___________, and a conservative description of the plant response to the accident. The calculated doses are presented in Table ___________. Technical specification limits on primary-secondary coolant leakage assure that the potential doses are well within 10 CFR Part 100 exposure guidelines."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. Standard Review Plan 4.3, "Nuclear Design."

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

5. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969).

APPENDIX

STANDARD REVIEW PLAN 15.4.8

RADIOLOGICAL CONSEQUENCES OF
CONTROL ROD EJECTION ACCIDENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Core Performance Branch (CPB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The AAB review under this appendix covers the following areas:

1. The plant response to a control rod ejection accident.

2. The calculation of whole body and thyroid doses at the exclusion area boundary and low population zone outer boundary due to the releases resulting from a rod ejection accident.

The purposes of the review are to assure that the plant procedures for recovery from a rod ejection accident and the plant technical specifications are properly taken into account in computing the whole body and thyroid doses at the nearest exclusion area boundary and low population zone (LPZ) outer boundary, and to compare the calculated doses against the appropriate guidelines.

The physics and thermal-hydraulic aspects of the accident are reviewed by CPB. Verification of the applicant's calculations of the number of fuel pins experiencing departure from nucleate boiling (DNB) and the amount of fuel reaching the clad melting temperature is obtained from the CPB.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a rod ejection accident at the construction permit stage, and the primary-secondary system leakage appropriately limited, if calculations show that the resulting doses at the nearest exclusion area boundary are on the order of 150 rem to the thyroid and 20 rem to the whole body, or less, for the first two hours after the accident, and 150 rem thyroid and 20 rem whole body, or less, for the course of accident at the LPZ outer boundary. Higher doses may be acceptable at the operating license (OL) review stage, up to the guidelines of 10 CFR Part 100. Technical specifications should be set to assure that the doses resulting from a rod ejection accident are limited to the guideline values.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix as may be appropriate for a particular case. The judgment on areas 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 detailed review of the radiological consequences of a rod ejection accident is done at OL stage when system parameters and accident analysis results are fully developed. At the CP stage, the reviewer estimates the doses from the rod ejection accident based on results from similar plants that have been recently reviewed.

The AAB review of the rod ejection accident at the OL stage covers the following topics:

1. Release of the radioisotopes to the environment via the containment building.

2. Release of radioisotopes to the environment through the secondary system.

3. Calculation of resulting doses.

Physical plant parameters, such as the steam generator steaming rates, are reviewed to ascertain their conservatism.

Regulatory Guide 1.77 (Ref. 2) should be used as a guide in the analysis of the accident. The release of radioisotopes through the secondary system should be analyzed independently by means of a digital computer code. Computer codes are currently under development within NRC. Documentation will be published in a NUREG report. In the analysis of this accident, a loss of offsite power is assumed. It is also assumed that nuclides released to the primary coolant due to any fuel failures or melting (this information is obtained from the CPB) are instantaneously and uniformly mixed in the coolant at the time of the accident. For releases via the containment building, Regulatory Guide 1.77 recommends that 100% of the noble gases and 25% of the iodines contained in the fuel which is estimated to reach initiation of melting be available for release from the containment. For releases through the secondary system, 100% of the noble gases and 50% of the iodines contained in the fuel which is estimated to reach initiation of melting are assumed to be released to the primary coolant.

The SAB provides the reviewer with the distance to the nearest boundary of the exclusion area, the accident (5 percentile) wind speed and X/Q, and the 0-8 hr and 8-24 hr X/Q values at the outer boundary of the LPZ. These X/Q values are used to estimate the consequences of releases from the containment and the consequences at the LPZ outer boundary of releases from the secondary system. The X/Q value for calculating the two-hour dose at the nearest exclusion area boundary from the releases through the secondary system is obtained from Regulatory Guide 1.5 (Ref. 3) and corrected for wind speeds differing from 1 m/sec (inverse ratio). A breathing rate of 3.47×10^{-4} m^3/sec is used in calculating the thyroid doses for the first 8 hours after the accident; from 8 to 24 hours, a breathing rate of 1.75×10^{-4} m^3/sec is used.

Although the resulting doses in case of an actual accident would be a composite of the doses computed for releases via the containment building and through the secondary system, both doses should be presented. If either dose approaches the limit, calculation of representative composite cases should be considered (the AAB branch chief should be consulted).

If the doses resulting from the releases through the secondary system exceed the limits specified in Section II above, the technical specification limit on primary-secondary system leakage is reduced accordingly. If the doses resulting from the potential releases from the primary containment exceed the specified limits, the pressure setpoint for actuation of the containment sprays may have to be reduced to obtain credit for spray removal of the fission products.

The physics and thermal-hydraulic aspects of the accident are reviewed by the CPB. Verification of the applicant's calculations of the number of fuel pins reaching DNB and the amount of fuel reaching the fuel melting temperature are obtained (and documented by buckslip) from the CPB. It is important to note that the fuel melting temperature criterion used for release of large fractions of fission gases corresponds to the initiation of melting as opposed to the 280 cal/gm used as a criterion by the CPB for core disruption.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions such as the following, to be included with the CPB findings in the staff's safety evaluation report at the operating license stage:

> "The consequences of the rod ejection accident have been evaluated, and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the recommendations of Regulatory Guide 1.77, the computer code ________, and a conservative description of the plant response to the accident. The calculated doses are presented in Table ____________. Technical specification limits on primary-secondary coolant leakage assure that the potential doses are well within 10 CFR Part 100 exposure guidelines."

At the construction permit stage the following paragraph is included with the CPB findings in the staff's safety evaluation report:

> "On the basis of our experience with the evaluation of the control rod ejection accident for PWR plants, we have estiamted the doses from this accident to be on the order of ___________rem to the thyroid and ___________rem to the whole body at the nearest boundary of the exclusion area. If a reevaluation of this accident at the operating license stage results in dose estimates that exceed the 10 CFR Part 100 guidelines, appropriate limits on primary-secondary leakage and the setpoint for containment spray actuation will be set."

V. REFERENCES

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

2. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection 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."

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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Reactor Systems Branch (RSB)
Accident Analysis Branch (AAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

CPB evaluates the consequences of a rod drop accident in a boiling water reactor (BWR) in the area of physics. The CPB review covers the applicant's description of the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, and methods used to analyze the accident. At the present time, CPB has analytical methods available for verifying rod worths for some events and for calculating energy deposition for a given rod drop. Techniques are under development for calculating the critical heat flux ratio (dryout) for a rod drop at rated power. A general reference on rod drop accident analysis is noted in Section V (Ref. 1). RSB, under Standard Review Plan (SRP) 4.4, reviews the thermal and hydraulic analyses.

AAB, as described in the appendix to this plan, reviews the radiological consequences of a rod drop accident, using the amount of failed fuel as obtained by CPB from physics and thermal-hydraulics calculations as the source for dose calculations.

EICSB, in SRP 7.2 and 7.3, reviews the reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion.

II. ACCEPTANCE CRITERIA

CPB focuses on two limiting criteria in evaluating the rod drop accidents:

1. Reactivity excursions should not result in radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the emergency condition stress limits as defined in the ASME Code (Ref. 2).

The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout (i.e. MCHFR less than 1.0) for rated power initial conditions.

III. REVIEW PROCEDURES

1. Review of the applicant's analyses showing compliance with the first of the above criteria is carried out as follows:

   a. The reviewer verifies that the applicant has considered a spectrum of initial conditions for this event that covers the range of time-in-cycle and initial power levels.

   b. The reviewer verifies that the maximum expected individual rod worths are used. In developing rod worth criteria, the nominal control rod withdrawal pattern must be considered, as well as those abnormal patterns that are not precluded by an instrumentation system acceptable to EICSB.

   c. The reviewer determines that an acceptable and conservative single control rod worth is used (Ref. 3) and that the function used to describe the control rod position as a function of time is acceptable to RSB.

   d. The reviewer determines that conservative reactivity coefficients, notably the Doppler coefficient, are used and that they are compatible with those described in SRP 4.3 (Ref. 3).

   e. The reviewer assures that the scram action is conservatively represented in the use of the integral scram worth curve (SRP 4.3) and in the use of the scram delay time.

   f. The reviewer checks the analytical methods or assures that they have been reviewed and approved previously. The reviewer may also perform an independent audit calculation using methods acceptable to the staff. The applicant's methods should account conservatively for all major reactivity feedback mechanisms.

2. The reviewer inspects the results of the calculation of maximum reactor pressure to determine compliance with the second criterion listed in Section II, (the reviewer may do an audit calculation when appropriate).

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedure:

   a. The reviewer determines that the transient critical heat flux (CHF) has been computed by an acceptable technique (either previously reviewed or reviewed de nova during this review).

   b. The reviewer determines that the number of rods with enthalpy exceeding 170 cal/gm has been computed by an acceptable method.

   c. The reviewer determines that the amount of fuel exceeding melting conditions has been computed by an acceptable method.

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 staff has evaluated the applicant's analysis of the assumed rod drop accident and finds the assumptions, calculational techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below the emergency condition stress limit (as defined in Section III of the ASME Boiler and Pressure Vessel Code) for the maximum rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained."

"The consequences of the rod drop accident have been evaluated and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the computer code ________, and a conservative description of the plant response to the accidents. The calculated doses are presented in Table ______, and are well within the 10 CFR Part 100 exposure guidelines."

V. REFERENCES

1. "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, General Electric Company, March 1972.

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. Standard Review Plan 4.3, "Nuclear Design."

4. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

APPENDIX

STANDARD REVIEW PLAN 15.4.9

RADIOLOGICAL CONSEQUENCES OF
CONTROL ROD DROP ACCIDENT (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Core Performance Branch (CPB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The AAB review under this appendix covers the following areas:

1. The plant response to a control rod drop accident.

2. The calculation of whole body and thyroid doses at the nearest exclusion area boundary and low population zone outer boundary due to the releases resulting from a rod drop.

The purposes of the review are to assure that the plant procedures for recovery from a rod drop accident are properly taken into account in computing the whole body and thyroid doses at the nearest exclusion area boundary and low population zone (LPZ) outer boundary resulting from the postulated release of radioactive gases following the accident. The calculated doses are compared with the 10 CFR Part 100 guideline values (Ref. 1).

The physics and thermal-hydraulic aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel pins experiencing departure from nucleate boiling (DNB) and the amount of fuel reaching the clad melting temperature is obtained from the CPB.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a control rod drop accident at the construction permit stage if calculations show that the resulting doses at the nearest exclusion area boundary are on the order of 150 rem to the thyroid and 20 rem to the whole body, or less, for the first two hours after the accident, and 150 rem thyroid and 20 rem whole body, or less, for the course of the accident at the LPZ outer boundary. Higher doses may be acceptable at the operating license review stage, provided these are within the 10 CFR Part 100 guidelines.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix 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 AAB review of the control rod drop accident covers the release of radioisotopes from the core to the environment via the turbine and condensers and the calculation of the resulting doses. Physical plant parameters, such as the leakage rate from the condenser, are reviewed to ascertain their conservatism.

The release of radioisotopes and the resulting doses are analyzed at the OL stage by means of a digital computer code (Ref. 2). In the analysis, it is assumed that the integrity of the turbine and condensers is unaffacted by the rod drop accident. However, a coincident loss of offsite power is assumed. It is also assumed that any nuclides released to the reactor coolant from fuel cladding failures or fuel melting (these data are obtained from the CPB) are instantaneously and uniformly mixed in the coolant at the time of the accident. The following additional assumptions are used in determining the radiological consequences of a control rod drop accident:

1. With respect to the release of radioactive material to the pressure vessel, the assumptions are:

    a. The combination of reactor operating mode, control rod positions, core burnup, etc., that results in the largest source term is selected for evaluation.

    b. The nuclide inventory in any fuel rods for which cladding performation or failure is assumed to occur is calculated and all gaseous constituents in the fuel-clad gaps are assumed released to the reactor coolant.

    c. The amount of activity accumulated in the fuel-clad gap is assumed to be 10% of the iodines and 10% of the noble gases of the core inventory estimated in item a above.

    d. No allowance is made for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

    e. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842°C) at any time during the course of the accident is calculated and 100% of the noble gases and 50% of the iodines contained in this fraction are assumed released to the reactor coolant.

2. With respect to the transport of radioactive material to the environment, the assumptions are:

    a. One tenth of the iodines and 100% of the noble gases released to the pressure vessel reach the turbine and condensers.

b. All noble gases remain in a gaseous state and are available for leakage from the turbine and condensers.

c. The quantity of airborne iodine available for leakage from the turbine and condensers, is reduced by a factor of 10 by partitioning and plateout in the turbine and condensers.

d. The turbine and condensers leak at a rate of 1.0%/day.

e. The effects of radiological decay during holdup in the turbine and condensers are taken into account.

The X/Q values (obtained from the SAB) and breathing rates are the same as those used in the calculation of doses from a loss-of-coolant accident (Ref. 3).

The physics and thermal-hydraulics aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel pins reaching 170 cal/gm and the amount of fuel reaching the fuel melting temperature are obtained from the CPB. It is important to note that the fuel melting temperature criterion used for release of large fractions of fission gases corresponds to the initiation of melting as opposed to the 280 cal/gm criterion used by the CPB in their evaluation of reactivity excursions.

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 with the CPB findings in the staff's safety evaluation report:

"The consequences of the rod drop accident have been evaluated and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the computer code ________, and a conservative description of the plant response to the accidents. The calculated doses are presented in Table ________, and are well within the 10 CFR Part 100 exposure guidelines."

V. REFERENCES

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

2. Computer codes are currently under development. Documentation will be published in a NUREG report.

3. Appendix A, Standard Review Plan 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident (Containment Leakage Contribution)."

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.5.1 INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME
15.5.2 CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

Various types of equipment malfunctions, operator errors, and abnormal occurrences that may occur with moderate frequency can cause an unplanned increase in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and lead to fuel damage or overpressurization of the reactor coolant system. Alternatively, a power level decrease and depressurization may result. In either case, if the transient is severe enough the reactor will trip from high water level, high flux, or high or low pressure. For pressurized water reactors (PWR's) operator action will probably be required to terminate the unwanted injection flow.

This review plan is intended to be applicable to these types of moderate frequency events that increase reactor coolant inventory. These transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Ref.1). The specific initiating events considered in this review plan are:

1. Boiling water reactors (BWR's) - Inadvertent operation of the high pressure core spray, high pressure coolant injection, or reactor core isolation cooling system.

2. PWR's - Inadvertent operation of high pressure emergency core cooling system (high pressure injection system) or a malfunction of the chemical and volume control system.

Other BWR transients that can result in an increase in reactor coolant inventory include feedwater system malfunctions (increasing flow), steam pressure regulator malfunctions (decreasing flow), loss of electrical load, turbine trip, main steam isolation valve (MSIV) closure, and loss of condenser vacuum. These transients are the subject of other standard review plans that consider their effects on system parameters other than coolant inventory. However, the impact of these transients on reactor coolant inventory is considered by the reviewer as a portion of the effort involved in this review plan.

The review of events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of those transients analyzed are reviewed to assure that the consequences meet the acceptance criteria given in Section II, below. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. <u>ACCEPTANCE CRITERIA</u>

1. The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:

   a. To identify which of the moderate frequency* events leading to a coolant inventory increase are the most limiting.

   b. To verify that, for the most limiting transients, the plant responds to the core flow increase in such a way that the criteria regarding fuel damage and system pressure are met.

*The term "moderate frequency" is used in this review plan in the same sense as in the descriptions of design and plant process conditions in References 8 and 9.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 2).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of Standard Review Plan (SRP) 4.4 (Ref. 7) is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 3 through 6 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

   The values of the parameters used in the analytical model are to be suitably conservative. The use of the following values is considered acceptable:

   a. The reactor is initially at rated power (licensed core thermal power) plus 2% for power measurement uncertainty.

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of core.

   c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's description of events leading to an increase in reactor coolant inventory is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The applicant should present a quantitative analysis in the SAR of the event leading to an increase in reactor coolant inventory which is the most limiting. For this event, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the event to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. In this phase of the review, the system reviews are performed as described in the standard review plans for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in Section II regarding maximum pressure in the reactor coolant and main steam systems and the minimum critical heat flux ratio (MCHFR) or departure from nucleate boiling ratio (DNBR). The variation with time during the transient of parameters listed in Sections 15.X.X.3(c) and 15.X.X.4(c) of the Standard Format (Ref. 1) are reviewed. The values of the more important of these parameters for the events leading to an increase in reactor coolant inventory are compared to those predicted for other similar plants to confirm that they are within the expected range.

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 (SER):

"A number of plant transients can result in an increase in reactor coolant inventory. Those that might be expected to occur with moderate frequency are inadvertent operation of the emergency core cooling system, chemical and volume control system malfunction, and various BWR transients.* All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ________________ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the ______________ transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable with regard to events that are expected to occur with moderate frequency and to result in an increase in reactor coolant inventory."

V. REFERENCES

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

3. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

*The SER draft should present one statement for all similar transients.
**Minimum critical heat flux ratio for a BWR.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.6.1 INADVERTENT OPENING OF A PWR PRESSURIZER SAFETY/RELIEF VALVE OR A BWR SAFETY/RELIEF VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

The inadvertent opening of a safety or relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. The effect of the pressure decrease is to decrease the neutron flux (via moderator density feedback). In a pressurized water reactor (PWR), a reactor trip occurs due to low reactor coolant system (RCS) pressure. In a boiling water reactor (BWR), the safety or relief valve discharges into the suppression pool. Normally there is no scram in a BWR. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCV) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell.

The review of these transients should consider the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition. The EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transient are reviewed to assure that the consequences meet the acceptance criteria given in Section II of this standard review plan (SRP). Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objective in the review of inadvertent primary safety or relief valve opening events is to confirm that one of the following criteria is met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of reactor coolant inventory and has the same anticipated frequency classification.

   b. The plant responds to the safety or relief valve opening transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. The specific criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures (Ref. 1).

   b. Fuel clad integrity should be maintained by ensuring that acceptance criterion 1 of SRP 4.4 (Ref. 7) is satisfied throughout the transient.

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, should not cause loss of function of any barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

3. The applicant's analysis of this transient should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 2 through 5 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by CPB.

The values of the parameters used in the analytical model are to be suitably conservative. The use of the following values is considered acceptable:

a. The reactor is initially at 102% of licensed core thermal power (2% allowance for power measurement uncertainty).

b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's description of the inadvertent safety or relief valve opening transient is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

If the SAR states that the inadvertent safety or relief valve opening transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the transient is presented in the SAR, the RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may alter the course of the transient. In this phase of the review, the system reviews are performed as described in the standard review plans for Chapters 5, 6, 7 and 8 of the SAR. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II regarding the maximum pressure in the reactor coolant and main steam systems. The variation with time during the transient of the core and barrier performance parameters listed in the Event Evaluation Section of Chapter 15 of the Standard Format (Ref. 6) are reviewed. Values of the more important of these parameters for the transient caused by inadvertent safety or relief valve opening are compared to those predicted for other similar plants to confirm that they are within the expected range.

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 (SER):

> "A number of plant transients can result in a decrease of reactor coolant inventory. Those that might be expected to occur with moderate frequency are safety or relief valve openings, minor primary pipe breaks, and (in BWR's) loss of feedwater.* All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ________ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis of the ________ transient showed that cladding integrity was maintained by ensuring that the minimum departure from

*The SER draft should present one statement for all similar transients.

nucleate boiling ratio (DNBR)* did not decrease below _____ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

"The staff concludes that the plant design is acceptable in regard to transients that are expected to occur with moderate frequency and result in a decreased primary coolant inventory."

V. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973 (under review).

3. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41" (under review).

4. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973 (under review).

5. "Standard Nuclear Steam System, B-SAR-241," Babcock and Wilcox Company, February 1974 (under review).

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

7. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

*Minimum critical heat flux ratio or critical power ratio (MCHFR or MCPR) for a BWR.

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SECTION 15.6.2 FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Containment Systems Branch (CSB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

This review plan covers the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary, such as instrument lines and sample lines. For the purpose of this review these lines are classified into two categories:

1. Those lines meeting the isolation requirements of General Design Criterion 55.

2. Lines exempted from General Design Criterion (GDC) 55, i.e., instrument lines.

Specifically, the AAB review covers the following areas for lines exempt from the GDC 55 isolation requirements:

1. The calculation of whole body and thyroid doses at the site boundary resulting from a failure of an instrument line exempt from the GDC 55 isolation criteria.

2. Any limitations on primary coolant radioactivity concentrations, as established in the technical specifications, required as a result of the analysis of instrument line failures.

For lines which meet the GDC 55 isolation requirements the AAB reviews the requirements, if any, on the isolation time and maximum leak rate of the isolation valves, which may be necessary to assure that the radiological consequences of a failure of these lines will not exceed a small fraction of 10 CFR Part 100 dose guidelines.

The RSB reviews the plant response to a failure of an instrument line, and will notify the AAB if this accident is predicted to cause fuel failures. In addition, the RSB may be requested by the AAB to confirm the value assumed for the mass of coolant released in the accident. The SAB provides the reviewer with the X/Q values to be used in the two-hour dose calculations. The CSB may be requested by the reviewer to verify that secondary containment integrity is maintained.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against the radiological consequences of a failure of containment-penetrating small line carrying reactor coolant (and the technical specifications for primary coolant activity and isolation time and maximum allowable leak rate of isolation valves in these lines appropriately limited) if calculations show that the resulting doses at the site boundary are small fractions of the 10 CFR Part 100 exposure guidelines.

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 detailed review of the radiological consequences of a failure of a small line carrying reactor coolant outside of the containment is done at the operating license (OL) stage, when system parameters and accident analyses are fully developed. At the construction permit stage the review is limited to a brief survey of the pertinent portions of the plant design and the applicant's discussion of these accidents to determine that there are no unusual features that would prevent limitation of radiological consequences to acceptable levels by appropriate limits on coolant activity concentrations or isolation valve closing times and leak rates.

The AAB review at the OL stage consists of the following steps:

1. Review of the applicant's descriptions of the accidents, in order to confirm the appropriateness and conservatism of the assumptions used in the analysis.

2. Performance of an independent analysis of the radiological consequences of the failure of the instrument or other line. At present, the following conservative assumptions are used to simplify the analysis:

    a. The mass of reactor coolant released during a two-hour period is estimated with the assumption of choked flow at a fluid enthalpy equal to that of the reactor coolant under normal operating conditions. (Assume this is correct unless otherwise notified by RSB.)

    b. The initial fission product concentrations in the primary coolant are those given as maximum equilibrium values in the technical specifications. The effects of an activity spike resulting from reactor shutdown or depressurization of the primary system is modeled by increasing the rate of activity release from the fuel by a factor of 500.

c. The fission products in the primary coolant are assumed to be released to the environment if the line carrying the primary coolant penetrates the secondary containment (if any). Otherwise, the fission products are assumed to be released to the secondary containment or auxiliary building, as appropriate. The CSB should verify the integrity of the secondary containment during the pressure transient associated with a failure within its boundary. An appropriate mixing volume is determined from the location of the assumed failure and the proximity to secondary containment ventilation systems assumed to be operating (if any).

d. Determination of the values of the meteorological parameters to be used in the dose calculations. The SAB provides the reviewer with the X/Q value for the calculation of the two-hour doses. Depending on the operability of any air treatment system(s), a ground-level or elevated (stack) release is assumed.

e. Review of the results of the dose calculations. The resulting doses from these accidents should be small fractions of the 10 CFR Part 100 limits. If this is not the case, the primary coolant activity concentration limits allowed by the technical specifications should be reduced.

3. A survey of all other lines carrying primary coolant and penetrating the containment to determine the need for any special technical specification limits for the isolation valves in these lines. A failure of the line downstream of the outboard isolation valve, combined with a single active failure of any valve in the line is postulated. If such an accident sequence could result in significant primary coolant release, it may be necessary to specify a maximum allowable valve closure time and maximum allowable leak rate for the isolation valves in the technical specifications to assure that the consequences of this accident sequence do not exceed small fractions of the exposure guidelines of 10 CFR Part 100.

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 at the operating license stage:

"The applicant's analyses of the failure of small lines carrying primary coolant outside the containment and the proposed technical specifications for limiting the consequences of such failures have been reviewed. Independent evaluations of these accidents have been performed, and we conclude that the technical specifications for the reactor coolant activity and the isolation times and leakage limits for the isolation valves in such lines [if appropriate] assure that the radiological consequences of these accidents are limited to small fractions of the dose guidelines of 10 CFR Part 100."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Containment."

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SECTION 15.6.3 RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The AAB review covers the following areas:

1. The release of secondary coolant due to a steam generator tube failure, with and without a concurrent loss of offsite power, in a pressurized water reactor (PWR) plant.

2. The calculation of whole body and thyroid doses at the nearest exclusion area boundary due to the releases resulting from these accidents.

The purposes of the review are to assure that the plant procedures for recovery from a steam generator tube failure, with and without offsite power available, are properly taken into account in the computation of whole body and thyroid doses at the nearest exclusion area boundary, and to assure that releases from the failure are adequately limited by the coolant activity concentration technical specifications. The RSB will notify the AAB if this accident is predicted to cause fuel failures, with and without a control rod held in the fully withdrawn position. The SAB provides the reviewer with the accident condition wind speed at the nearest exclusion area boundary.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a steam generator tube failure, and the primary and secondary coolant activities adequately limited, if calculations show that the resulting doses at the nearest exclusion area boundary are small fractions of the 10 CFR Part 100 exposure guidelfines, and are within 10 CFR Part 100 guidelines for the case of a coincident iodine spike or for the case of one rod held out of the core.

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 during 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 detailed review of the radiological consequences of a steam generator tube failure is done at the operating license (OL) stage when system parameters and accident analyses are fully developed. At the construction permit (CP) stage, the review is limited to a brief survey of the pertinent portions of the plant design and the applicant's discussion of this accident to determine that there are no unusual features that would prevent limitation of radiological consequences to acceptable levels by appropriate limits on coolant activity concentrations.

The AAB review of the steam generator tube failure accident at the OL stage consists of the following steps:

1. Review of the applicant's description of the tube failure accidents (with and without offsite power). This includes a review of the time steps used in the descriptions, the bases for their selection, and assurance of an adequate degree of conservatism.

2. Review of the signals available to the reactor operator that indicate the occurrence of the accident and the state of the system throughout the recovery procedure. Automatic and required manual operations by the operator as a function of time must also be determined.

3. Preparation of the input data required to run a digital computer code based on the preceding information. For this purpose the reviewer sets up a series of time intervals similar to those described by the applicant, or modified, if necessary, in order to obtain an adequate degree of conservatism (Ref. 2). The values of the parameters describing the primary and secondary system operating conditions are obtained from the summary table that the applicant provides in this section of the safety analysis report (SAR). These data may also be found elsewhere in the SAR, mainly in Chapters 4 and 10.

4. Determination of the values of the meteorological parameters for the dose calculations. The SAB provides the reviewer with the accident condition (5 percentile) wind speed at the nearest exclusion area boundary. The X/Q value for the calculation of the two-hour doses is obtained from Regulatory Guide 1.5 (Ref.3) and corrected for wind speeds differing from 1 m/sec (inverse ratio).

5. Determination of the parameters for the thyroid dose calculation. An appropriate value for the iodine decontamination factor is used in the calculation of the thyroid doses. A decontamination factor of 10 is currently being used between the water and steam phases for most plants unless the applicant presents reasonable evidence that the use of another value is justified. A breathing rate of 3.47×10^{4} m^3/sec is used in the calculation of the thyroid doses.

6. Determination of the coolant activity concentrations. The reviewer assumes the primary and secondary coolant activity concentrations allowed by the technical specifications (SAR Chapter 16) as equilibrium conditions prior to the accident. Additional coolant activity may become available for release if fuel failures result from the accident. The RSB reviews, on a generic basis, the effect of a steam generator tube failure on core thermal margins. If this event is predicted to cause fuel failures, RSB notifies AAB so that the predicted magnitude and extent of fuel failures can be properly considered in the evaluation of the radiological consequences.

7. Determination of the iodine spiking effects. The effect of iodine spiking following the accident (Ref. 4) can be accounted for by increasing the iodine source term in the primary system upon depressurization. At the present time, the I-131 equivalent source term (release rate from fuel) is increased by a factor of 500 at the time of reactor trip. A case with an iodine spike which already exists (due to a previous power transient) is also considered, assuming the I-131 equivalent coolant concentration technical specification limit for an iodine spike.

8. Determination of the leakage into the unaffected steam generators. Normal operating primary-to-secondary leakage is assumed to exist in the unaffected steam generators. The leakage rate should be the maximum allowed by the technical specifications (SAR Chapter 16). Currently this value is about 1 gpm but may be lower because of fuel densification, rod ejection accident consequences, or anticipated transient without scram (ATWS) consequences.

9. Determination of the coolant flow through the failed tube. The computer code is run for different values of the primary-to-secondary leakage through the failed tube. The flow rate resulting in the highest offsite dose is used as representative of the accident if it is smaller than the value calculated for a complete double-ended break.

10. Calculation of the exclusion area boundary doses. The reviewer uses a digital computer code with the input data and assumptions developed in the preceding steps, to determine the nearest exclusion area boundary doses for the steam generator tube failure accident. Doses are calculated with and without coincident iodine spiking.

11. Review of the results of the dose calculations. The reviewer compares the doses at the nearest exclusion area boundary, calculated without coincident iodine spiking, to the 10 CFR Part 100 guidelines. If the doses are a small fraction of the guideline values, the design is accepted. If not, the technical specification limits on the radioactivity concentrations in the coolant should be reduced accordingly. The doses calculated with coincident iodine spiking are also compared to the 10 CFR Part 100 guidelines. If they are within the guidelines, the design is accepted; if not, appropriate reductions of the technical specification limits on coolant activity concentrations are made.

12. Review of the effects of possible fuel damage in the accident on exclusion area boundary doses. The reviewer assumes that the applicant's calculations of fuel damage are correct for the case of a control rod held at the fully withdrawn position unless informed otherwise by RSB. If fuel damage does occur, calculations should be performed in order to assure that 10 CFR Part 100 guidelines are not exceeded (without a coincident iodine spike).

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 at the operating license stage:

"The steam generator tube failure accident has been evaluated with and without a concurrent loss of offsite power. The design of the plant has been found to assure an expeditious recovery from these accidents with acceptably limited activity releases.

"A decontamination factor of ____ between the water and steam phases and a X/Q value of ____ sec/m^3 has been used in our evaluation of the radiological consequences. The calculated doses are presented in Table ____. Technical specification limits on primary and secondary coolant activities will limit potential doses to small fractions of the 10 CFR Part 100 exposure guidelines. The potential doses are within the 10 CFR Part 100 exposure guidelines even if the accident were to occur coincident with an iodine spike."

The following paragraph is added if fuel damage is found to be a possible consequence of the accident:

"The evaluation of the steam generator tube failure accident has also been evaluated with ____% fuel damage in the core (as a result of the most reactive control rod remaining fully withdrawn). The resulting doses are within the guidelines of 10 CFR Part 100."

At the construction permit stage the following paragraph is included in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam generator tube failure accidents for pressurized water reactor plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications."

V. REFERENCES

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

2. H. M. Fontecilla, "Analysis of Accidental Iodine Releases from the Secondary Coolant System," Trans. Am. Nucl. Soc., 17, 336 (1973).

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

4. W. F. Pasedag, "Effects of Iodine Spiking on Light-Water Reactor Accident Analysis," Trans. Am. Nucl. Soc., 17, 336 (1973).

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SECTION 15.6.4 RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The AAB review covers the following areas:

1. The release of primary coolant due to a main steam line failure outside of containment in a boiling water reactor (BWR) plant.

2. The calculation of whole body and thyroid doses at the site boundary due to the releases resulting from these accidents.

The purposes of the review are to calculate the whole body and thyroid doses resulting from a postulated failure of a main steam line outside containment, and to assure that releases due to the failure are adequately limited by the technical specifications on primary coolant activity. The RSB will notify the AAB if this accident is predicted to cause fuel failures. The RSB may also be requested to confirm the assumption of instantaneous reactor coolant release. The SAB provides the reviewer with the accident condition wind speed at the nearest exclusion area boundary.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against the radiological consequences of a main steam line failure outside containment, and the primary coolant activity appropriately limited by the technical specifications if calculations show that the resulting doses at the nearest exclusion area boundary are small fractions of the 10 CFR Part 100 exposure guidelines.

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 detailed review of the radiological consequences of a main steam line failure outside containment is done at the operating license (OL) stage when system parameters and accident analyses are fully developed. At the construction permit (CP) stage, the review is limited to a brief survey of the pertinent portions of the plant design and the applicant's discussion of the accident to determine that there are no unusual features that would prevent limitation of doses to acceptable levels by appropriate limits on coolant activity concentrations.

The AAB review of main steam line failure accidents at the OL stage consists of the following steps:

1. Review of the applicant's descriptions of the steam line failure accidents.

2. Performance of an independent analysis of the radiological consequences of the failure of the main steam line, using the assumptions of Regulatory Guide 1.5 (Ref. 2). At present, the following conservative assumptions are used to simplify the analysis:

    a. The mass of reactor coolant released to the environment is 140,000 lbs. The release is assumed to occur instantaneously. (These assumptions are made unless notified otherwise by RSB.)

    b. The initial fission product concentrations in the primary coolant are those given as the maximum values in the technical specifications. The RSB reviews, on a generic basis, the effect on core thermal margins of a steam line failure in a BWR. If this event is predicted to cause fuel failures, RSB notifies AAB so that the predicted magnitude and extent of fuel failures can be properly considered in the evaluation of the radiological consequences. No decontamination factors, or other reductions in the fission product concentrations are assumed.

3. Determination of the values of meteorological parameters for the dose calculations. The SAB provides the reviewer with the accident condition (5 percentile) wind speed at the nearest exclusion area boundary. The X/Q value for the calculation of the two-hour doses is obtained from Regulatory Guide 1.5 (Ref. 2) and corrected for wind speeds differing from 1 m/sec (inverse ratio).

4. Review of the results of the dose calculations. The resulting doses from these accidents should be small fractions of the 10 CFR Part 100 limits. If this is not the case, the primary coolant concentration limits in the technical specifications should be reduced.

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 at the operating license stage:

"The applicant's analysis of a failure of a main steam line has been reviewed, and an independent analysis of this accident has been performed by the staff. The results of the staff's calculations are presented in Table ____. The technical specifications for the primary coolant activity concentrations limit the consequences of this accident to a small fraction of 10 CFR Part 100 exposure guidelines."

At the construction permit stage, the following paragraph is included in the staff's safety evaluation report:

"On the basis of our experience with the evaluation of steam line failure accidents for boiling water plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary coolant radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits on the primary coolant activity concentrations in the technical specifications."

V. REFERENCES

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

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

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SECTION 15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Loss-of-coolant accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and to include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished.

Each pressurized water reactor (PWR) and boiling water reactor (BWR) must be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS (Ref. 1) and the applicable general design requirements (see Standard Review Plan 6.3). The analysis of ECCS performance has an impact on the design of the piping and support structures for the reactor coolant system, the design of the steam generators, the containment design, and the possible need for pump overspeed protection.

The review of the applicant's analysis of the spectrum of postulated loss-of-coolant accidents is closely associated with the review of the ECCS, as described in Standard Review Plan (SRP) 6.3. As a portion of the review effort described in this plan and in SRP 6.3, RSB evaluates whether the entire break spectrum (break size and location) has been covered; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating

ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on ECCS performance have been provided. For postulated break sizes and locations, the RSB review includes the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

The calculational framework used for the evaluation of the ECCS system in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS (Ref. 1). The evaluation model must have been previously documented and reviewed and approved by the staff. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, reactor coolant system piping and support structure design), all differences should be identified and described by the applicant. On request, RSB reviews these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability to other branches as required. RSB requests generic computer code reviews from CPB as required.

RSB is also responsible for the review of the failure mode analysis of the ECCS in conjunction with the effort described in SRP 6.3. APCSB and EICSB provide assistance in this review, on request.

AAB provides an evaluation of fission product releases and radiological consequences. This effort is described in the appendices to this review plan.

APCSB, as described in the plans for SAR Chapters 9 and 10, provides an evaluation of auxiliary systems (e.g., service water system, component cooling system, ultimate heat sink, condensate storage facility) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a loss-of-coolant event. APCSB also, on request from RSB, reviews the failure mode analysis of the ECCS.

CSB, as described in SRP 6.2.1, evaluates the functional capability of the containment for the spectrum of loss-of-coolant events. The assumptions used for the containment response analysis must be selected in a manner conservative for the purpose. CSB, on request from the RSB, also provides an evaluation of containment pressure calculations utilized in the reflood portion of the ECCS performance analyses.

CPB, upon RSB request, reviews the power transient calculations including moderator temperature, void and fuel temperature feedback effects, and decay heat; reviews the analytical techniques used for blowdown, reflood, and clad temperature calculations; and performs independent blowdown, reflood, and clad temperature calculations, as described in SRP 4.3 and the appendix to SRP 4.4.

EICSB, as described in SRP 7.2, 7.3, 8.3.1 and 8.3.2, reviews the protection system and ECCS-associated controls and instrumentation with regard to automatic actuation, remote sensing and indication, remote control, redundancy, and emergency onsite power functional capabilities.

EICSB also, on request from RSB, reviews the failure mode analysis related to the instrumentation and electrical power supply submitted by the applicant to show that the most damaging single active failure of the ECCS was selected for the LOCA analysis.

MEB, as described in SRP 3.62 and the 3.9 plans, is responsible for the review of the effects of blowdown loads on core support structures and on control rod guide structures. MEB verifies that the core remains in place in case of a LOCA and that the control rods can be inserted. MEB is also responsible for evaluating the effects of blowdown loads including jet forces on the piping of the reactor coolant system and on the support structures of the components of the reactor coolant system. MEB verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures in the reactor coolant pressure boundary or in engineered safety feature equipment in the event of a LOCA.

II. ACCEPTANCE CRITERIA

The objectives of the review of the applicant's analysis of loss-of-coolant accidents are to verify that:

1. An devaluation of ECCS performance has been performed in accordance with an approved

   breaks, the results of the evaluation must show that the requirements of the acceptance criteria for ECCS are satisfied, namely:

   a. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

   b. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

   c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

   d. Calculated changes in core geometry are such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

For the review of the ECCS performance analysis, as presented in the applicant's safety analysis report (SAR), the reviewer verifies the following:

1. The calculations were performed using an approved evaluation model. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required.

2. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under SRP 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

3. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, sensitivity studies on the important parameters should have been made by the applicant. If such sensitivity studies are not presented in the SAR, the reviewer requests that they be made.

4. The parameters and assumptions used for the calculations conform to those of the approved evaluation model and were conservatively chosen, including the following points:

    a. Initial power level should be 102% of the proposed licensed core thermal power, as given in SAR Section 4.1.

    b. The maximum linear heat generation rate used should be based on 102% of the proposed licensed core thermal power and the technical specification limit on peaking factors, or on the technical specification limits on maximum linear heat generation rate.

c. All permitted axial power shapes, as given in Section 4.3 of the SAR should be covered by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

d. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in SAR Section 4.4, similiar calculations performed by the staff, or calculations done for similar plants by previous applicants.

e. Appropriate analyses are presented to support any credit taken for control rod insertion.

5. Reactor protection system actions and safety injection actuation and delivery are consistent with the set points and the associated uncertainties and delay times listed in the SAR (OL review). The ECCS flow rates should be checked against the applicant's data on head-flow characteristics of the ECCS pumps given in Section 6.3 of the SAR and against typical safety injection tank discharge curves used for the analysis. The Regional Offices may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

6. The results of the applicant's calculations are consistent with those of staff calculations for typical plants and also with the results of calculations performed for similar systems by previous applicants. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in core, and in the downcomer; reactor coolant temperature and quality at core inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, refill, and reflood; heat flux transients from piping and vessel walls; primary-secondary heat transfer (PWRs only); timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested); predicted "end-of-bypass" time compared to calculated downcomer flow and to staff calculations for typical plants; pump speed transients; containment pressure transients (if staff calculations are not available, these are requested from CSB); and carryover fraction (if it is not an input to the calculations).

7. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS (Ref. 1).

8. The applicant's analysis covers the full LOCA sequence of events to the point where the plant is in the long-term cooling mode and removal of decay heat has been well

established. The reviewer checks the assumed sources of coolant water, the redundancy of delivery routes, the alignment of valves, and all required operator actions.

The review of fission product releases and radiological consequences of design basis (most severe) LOCA is performed by AAB as described in the appendix to this plan.

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 applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR § 50.46). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff. The results of the analyses show that the ECCS satisfy the following criteria:

1. The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

2. The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry are such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

"The radiological consequences of the postulated spectrum of loss-of-coolant accidents (LOCA) were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS."

The evaluation findings of the AAB resulting from the reviews detailed in Appendices A, B, C, and D, as applicable, should be inserted in the safety evaluation report draft at this point. See Appendices A - D for typical findings and conclusions.

"The staff concludes that the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable."

V. REFERENCES

1. 10 CFR § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to Part 50, "ECCS Evaluation Models."

2. Standard Review Plan 6.3, "Emergency Core Cooling System."

3. Appendices A, B, C, and D, attached to this plan.

APPENDIX A

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: CONTAINMENT LEAKAGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

1. The review is concerned with the selection of the values of plant parameters used in calculating the radiological consequences of containment leakage following a loss-of-coolant accident. It is also concerned with selecting a dose computation model that incorporates conservative transport mechanisms and rates from various parts of the containment to the atmosphere, suitable breathing rates, dose conversion factors, and other physical and biological data that may affect the computed dose.

2. The calculated doses are compared with the appropriate exposure guidelines to confirm the acceptability of the nearest exclusion area boundary and low population zone (LPZ) outer boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

The ETSB reviews ESF filter system design and filter efficiencies in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

1. The fractions of the fission product inventory assumed to be available for release from the containment are acceptable if they agree with the values listed in Section C of Regulatory Guide 1.3 or Regulatory Guide 1.4. No specific list of isotopes or decay constants has been selected as standard.

   Where the applicant claims a single containment system, this is accepted. To receive credit for a dual containment system, a determination must be made that the system meets the necessary requirements. These requirements are detailed in Standard Review Plan (SRP) 6.2.3 and SRP 6.5.3. Containments falling outside of these categories are evaluated on a case-by-case basis. For single containment systems, or leakage from the primary containment, the leakage rate stated in the technical specifications is accepted subject to verification by the CSB, provided a leakage rate of at least 0.1% per day is stated. For a boiling water reactor (BWR) the leakage rate is currently assumed constant over the course of the accident, while for a pressurized water reactor (PWR), the leakage rate is reduced after 24 hours to one-half its original value (see Refs. 2 and 3). Where a single containment is specified, no credit for exhaust filters is allowed, although internal recirculation filters can be credited, if present.

2. The methods used to calculate radiological consequences of a postulated LOCA are acceptable if they reflect the use of conservative design basis assumptions as outlined in Regulatory Guide 1.3 or 1.4 (Refs. 2 and 3). The requirements of 10 CFR Part 100 are that the total dose from a postulated loss-of-coolant accident (LOCA) to an individual (located at positions specified in 10 CFR Para. 100.11(a)) must be no greater than 300 rem to the thyroid and 25 rem to the whole body. At the construction permit (CP) stage, exposures of no more than 150 rem to the thyroid and 20 rem to the whole body are considered acceptable to allow for uncertainties in meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors allowed for these features. This lower guideline is required at the CP stage to provide reasonable assurance that the 10 CFR Part 100 guideline values can be met at the operating license (OL) review stage.

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 it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The design (stretch) power level of the core is taken from the applicant's safety analysis report (SAR). The core is assumed to have operated at this power level for a sufficiently extended period (typically about 3 years) that a maximum equilibrium fission product inventory is present. At time of the accident, 25% of all the equilibrium iodine fission products and 100% of the noble gas fission products are assumed available for release from the containment within a very short time (effectively instantaneously) after the accident. The iodine is assumed to be composed of 91% elemental iodine, 4% organic iodides, and 5% particulate iodine.

2. From the applicant's SAR (parts of Section 1, Sections 6.2.1, and 6.2.3), the reviewer ascertains the type of containment system used. A check is made of the LOCA assumptions listed in Chapter 15 of the SAR to verify that the primary containment leakage rate has been assumed to remain constant over the course of the accident (for a BWR) or to be halved after 24 hours (for a PWR) and that the initial leak rate is at least 0.1% per day (a lower limit is set because of integrated containment leakage test sensitivity limitations). The leakage rate used should correspond with that given in the technical specifications in SAR Chapter 16.

3. Where credit for a dual containment system is claimed, the reviewer verifies (see SRP 6.2.3 and 6.5.3) that the system meets requirements such as existence of separate primary and secondary containments, adequate separation of the two, and ability to test the negative pressure capability of the secondary containment volume. Where credit

for a secondary containment with recirculation is claimed, adequate mixing in the secondary containment volume should be demonstrated in addition to meeting the above requirements for a dual containment system. For dual containment systems, the bypass leakage rate is noted. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to go from the primary containment directly to the environment, bypassing the secondary containment. This bypass leakage rate, as well as any positive pressure conditions should be verified by the CSB. See SRP 6.2.3 for a detailed treatment of bypass leakage.

4. Credit, if any, to be given for any engineered safety features such as filters, sprays, or ice condenser that may be present, is determined in the review of Section 6.5 of the SAR. These features operate during the LOCA to mitigate the consequences by reducing the amount of iodine fission products released to the environment. Noble gas releases to the environment are unaffected by the presence of filters or sprays. Typically, single containments employ spray systems with a chemical additive (e.g., sodium hydroxide, sodium tetraborate) designed to scavenge iodine from the containment atmosphere. The iodine removal rates of an ice condenser or a chemical additive spray system are determined after consultation with specialists in this area. For filters, verification of acceptability of design and filter efficiencies is provided by the ETSB in SRP 6.5.1. In dual containment systems, a determination must be made by the AAB of the operational modes of the ESF with respect to the accident sequence in order for proper credit to be given.

5. Sections 2.1.2 and 2.1.3 of the applicant's SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures given in SRP 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From the SAR, the reviewer also obtains relevant information (e.g., locations and time durations) concerning activities unrelated to plant operation that may exist inside the exclusion area boundaries (see SRP 2.1.2). In some cases specific dose computations may have to be performed to assist in determining the adequacy of evacuation plans.

6. The SAB is requested to furnish suitable X/Q values to be used in analyzing the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion boundary where significant activities may occur involving members of the public.

7. Based upon the review procedures already performed, a dose computation model is selected which conservatively represents the transfer of radioactivity from the containment to the environment. The reviewer may find it convenient to sketch a schematic arrangement to illustrate the compartments where radioactivity is located, with arrows drawn from one compartment to another indicating transport paths. The leak rates, spray removal rates, ice condenser efficiencies, filter efficiencies, and flow rates are all used to indicate the rates at which the activity moves from one compartment to another. Digital computer codes (Ref. 4) have been written to perform

the actual dose calculation. The analyst should select the code with capabilities that most closely fit the schematic model obtained above. The codes contain a basic library of physical and biological data which enter into the dose calculation, such as isotopic fission yields, half-lives, energies, and dose conversion factors.

8. The calculated doses, including the 2-hour thyroid inhalation and whole body doses at the nearest exclusion area boundary, the thyroid inhalation and whole body doses for the course of the accident at the outer boundary of the LPZ, and those doses calculated at other points within the exclusion area boundary where there may be activities unrelated to to plant operation at certain times, are compared with the dose guidelines as discussed in Section II.2 of this plan. Where the results of the dose calculations exceed the guidelines, the alternatives which would reduce the doses to an acceptable level are explored (e.g., increased distance, secondary containment, better filter or spray systems). The feasibility of the alternatives is also examined. The AAB Branch Chief is consulted as to appropriate action in this case.

IV. EVALUATION FINDINGS

If the AAB reviewer finds that the radiological consequences of the containment leakage contribution to a loss-of-coolant accident are acceptable, conclusions of the following type may be included with the RSB findings for this area in the staff's safety evaluation report:

"The radiological consequences of a loss-of-coolant accident as a result of leakage from the containment were evaluated. The analysis of the containment leakage doses following a postulated design basis loss-of-coolant accident included the influence of fission product removal and holdup systems and the containment leakage routes on the estimated radiological consequences.

"The review has included the applicant's proposed design criteria and design bases for the effect of containment leakage and his analysis of the manner in which the containment leakage consequences conform to the design criteria.

"The basis for acceptance in the staff review has been confirmation that the applicant's analysis conforms with the applicable regulations, regulatory guides, technical positions and industry standards as listed in Table 15.13.2-1. The staff concludes that the proposed design, including leakage rates and fission product removal and control systems conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and that the conservatively computed doses from containment leakage following a loss-of-coolant accident are within the exposure guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SER. The following should be added at the CP stage: "Because the proposed design meets the recommendations of Regulatory Guide 1.3(1.4), there is reasonable assurance that the exposure guidelines of 10 CFR Part 100 can be met at the OL stage."

V. REFERENCES

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

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.

4. Computer codes are currently under development. Documentation will be published in a NUREG report.

APPENDIX B

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM ENGINEERED SAFETY FEATURES COMPONENTS OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage from engineered safety features (ESF) equipment which is located outside the primary containment. Such leakage could occur during the recirculation phase for long-term core cooling and primary containment (spray) cooling. The total leakage from these sources is added to that resulting from the containment leakage following a LOCA. To calculate the maximum potential leakage from the recirculation loop, such sources as the following are considered: containment spray system, low pressure safety injection system, and high pressure safety injection system.

The ETSB reviews ESF ventilation system filters for conformance with Regulatory Guide 1.52 in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

The source of leakage is related to the requirement to detect and isolate failures of passive components in the long-term (recirculation) mode for ESF systems. Therefore, leakage anywhere in the systems carrying recirculation water outside of containment is postulated. ESF-grade filtration systems to process potential leakage are required as the dose could exceed 10 CFR Part 100 guidelines without filters even at relatively low leakage rates resulting from passive failures. When ESF-grade filters are supplied, no doses resulting from passive failures need be considered.

The acceptance criterion for the dose resulting from leakage outside primary containment from the recirculation systems is that when it is added to the dose attributable to containment leakage, including any main steam isolation valve sealing system leakage (Appendices A and D of Standard Review Plan 15.6.5), the total dose is to be within the guideline values of 10 CFR Part 100. To provide assurance that this criterion is met at the operating license stage, the doses indicated in Regulatory Guides 1.3 and 1.4 are used as acceptance criteria at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review 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 if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's recirculation leakage calcualtion is checked against previously licensed plants for accuracy and completeness. It is assumed that 50% of the core iodine inventory, based upon the maximum reactor power level, is mixed in the sump water being circulated through the external piping systems. Credit may be allowed for radioactive decay of the iodine during the time period from the occurrence of the LOCA up to the beginning of recirculation when the sump water is circulated outside the containment.

The dose computed for presentation in the staff safety evaluation report (SER) should be based upon twice the maximum operational leakage and should be assumed constant for the course of the accident. The maximum operational leakage is defined as the sum of the leakage for all the recirculation systems (1) which is detectable during test and (2) above which the technical specifications whould require declaring a system out of service. The leakage is assumed to occur throughout the accident, starting at the earliest time that recirculation mode is initiated.

The applicant's data on sump water temperature versus time after the LOCA should be consulted and used. During the time that the circulating water temperature exceeds 212°F, the fraction of water flashing to steam should be computed and taken as the fraction of iodine in the water which becomes volatile. In those cases where the circulating water temperature is less than 212°F, 10% of the iodine in the water which leaks is assumed to become volatile unless a smaller amount is justified based on actual sump pH history and ventillation rates.

All the iodine becoming volatile is assumed to be released immediately to the environment, and atmospheric dispersion is based upon the ground level X/Q values determined by the SAB. Any ventilation system filters are evaluated by the ETSB for compliance with Regulatory Guide 1.52 (Ref. 4) and appropriate credit for iodine removal by the filters given. The nearest exclusion area boundary and LPZ outer boundary doses are calculated by standard methods as described in Appendix A to SRP 15.6.5.

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 with the RSB findings for Section 15.6.5 in the SER:

> "The staff concludes that doses resulting from the postulated leakage of post-LOCA recirculation water from pump seals, valve packings, etc., are low and, when added to the direct leakage LOCA doses, result in total doses that are within the guideline values of 10 CFR Part 100. Engineered safety feature-grade filtration systems are provided to process potential leakage from postulated failures of passive components in systems carrying post-LOCA recirculation water outside of containment."

V. REFERENCES

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

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.

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. Appendices A and D, Standard Review Plan 15.6.5.

APPENDIX C

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: HYDROGEN PURGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The radiological consequences of purging any hydrogen accumulation in the containment after a postulated loss-of-coolant accident (LOCA) are reviewed to establish that the LOCA-plus-purge doses are acceptable and, in some cases, to determine whether additional filtration systems are needed. The ETSB reviews hydrogen purge system filters in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

The acceptance criteria for hydrogen purging doses are given in Section B of Branch Technical Position CSB 6-2 (Ref. 2).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the area 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 determines which criteria in Reference 2 are to be met and then performs a purging dose calculation following the procedures outlined below.

1. Source Terms

a. Iodine

The initial airborne iodine-131 component is assumed to be 25% of the core inventory, as stated in Regulatory Guides 1.3 and 1.4. The iodine airborne activity at any subsequent time is subject to a removal factor due to the plant engineered safety features and to radioactive decay. (See Enclosure 1)

b. Noble Gases

The initial Xe-133 and Kr-85 activities are assumed to be 100% of the core inventory, as stated in Regulatory Guides 1.3 and 1.4. These nuclides are subject to removal through radioactive decay only. (See Enclosure 2, 3)

2. Purging

The model assumes a constant purge rate after initiation of purge. No credit is included for depletion of activity due to containment leakage.

3. Dose Model for I-131

a. Assumptions

	Code	Units
(1) 50% plateout	PLTOUT	-
(2) 50% released	REL	-
(3) Core inventory	TID	Ci/MWt
(4) Dose conversion factor	DCFLOD	Rad/Ci
(5) Breathing rate (4-30 day rate)	BR	m^3/sec
(6) Iodine-131 decay constant	LAMDA	day^{-1}

b. Variables

	Code	Units
(1) Purge rate	PURGRT	SCFM
(2) Iodine reduction factor	RF	-
(3) Purge time	PURGTM	days
(4) Power level	POWLEV	MWt
(5) X/Q	XQ	sec/m^3
(6) Hold up time	HOLDUP	days
(7) Containment building volume	VOLCON	ft^3

c. Model

	Code	Units
(1) Core inventory:	I-131	Ci
I-131 (PLTOUT)(REL)(POWLEV)(TID)/RF	-	-
(2) Activity in containment at time of purge:	CI-131	Ci
CI-131 = I-131 exp (-LAMDAxHOLDUP)	-	-
(3) Concentration in containment at time of purge:	CONCON	Ci/cm^3
CONCON = CI-131/VOLCON	-	-
(4) Differential change in containment atmosphere concentration due to replacing the portion of the atmosphere vented per unit time with clean air:		
BETA = PURGRT/VOLCON	BETA	$days^{-1}$
differential change = exp (-BETAxPURGTM)	-	-
(5) Total activity released during course of purge:	TAR	Ci
$TAR = \int_0^{PURGTM} (CONCON)(PURGRT) \times \exp(-LAMDA-BETA)\, t \; dt$	-	-
(6) Dose at boundary due to iodine-131:	DOSEIOD	Rem
DOSEIOD = (TAR)(BR)(DCFlOD)(XQ)	-	-

4. Dose Model for Xe-133

a. Assumptions

	Code	Units
(1) 0% plateout	PLTOUT	-
(2) 100% released	REL	-
(3) Core inventory	TIDXE	Ci/MWt
(4) Xe-133 decay constant	LAMDA	$days^{-1}$

b. Variables

	Code	Units
Use the I-131 variables and let iodine reduction factor = 1.	-	-

c. Model

	Code	Units
Use the I-131 model with Xe-133 assumptions.	-	-

5. Dose Model for Kr-85

a. Assumptions

	Code	Units
(1) 0% plateout	PLTOUT	-
(2) 100% released	REL	-
(3) Core inventory	TIDKR	Ci/MWt
(4) Kr-85 decay constant	LAMDA	$days^{-1}$
(5) Average gamma energy	GAMENG	Mev
(6) Average beta energy	BETENG	Mev

b. Variables

	Code	Units
Use Xe-133 variables.	-	-

c. Model

	Code	Units
(1) Core inventory:	KR-85	Ci
KR-85 = (PLTOUT)(REL)(POWLEV)(TIDKR)	-	-
(2) Activity in the containment at time of purge:	CIKR-85	Ci
CIKR-85 = KR-85 exp (-LAMDAxHOLDUP)		
(3) Concentration in containment at time of purge:	CONCON	Ci/cm^3
CONCON = CIKR-85/VOLCON		
(4) Differential change in containment atmosphere concentration due to replacing the portion of the atmosphere vented per unit time with clean air:		
BETA = PURGRT/VOLCON	BETA	$days^{-1}$
differential change = exp (-BETAxPURGTM)	-	-
(5) Total activity released during course of purge:	TAR	Ci
$TAR = \int_0^{PURGTM} (CONCON)(PURGRT) \times \exp(-LAMDA-BETA)\, t\, dt$	-	-
(6) Dose at boundary due to Kr-85 beta:	DOSWBB	Rem
DOSWBB = 0.246(TAR)(BETENG)(XQ)	-	-
(7) Dose at boundary due to Kr-85 gamma:	DOSWBG	Rem
DOSWBG = 0.246(TAR)(GAMENG)(XQ)	-	-
(8) Total dose at boundary due to Kr-85:	TKR-85	Rem
TKR-85 = DOSWBB + DOSWBG	-	-
(9) Total whole body dose at boundary:	TWBD	Rem
TWBD = TKR-85 + DOSXE	-	-

The data required for this calculation are obtained from the following sources. The CSB determines the purge rate, in SCFM, and the hold-up time (in days) prior to purging. The SAB determines the ground level release X/Q (30-day value) derived from onsite data. The ETSB in SRP 6.5.1 determines filter efficiencies in cases where filters are required to meet the dose criteria. LOCA analysis assumptions as to reactor power level, primary containment volume, and iodine reduction factor are obtained from the results of the AAB review under Appendix A to Standard Review Plan (SRP) 15.6.5.

For those plants not excepted from the requirements of Section B of Reference 2, the reviewer is responsible for transmitting the requirements for filters to the ETSB when such requirements are indicated by the results of the dose calculation.

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 with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

> "The analysis of the radiological consequences of containment hydrogen purging following a LOCA yields acceptable thyroid and whole body dose values."

If the reviewer finds the consequences unacceptable, then the following may be stated for current reviews:

> "The analysis of the radiological consequences of containment hydrogen purging following a LOCA indicates that the total long-term doses from the LOCA and the purge exceed the guideline values of 10 CFR Part 100 at the LPZ outer boundary. Accordingly, remedial measures (inert gas injection or filters) are required to achieve acceptable dose levels."

Conclusions which match the acceptance criteria for older plants should be drafted for such plants.

V. REFERENCES

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

2. Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," attached to Standard Review Plan 6.2.5.

3. 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."

4. Appendix A, Standard Review Plan 15.6.5.

APPENDIX D

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Site Analysis Branch (SAB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage past the main steam isolation valves in a BWR. This leakage is required to be controlled by a main steam isolation valve leakage control system (MSIVLCS). This system may act as a positive sealing system or a vacuum-type system which collects leakage between the closed isolation valves and releases it to the atmosphere through a filter system. The method of operation, time of operation, and release paths associated with the operation of the MSIVLCS are reviewed to calculate the fission product releases and their contributions to the doses at the nearest exclusion area boundary and LPZ outer boundary. Any leakage from the isolation valves (e.g., valve stem leakage) or any release from the MSIVLCS is added to the containment leakage and ESF leakage (Appendices A and B of Standard Review Plan 15.6.5) following a LOCA.

II. ACCEPTANCE CRITERIA

The calculated doses associated with operation of the MSIVLCS following a postulated LOCA should be limited so that when they are added to the dose contribution from containment leakage and leakage from ESF components outside containment, the total does not exceed the guideline values of 10 CFR Part 100 (Ref. 1) at the operating license stage or Regulatory Guide 1.3 (Ref. 2) at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review 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 if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's description of the MSIVLCS is reviewed to familiarize the reviewer with the system performance and to obtain the information needed to perform the dose calculation. For a positive sealing system, verification of the system operability assuming a single active failure, actuation time, and identification of any potential release paths is obtained from the APCSB. If the reviewer finds that no release paths exist and that the system can be actuated within an appropriate time after the accident, no further review is required.

For a vacuum-type system, which processes rather than seals the leakage, the following information, assuming the most adverse single failure of an active component, must be verified by the APCSB (and documented by buckslip to the AAB):

1. Release paths and fractions of the leakage through these paths, as a function of time, e.g., steam leakage, releases through a depressurization line, releases through drain lines, etc.

2. System actuation time.

3. Flow rates as a function of time.

4. Release points.

Interaction with systems used to mitigate the consequences of containment leakage should be noted. It may be necessary to establish with the CSB that the operation of the MSIVLCS does not adversely affect pressure transients in secondary containment regions.

The system is then modeled using a computer code (Ref. 3). The source assumed is the same as that used to estimate the containment leakage dose, but it is assumed to be instantaneously distributed in the drywell free volume at the time of the accident. Credit for decay in the drywell is given; no release is assumed up to the time of system actuation; but no credit is given for leakage from the drywell to the containment (Mark III) or the suppression pool region (Mark I and II). The main steam isolation valves are assumed to be leaking at their technical specification limit. Leakage through valve stems or drain lines to an untreated region is assumed to be released to atmosphere; releases through the MSIVLCS which are directed to treated regions are assumed to be directly to the filter intake unless the MSIVLCS flow is mechanically directed to a distributed header. If the latter is the case, then credit for mixing is given on the same basis as in other leakage to this system (see Standard Review Plan 6.5.3).

The resulting doses are calculated using the model described in Regulatory Guide 1.3 (Ref. 2). The X/Q values to be used are the accident X/Q's provided by the SAB. For systems which are designed for initial releases at later times into the accident, application of worst meteorology at the time of release may have to be considered; this will be handled on a case-by-case basis.

The doses are added to those estimated from containment leakage and the leakage from ESF components outside containment and the total is compared to the guidelines of 10 CFR Part 100 (Ref. 1) if the application is for an operating license and to the guidance of Regulatory Guide 1.3 (Ref. 2) if the application is for a construction permit.

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 with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

"The radiation doses resulting from main steam isolation valve leakage and operation of the main steam isolation valve leakage control system following a postulated LOCA were estimated assuming a single failure that is most adverse from the standpoint of radiological consequences. The analysis included the influence of fission product removal systems, delay times, and various release paths. The review has established that the applicant's design is sufficient to limit the radiological consequences due to the main steam isolation valve leakage or due to operation of the MSIVLCS such that when combined with the releases from other paths, the total potential consequences at the nearest exclusion area boundary and at the low population zone outer boundary are well within the guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SAR. The following should be added at the CP stage: "Because the proposed design meets the guideline values of Regulatory Guide 1.3, there is reasonable assurance that the exposure guideline of 10 CFR Part 100 can be met at the operating license stage."

V. REFERENCES

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

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. Computer codes are currently under development. Documentation will be published in a NUREG report.

4. Appendices A and B, Standard Review Plan 15.6.5.

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SECTION 15.7.1 WASTE GAS SYSTEM FAILURE

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

1. The radiological consequences of an unexpected and uncontrolled release to the atmosphere of radioactive fission gases that are stored or transferred in the waste gas system are reviewed to determine compliance with dose criteria.

2. The applicant's safety analysis report (SAR) is reviewed to see that there are technical specifications to limit to an appropriate level the activity which could be released, assuming the failure of any active component in the waste gas system. (The ETSB reviews the waste gas system design on SRP 11.3.)

II. ACCEPTANCE CRITERIA

Failure of waste gas systems that comply with the current staff position on seismic and quality group design requirements should result in doses well within the guideline values of 10 CFR Part 100 using conservative calculational assumptions. Systems whose failure would result in exposures approaching or exceeding the guideline values of 10 CFR Part 100 should have a technical specification limiting the amount of activity in the system. The technical specification should limit the activity in waste gas system components such that any single failure of a component will not result in a whole body dose at the nearest exclusion area boundary in excess of 0.5 rem.

For older plants, where it is proposed to modify the waste gas systems, systems which do not meet the current seismic requirements should have a technical specification such that the two-hour whole body dose at the nearest exclusion area boundary, in the event of a system failure, is less than 5 rem using conservative calculational assumptions.

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.

If the current staff positions on quality and seismic design provisions are complied with, the probability of failure of passive components is sufficiently low that the consequences may be compared with the exposure guidelines of 10 CFR Part 100. Conservative calculational assumptions, similar to those used in analyzing other design basis accidents, should be used. (See, for example, Standard Review Plan 15.6.5.)

The reviewer obtains assistance from the ETSB as necessary to account for special characteristics of the waste gas system. The appropriate X/Q values are obtained from the SAB. The radiological consequences are then computed using a digital computer code (Ref. 2).

At the operating license stage, the reviewer verifies that a technical specification has been provided to limit the waste gas inventories in components such that single active failures of components, such as the lifting and sticking of a relief valve, will result in small exposures when the consequences are conservatively calculated. Unless otherwise demonstrated, all the inventory of a pressurized tank (and normally interconnected tanks if these would not be automatically isolated) is assumed to be released on lifting and sticking of a relief valve. If charcoal is present, a slow evolution of gas may be assumed for the fraction of gas shown not to be released during the tank depressurization.

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:

> "Potential failures of the waste gas system have been reviewed and the computed doses have been found to be well within the guideline values of 10 CFR Part 100. Technical Specification limits have (will be) set to limit potential doses from single failures of active components to small fractions of the 10 CFR Part 100 guideliens."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."
2. Computer codes are currently under development. Documentation will be published as a NUREG report.

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SECTION 15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The radiological consequences of the release to the atmosphere of radioactive fission gases resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in a waste system are reviewed by the AAB to determine that they are small fractions of the 10 CFR Part 100 guideline values.

The ETSB reviews the radioactive liquid waste system in SRP 11.2 and the liquid radioactive source term in SRP 11.1.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a radioactive liquid waste system leak or failure if the conservatively calculated exposures resulting from the release of radioactive gases from the system are small fractions of the 10 CFR Part 100 guideline values. Iodine decontamination factors similar to those specified in Standard Review Plan (SRP) 15.6.5, Appendix B, are acceptable.

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 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.

If the major parameters of the applicant's calculation (e.g., iodine decontamination factors, direct release of gases to environment, and X/Q values) are in agreement with the staff's values and if the calculated doses are less than 0.5 rem to the whole body and 1.5 rem to the thyroid, no further review is required.

If the reviewer finds that an independent calculation is necessary, the following steps are taken:

1. The appropriate atmospheric dispersion parameters are obtained from the SAB for use in calculating the thyroid and whole body doses.

2. Information describing an accident involving a radioactive liquid leak or tank failure, including the source term, is obtained from the ETSB. This should include a case with dissolved noble gases and one with high iodine inventory.

3. The thyroid and whole body doses resulting from gaseous transport from liquid spills are calculated using the above information and that presented by the applicant in the safety analysis report.

4. The calculations described above should take into account any related technical specifications and should be based on assuming an appropriate value for the iodine decontamination factor. (See SRP 15.6.5, Appendix B.)

5. The resulting doses are evaluated to determine that they are small fractions of the 10 CFR Part 100 guideline values.

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 radiological consequences of the release to the atmosphere of radioactive fission gases resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in a waste system have been evaluated and the estimated doses found to be small fractions of the 10 CFR Part 100 guidelines values."

V. REFERENCES

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

2. Standard Review Plan 15.6.5, Appendix B.

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SECTION 15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

REVIEW RESPONSIBILITIES

Primary - Effluent Treatment Systems Branch (ETSB)

Secondary - Site Analysis Branch (SAB)

I. AREAS OF REVIEW

1. The ETSB reviews the consequences of single failures involving tanks and associated components containing radioactive liquids.

2. The SAB provides information on the site geology, hydrology, and the parameters governing liquid waste movement through the soil, i.e., dilution by groundwater, lateral dispersion, transit time, hydraulic gradient, permeability, and effective porosity, based on single failure assumptions.

II. ACCEPTANCE CRITERIA

Tanks and associated componets containing radioactive liquids are acceptable if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2, at (1)l the nearest potable water supply, and (2) the nearest surface water supply in a unrestricted area, or special design features are pro vided to mitigate the effects of postulated failures for systems not meeting these limits.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from this plan as may be appropriate for a particular case.

1. The reviewer will evaluate the consequences of postulated single failures for tanks and associated components that contain contaminated liquids, where the leaked fluid is capable of affecting the nearest potable water supply or the nearest surface water in an unrestricted area. The reviewer will select tanks or components for which a failure is assumed for evaluation purposes based on the radionuclide inventory in the components, and the potential for the contaminated liquids entering the groundwater or a potable water supply.

   a. The radionuclide inventory in failed components is based on assuming 80% of the liquid volume in each component and the design basis failed fuel fraction, i.e., 1% of the fuel producing power in a pressurized water reactor (PWR), or, consistent with an offgas release rate of 100 μCi/sec-MWt after 30 minutes delay for a boiling water reactor (BWR). The radionuclide inventory is calculated by the GALE Code using values of parameters for radionuclide removal by processing components given in Regulatory Guides 1.BB or 1.CC (Refs. 1, 2).

b. The reviewer will consider the design features incorporated to mitigate the effect of a postulated failure, e.g., steel liners in building areas housing components. Normally, because of the potential radionuclide inventory, the failed components that are considered are (1) waste collector tanks, (2) evaporator concentrate tanks, (3) phase separator tanks, and (4) spent resin storage tanks. The components selected for evaluation are based on the individual plant design.

2. The radionuclide concentrations at the nearest potable water supply and nearest surface water supply are calculated by the GALE Code using the values of hydrological parameters provided by SAB. Credit for liquid retention by unlined building foundations will not be given regardless of the building seismic category because of the potential for cracks. Credit is not allowed for retention by coatings or leakage barriers outside the building foundation.

3. The calculated radionuclide concentrations at the nearest potable water supply and nearest surface water in an unrestricted area are compared to the concentration limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

4. The reviewer may elect to use the applicant's evaluation in lieu of an independent calculation. In this case, the applicant's parameters are verified by ETSB and SAB, as appropriate, and the calculated concentrations are adjusted for inconsistencies between the respective models.

IV. EVALUATION FINDINGS

If the review confirms that the consequences of liquid-containing tank failures would be acceptable according to the criteria stated in Section II, conclusions of the following type are provided for the staff's safety evaluation report:

"The consequences of tank and associated component failures which could result in contaminated liquid releases to the environs were evaluated for tanks and components containing radioactive materials located outside reactor containment. The scope of the review included the calculation of radionuclide inventories in station components at design basis fission product levels, the mitigating effects of the plant design, and the effect of site geology and hydrology. Radionuclide concentrations were calculated at the nearest potable water supply and at the nearest surface water in an unrestricted area.

"The basis for acceptance in our review has been that the postulated failures would not result in radionuclide concentrations in excess of 10 CFR Part 20 limits at the water sources considered above.

"Based on the foregoing evaluation we conclude that the provisions incorporated in the applicant's design to mitigate the effects of tank and component failures involving contaminated liquids are acceptable."

V. REFERENCES

1. Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors"

2. Regulatory Guide 1.CC, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Boiling Water Reactors"

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SECTION 15.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

1. The review is concerned with the selection of values of plant parameters for use in calculating the radiological consequences of a fuel handling accident, and the selection of the dose computation model, including assumptions of transport mechanisms and rates from the fuel handling area to the atmosphere, breathing rates, dose conversion factors, and other physical and biological data that may affect the calculated dose.

2. The calculated doses are compared with the appropriate exposure guidelines to determine the acceptability of the nearest exclusion area boundary and low population zone (LPZ) outer boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

The ETSB reviews the fuel handling building ventilation system filters in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

1. The acceptance criteria for those reactor operating parameters and other assumptions and conditions that determine the amounts of fission product activity assumed to be available for release above the fuel pool are largely given in Regulatory Guide 1.25 (Ref. 2). These include such items as gap activities, pool decontamination factors, and radial peaking factors.

2. The dose to an individual from a postulated fuel handling accident should be well within 10 CFR Part 100 exposure guidelines. At the construction permit (CP) review stage, the doses calculated should allow adequate margin for uncertainties to assure that the doses will be well within 10 CFR Part 100 guidelines at the operating license (OL) review stage.

3. In view of the possibility of a fuel handling accident, an ESF filter system is required for the spent fuel storage facility to reduce the potential consequences.

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 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 relevant portion of Chapter 15 of the applicant's safety analysis report (SAR) should be read to assure that the values of those fuel parameters which affect fission product release and fuel pool iodine decontamination factors, including the maximum fuel rod pressurization, peak linear power density for the highest power assembly, maximum centerline operating fuel temperature for the peak assembly, average burnup for the peak assembly, and minimum water depth between the top of any damaged fuel rods and the fuel pool surface are such that Regulatory Guide 1.25 (Ref. 2) is applicable. Where Regulatory Guide 1.25 is applicable, Chapter 15 and the technical specifications of the SAR are reviewed to assure that the "conservative analysis" assumptions given for the fuel handling accident agree with those listed in the guide.

   Where the values of the fuel parameters are such that Regulatory Guide 1.25 is not applicable, an individual case-by-case analysis is performed. The AAB Branch Chief is consulted as to appropriate action.

2. Three important parameters affecting the radiological consequences of a fuel handling accident are not covered in Regulatory Guide 1.25. These are the reactor design (stretch) power level, the earliest time after reactor shutdown that fuel handling operations commence, and the number of fuel rods assumed to be damaged in a fuel handling accident. The reactor design power level is obtained from Section 1.1 or Chapter 15 of the applicant's SAR. The "conservative analysis" assumptions listed in Chapter 15 of the SAR are used to determine the earliest time after shutdown for fuel handling and the number of fuel rods damaged. The "conservative analysis" assumptions listed in Chapter 15 are generally considered acceptable and can be used for obtaining the earliest time after shutdown that fuel handling commences (typical values are 24 hours for a boiling water reactor and 100 hours for a pressurized water reactor) and the number of fuel rods damaged. All the pins in one fuel bundle are assumed damaged unless the applicant's analysis shows additional damage, in which case the applicant's assumptions should be adopted. The technical specifications are also reviewed to confirm the earliest time after shutdown that fuel handling commences.

3. The credit to be given for engineered safety features such as filter systems must be determined. For the filters themselves, verification of acceptability and efficiencies is provided by the ETSB in SRP 6.5.1. In a dual containment system where the fuel building may be exhausted through the standby gas-treatment system (SGTS), the AAB must determine the relation between the operational modes of the SGTS and the time sequence of the accident in order to give proper credit.

4. Sections 2.1.2 and 2.1.3 of the SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures

given in Standard Review Plan (SRP) 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From the SAR, the reviewer also obtains relevant information (e.g., locations and time durations) concerning activities within the exclusion area boundary unrelated to facility operation.

5. The SAB is requested to furnish suitable X/Q values to be used in analyzing the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion area boundary where there may be significant activities unrelated to facility operation.

6. The relevant plant parameters and the X/Q values obtained from SAB are used as input to an appropriate digital computer code (Ref. 3) and the doses due to a postulated fuel handling accident are calculated at the nearest exclusion area boundary, the outer boundary of the LPZ, and at those locations within the exclusion area boundary where significant activities unrelated to facility operation may occur.

7. The calculated doses are compared to the acceptance criteria in Section II. Where results of the dose calculations indicate the guidelines may be exceeded, alternatives which would reduce the doses to an acceptable level are explored (e.g., increased distance, better filters); the feasibility of the alternatives is also examined. The AAB Branch Chief is to be consulted as to appropriate action.

8. The applicant's SAR is examined to assure that an ESF filter system to mitigate the radiological consequences of a fuel handling accident are included in the design of the spent fuel storage facility.

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 staff has evaluated the applicant's analysis of postulated fuel handling accidents and finds the assumptions and calculational techniques acceptable. After performing an independent analysis of the radiological consequences of a fuel handling accident to any individual located at the nearest exclusion area boundary, at the outer boundary of the low population zone (LPZ), or at any point within the exclusion area boundary where there may be significant activity unrelated to plant operation, the staff concludes that the doses are well within the guideline values of 10 CFR Part 100."

V. REFERENCES

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

2. 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."

3. Computer codes are currently under development. Documentation will be published as a NUREG report.

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SECTION 15.7.5 SPENT FUEL CASK DROP ACCIDENTS

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

The AAB reviews accidents involving a drop of a spent fuel cask, as described in the applicant's safety analysis report (SAR), Section 15.7.5. The points covered in the review are as follows:

1. APCSB is consulted for verification of the potential drop height during handling of a loaded cask and the procedures for handling the cask with respect to the impact limiter. If the handling procedures meet all applicable criteria, then the radiological consequences of a spent fuel cask drop accident need not be estimated.

2. A design basis radiological analysis is performed if a cask drop exceeding 30 feet can be postulated or the impact limiter is removed from the cask during handling within the plant. If the radiological consequences of a cask drop accident are to be computed, then information on whether building integrity can be expected after a cask drop is obtained from APCSB (e.g., whether the technical specifications require large doors to be closed during fuel handling and whether the building integrity would be violated by the cask drop). Verification that loss of coolable fuel geometry would not be expected to occur is also obtained from APCSB to justify the assumption that only gap activity is released.

3. The SAR and technical specifications are reviewed and the relevant plant parameters are evaluated for incorporation into the dose computation model. The model incorporates conservative transport mechanisms and rates from the fuel release to the atmosphere, suitable breathing rates, dose conversion factors, and other physical and biological data that may affect the dose. The X/Q data are obtained from SAB. It may be found appropriate to utilize analyses from previous cases to determine the consequences on a generic basis.

4. The calculated doses are compared with exposure guidelines to determine the acceptability of the exclusion and low population zone (LPZ) distances and to confirm the

adequacy of engineered safety features (ESF) provided for the purpose of mitigating potential doses from spent fuel cask drop accidents.

II. ACCEPTANCE CRITERIA

1. The plant design with regard to spent fuel cask drop accidents is acceptable without calculation of radiological consequences if potential cask drop distances are less than 30 feet and appropriate impact limiting devices are employed during cask movements.

2. If the radiological consequences of a spent fuel cask drop accident are to be considered, the plant design is acceptable in this regard if the doses to an individual at the nearest exclusion area boundary and LPZ outer boundary distances are calculated to be well within 10 CFR Part 100 exposure guidelines. At the construction permit (CP) review stage, the doses calculated should allow adequate margin for uncertainties to assure that the doses will be well within 10 CFR Part 100 guidelines at the operating license (OL) review stage.

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 procedure is to determine, with the assistance of the APCSB as described in Section I of this plan, whether radiological consequences of a spent fuel cask drop accident need be evaluated. If a radiological consequence calculation is found to be necessary, the procedure is as follows:

1. The fuel element gap inventory is determined in a manner similar to that for a fuel handling accident (see Ref. 2). The differences are that a longer decay time is allowed (earliest time after reactor fueling that cask loading operations commence) and the number of fuel elements involved is based on the largest capacity cask available or projected to be available.

2. If the drop is assumed to occur inside the refueling facility at a time when the facility is closed, at a minimum negative pressure of 0.25-inch water gauge, and ESF-grade charcoal filtration is available, credit may be allowed for iodine filtration. For the filters themselves, verification of acceptability and efficiencies is provided by the ETSB. In a dual containment design where the fuel building may be exhausted through the standby gas-treatment system (SGTS), AAB determines the relationship of the operational modes of the SGTS to the time sequence of the accident in order to give proper credit.

3. If the spent fuel cask drop is assumed to occur at a time when the facility is open to the outside atmosphere, an untreated puff release is assumed.

4. Sections 2.1.2 and 2.1.3 of the applicant's SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures given for Standard Review Plan (SRP) 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From these SRP's the reviewer also obtains relevant information (locations and time durations) concerning any significant activities within the exclusion area boundary which are unrelated to facility operation.

5. The SAB is requested to furnish suitable X/Q values to analyze the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion area boundary where there may be significant activities unrelated to plant operation.

6. The relevant plant parameters and the X/Q values obtained from the SAB are used as input to a digital computer code (Ref. 3). The doses due to a postulated spent fuel cask drop accident are calculated at the nearest exclusion area boundary, the outer boundary of the LPZ, and at those locations within the exclusion area boundary where there may be significant activities unrelated to plant operation.

7. The calculated doses are compared with the acceptance criteria in Section II. Where results of the dose calculations indicate the guidelines may be exceeded, alternatives which would reduce the dose to acceptable levels are explored (e.g., increased distance, better filters); the feasibility of the alternatives is also examined. The AAB Branch Chief is to be consulted as to appropriate action.

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:

"We have evaluated the applicant's analysis of postulated spent fuel cask drop accidents and find the assumptions and calculational techniques acceptable. After performing an independent analysis of the radiological consequences to any individual located at the nearest exclusion area boundary, at the outer boundary of the low population zone (LPZ), or at any point within the exclusion area boundary where there may be significant activities unrelated to plant operation, we conclude that the doses are well within the guideline values of 10 CFR Part 100. The doses are listed in Table ____. "

V. REFERENCES

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

2. 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," and Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents."

3. Computer codes are currently under development. Documentation will be published as a NUREG report at a later date.

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SECTION 15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Anticipated transients are transients expected to occur during the life of the plant. Anticipated transients without scram (ATWS) are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram). The failure of the reactor to shut down quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. Typical transients that may have unacceptable consequences if there is a scram failure are: loss of feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of a-c power, loss of condenser vacuum, and, in a boiling water reactor (BWR), closure of main steamline isolation valves. The effects of all ATWS events must be evaluated, including a complete analysis for those resulting in the most severe consequences. Each should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1).

According to the licensing position on ATWS (Ref. 2), plants for which a construction permit (CP) application is filed after October 1, 1976 must have reactor shutdown systems of such high reliability that ATWS events need not be considered. This standard review plan (SRP), therefore, applies only to plants where application was made for a CP before that date.

For such plants, the review of ATWS events is concerned with the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted results and consequences of the transient.

The sequence of events described in the applicant's SAR is reviewed by both RSB and EICSB. The RSB reviewer concentrates on the effect of the reactor protection system, engineered safety systems, other operational and control systems, and operator action on the course of the transient and the securing and maintenance of the reactor in a safe condition. The

EICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and operational safeguards controls and instrumentation systems (other than the scram system which is assumed to have failed) will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. EICSB also evaluates potential bypass modes and the possibility of manual control by the operator.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by CPB. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

CSB reviews ATWS effects on the containment system with regard to the effects of the resulting pressures and temperatures on its functional capabilities.

MEB reviews the effects of ATWS events on the structural integrity of the reactor coolant pressure boundary.

AAB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from ATWS events are within acceptable limits. (See attached appendix.)

The predicted results of ATWS events are reviewed to assure that the consequences meet the acceptance criteria given in Section II below. Further, the results of the analyses are reviewed to ascertain that the values of the pertinent initial conditions and system parameters are within ranges expected for the type and class of reactor under review.

II. ACCEPTANCE CRITERIA

1. The general objectives in the review of ATWS events are:

    a. To identify the ATWS events that are most limiting.

    b. To verify that for the most limiting events the specific criteria applicable to ATWS events are met.

2. The specific criteria that apply depend on the date the CP application is filed and whether the need to take ATWS into account was noted in the staff's safety evaluation report (SER) or in the report of the Advisory Committee on Reactor Safeguards (ACRS). If the need to take ATWS into account was noted in either of these reports and if an application for a CP is filed before October 1, 1976, then the following criteria (Appendix A of Ref. 2) apply:

a. Calculation of Consequences

The calculated radiological consequences should be within the guideline values set forth in 10 CFR Part 100. In addition, the limits listed below on calculated system pressure, fuel performance, and containment conditions should be met.

1. Reactor Coolant System Pressure

The maximum acceptable calculated transient reactor coolant system pressure should be based on the system boundary pressure limit or the fuel pressure limit, whichever is more restrictive:

(i) Reactor Coolant System Boundary Pressure Limit

The calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Code, Section III (Ref. 3).

(ii) Fuel Pressure Limit

The calculated reactor coolant system transient pressure should not exceed a value for which tests and analyses demonstrate that there is no significant safety problem with the fuel.

2. Fuel Thermal and Hydraulic Performance

(i) The calculated average enthalpy of the hottest fuel pellet should not result in significant cladding degradation or significant fuel melting.

(ii) A calculated critical heat flux event should not occur unless the calculated peak cladding temperature can be shown not to result in significant cladding degradation.

3. Containment Conditions

Calculated maximum containment pressure should not exceed the design pressure of the containment structure. Equipment located within the containment that is relied upon to mitigate the consequences of ATWS should be qualified by testing in the combined pressure, temperature, and humidity environment conservatively predicted to occur during the course of the event.

b. Evaluation Techniques

Analysis models and techniques, including computer codes, used for conservative evaluations of the consequences of postulated ATWS events, together with associated assumptions and parameters, should be described and justified in topical reports.

c. Review of Reactor Shutdown System Design

A review of the reactor shutdown system design should be made with the aim of identifying and correcting areas that might be particularly vulnerable to common mode failures.

d. Diversity Requirements

Design changes to make the calculated consequences of postulated ATWS acceptable should not rely on equipment which has a failure mode common with the anticipated transient or the shutdown system. To the extent practical, the equipment involved in the design should operate on a different principle from equipment in the shutdown system. As a minimum, the equipment relied on to make the consequences of a postulated ATWS event acceptable should not include equipment identical to equipment in the associated shutdown system. Such designs should be shown not to result in violations of safety criteria for steady-state, transient, or accident conditions and should not adversely affect the operation of any safety-related systems.

III. REVIEW PROCEDURES

The review procedures below are used during both the CP and operating license (OL) reviews for plants for which this plan is applicable. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL stage, final values should be used in the analysis and the reviewer should compare these to the expected values, control system bands, and limiting safety system settings included in the proposed technical specifications.

The description of the ATWS events (Table 15-1 of Ref. 1) presented by the applicant in the SAR are reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which engineered safety system actuation instrumentation and controls are required to function, and the credit taken for reactor trip alarms to inform the operator of abnormal conditions.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

The applicant should present a quantitative analysis in the SAR of the ATWS events that are determined to be most limiting. The justification for selecting these events should be given. The RSB reviewer, with the aid of the EICSB reviewer, reviews the timing of the initiation of this protection, engineered safety, and other systems needed to limit the consequences of the events to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The EICSB reviewer evaluates automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation where the SAR states that operator action is needed or expected. He also verifies that the applicant has reviewed the design of the reactor shutdown system and identified and corrected areas which are vulnerable to common mode failures. Both the RSB and EICSB reviewers verify that any design changes made to mitigate the consequences of the ATWS events have no failure mode common with the cause of the anticipated transient or with the shutdown system.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam system are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients used by the applicant in his analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in Section II of this SRP regarding maximum pressure in the reactor coolant and main steam systems and fuel performance limits. For each ATWS event the variation with time during the transient of the core and barrier performance parameters listed in the "Event Evaluation" section of Chapter 15 of the Standard Format (Ref. 1) are reviewed. The more important of these parameters for ATWS events are compared to those predicted for other similar plants to verify that they are within the expected range.

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:

"A number of plant transients can be affected by a failure of the scram system to function. For a PWR* the most important include loss of feedwater, loss of load, inadvertent control rod withdrawal, and loss of a-c power. All postulated anticipated transients which can be affected by a failure to scram have been reviewed.** It was found that the most limiting ATWS events, with respect to possible fuel damage and

*For a BWR they are loss of condenser vacuum, closure of the main steam isolation valves, and turbine trip.

**The SER should present one statement for all similar transients.

pressure within the reactor coolant and main steam systems, were the ___________ and ___________ events respectively. The events were evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. The results of the analysis showed a peak reactor coolant system pressure of _____ psia which is within the 'emergency conditions' defined in Section III of the ASME Boiler and Pressure Vessel Code. The calculated (fuel clad temperature, DNBR, CPR, or MCHFR) following a postulated ___________ transient without scram is ________ and will not result in significant fuel damage. The radioactivity release from this event has been calculated and the resulting doses found to be within the guidelines of 10 CFR Part 100.

"Based on the above evaluation, the staff concludes that the plant design is acceptable with regard to anticipated transients without scram."

V. REFERENCES

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

2. Regulatory Staff, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, U. S. Atomic Energy Commission, Sept. 1973.

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-3000, "Design," American Society of Mechanical Engineers.

APPENDIX

STANDARD REVIEW PLAN 15.8

RADIOLOGICAL CONSEQUENCES OF AN ATWS EVENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Reactor Systems Branch (RSB)

I. AREAS OF REVIEW

The review of ATWS events will include estimates of the radiological consequences. These estimated consequences will be compared to those estimated for design basis events as well as the exposure guidelines of 10 CFR Part 100 and Regulatory Guides 1.3 and 1.4. If the consequence estimates for any ATWS event are significant when compared to those for design basis events or require imposition of a technical specification to limit the consequences, then that ATWS event will be included as a design basis event in the staff safety evaluation. The review method and plan are under development and will rely on the result of current staff reviews of vendor analyses of ATWS events.

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SECTION 16.0 TECHNICAL SPECIFICATIONS

REVIEW RESPONSIBILITIES

Primary - Licensing Project Manager and Standard Technical Specification Group

Secondary - Review Branches

Section 50.36 of 10 CFR of Part 50, requires that each operating license issued by the Commission contain technical specifications that set forth the limits, operating conditions, and other requirements imposed upon facility operation for the protection of the health and safety of the public. As part of the regulatory standardization effort, the staff has prepared generic standard technical specifications (STS) for each of the light water reactor nuclear steam supply system and associated balance of plant equipment systems. These STS's are subject to revision and the latest versions are available from the Office of Nuclear Reactor Regulation, NRC. The initial implementation of the STS program was made on the D. C. Cook operating license issued in October 1974. All subsequent operating licenses issued by the Commission will utilize the appropriate generic STS as the basis for issuance of Appendix A of licenses, "Technical Specifications."

Applicants should use the current generic STS as the basis for preparation of proposed Appendix A "Radiological Technical Specifications" for Section 16.0 of preliminary safety analysis reports (PSAR). The proposed Appendix A "Technical Specifications" will be reviewed to determine that the content and format are consistent with the applicable generic STS. Special attention will be given to those specifications which deviate from the generic STS to determine that proposed differences are justified on the basis of uniqueness in plant design or other considerations. Specifications so identified will be reviewed in detail to identify areas that may influence the acceptability of the final facility design. In particular, this portion of the review will determine the acceptability of proposed specifications that describe features affecting the type, capacity, number or performance of surveillance activities involving safety-related systems.

Numerical values, graphs and other data proposed will not be as complete as specified in the generic STS because of the preliminary nature of the plant design. The review of information that is provided in this area will be limited to determining that the values are in reasonable agreement with the expected operational capability of the plant.

Applicants should use the current generic STS as the bases for preparation of proposed Appendix A "Radiological Technical Specifications" for Section 16.0 of final safety analysis reports (FSAR). The Appendix A technical specifications submitted in support of an operating license will be the finalized version of those specifications originally included in the PSAR and will reflect the final refinements in design, results of tests and expected method of operation.

The review of these specifications will be done on an item by item basis to determine the comparability with the current generic STS. This comparative review will identify in a detailed manner all deviations from the generic standard in applicability, format and specific content. The noted specification differences and supporting bases will be reviewed to determine their acceptability on the basis of plant design differences or other considerations. Deviations that are found to be acceptable at this stage of review will be evaluated for suitability of incorporation into one or more of the generic STS's.

The numerical values, graphs, tables and other data proposed for each specification should be as complete as specified in the generic STS. This information will be reviewed to insure conformance with material presented in applicable portions of the FSAR as summarized in the supporting basis for each specification.

Each generic STS will be maintained current and updated periodically to reflect:

(1) changes in classes of plants or modifications of nuclear steam supply systems or balance of plant equipment systems.

(2) Revised regulatory requirements.

(3) Experience obtained by the NRC staff in reviewing proposed technical specification changes from licensees.

(4) Operational experience obtained from licensees and Office of Inspection and Enforcement inspection personnel.

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SECTION 17.1 QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

QAB will review and evaluate the design and construction quality assurance (QA) programs of the applicant and his principal contractors. Prior to docketing a construction permit (CP) application, the QAB performs a substantive review of the QA program description relative to ongoing design and procurement activities. This review is performed immediately after tendering of a CP application to determine that a satisfactory QA program; particularly in the areas of organization, program, design control, procurement document control, and audit; has been established..

A detailed CP-stage review will cover the QA controls to be applied to those activities (e.g., designing, constructing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, and testing) that may affect the quality of safety-related structures, systems, and components. This review will extend to the determination of how the applicable requirements of the eighteen criteria of Appendix B to 10 CFR Part 50 will be satisfied by the proposed QA program.

The areas of review are as follows:

1. ORGANIZATION
   a. Organizational descriptions of the lines, interrelationships, and areas of responsibility and authority for all organizations performing quality-related activities including those of the applicant, architect-engineer (A/E), nuclear steam supply system (NSSS) vendor, constructor, and construction manager when other than the constructor. Organization charts should show lines of authority.
   b. The organizational location, freedom, and authority of the individuals or groups assigned the responsibility for checking, auditing, inspecting, or verifying that an activity has been correctly performed.

c. The applicant's mechanism for maintaining responsibility for the QA program, including verification of the adequacy and implementation of the suppliers'[1/] QA programs, even for those cases where the applicant has delegated to other organizations the work of establishing and implementing the QA program or any part thereof.

d. Qualification requirements for the principal management positions in QA and quality control (QC) organizations.

2. QUALITY ASSURANCE PROGRAM

a. Identification of the structures, systems, and components which are covered by the QA program.

b. Documentation of policies, procedures, or instructions implementing the program to Appendix B requirements.

c. The measures provided to assure that suppliers meet the requirements of Appendix B.

d. Indoctrination or training programs for personnel performing quality related and QA activities.

e. Regulatory Guides and standards met by the QA program.

f. Quality related activities initiated prior to submittal of the preliminary safety analysis report (PSAR).

3. DESIGN CONTROL

a. The design control measures for:

- Translation of design bases into design documents.
- Specification of appropriate quality standards in design documents and control of deviations from these standards.
- Accessibility for inservice inspection, maintenance, and repair.
- The selection and review for suitability of application of materials, parts, equipment, and processes.
- The identification and control of design interfaces and coordination among participating design organizations.
- Verifying or checking the adequacy of design, such as by design reviews, alternate calculational methods, or testing programs.
- Assuring that design changes, including field changes, be subject to the same measures applied to the original design.

b. The organization structure, authority, and responsibilities of those positions or groups responsible for design and design verification activities.

4. PROCUREMENT DOCUMENT CONTROL

a. The procurement document control measures which assure that applicable regulatory requirements, design bases, and other QA program requirements are included or referenced in procurement documents.

b. The provisions for review and approval of procurement documents.

[1/] Supplier refers to the A/E, the NSSS vendor, the constructor, construction manager when other than the constructor, consultants performing quality-related services, and those constructors, subcontractors, and vendors providing safety-related structures, systems, components, and services.

5. INSTRUCTIONS, PROCEDURES, AND DRAWINGS

a. The means for assuring that activities affecting quality will be prescribed by and accomplished in accordance with documented instructions, procedures, or drawings.

b. Provisions for inclusion of quantitative and qualitative acceptance criteria in instructions, procedures, and drawings.

6. DOCUMENT CONTROL

a. Document control measures which assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed.

b. Control measures for obsolete or superseded documents to prevent inadvertent use.

7. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

a. Provisions for evaluation and selection of suppliers.

b. The measures for the control of purchased material, equipment, and services including provisions for assessing the adequacy of quality furnished by suppliers; for surveillance at the supplier source; and for receiving inspection.

c. Control measures taken to assure that documentary evidence of the conformance of material and equipment to procurement requirements is available at the plant site prior to installation or use.

d. Assessment by the utility of the effectiveness of the control of quality by suppliers.

8. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

a. Provisions to identify and control materials, parts, and components.

b. Measures which assure traceability of each item identified.

c. Control measures to assure that incorrect or defective items will not be used.

9. CONTROL OF SPECIAL PROCESSES

a. Control measures to assure adequate performance of special processes such as welding, heat treating, nondestructive testing, and cleaning.

b. Provisions for qualification requirements of procedures, equipment, and personnel connected with special processes.

c. Measures to assure that special processes are performed by qualified personnel using qualified procedures.

10. INSPECTION

a. Organization of the individuals or groups performing inspections, including their independence from the group performing the activity being inspected.

b. The program for the inspection of activities affecting quality, including the items and activities to be covered.

c. Provisions for preparation and use of inspection procedures, instructions, and check items.

d. Provisions for mandatory inspection hold points.

11. TEST CONTROL

a. The test program which assures that structures, systems, and components will perform satisfactorily in service.

b. Prerequisites to be provided in the written test procedures.

c. Provisions for documenting and evaluating test results.

12. CONTROL OF MEASURING AND TEST EQUIPMENT

a. The measures to assure that tools, gages, instruments, and other measuring and testing devices are properly controlled, calibrated, and adjusted at specified intervals.

b. Provisions for the identification of measuring and test equipment and identification of the corresponding calibration data.

13. HANDLING, STORAGE, AND SHIPPING

a. The measures employed to control handling, storage, shipping, cleaning, and preservation of items in accordance with work and inspection instructions to prevent damage, loss, or deterioration by environmental conditions such as temperature or humidity.

14. INSPECTION, TEST, AND OPERATING STATUS

a. The measures to indicate the inspection and test status of items to prevent inadvertent bypassing of such inspections and tests.

b. The measures for indicating the operating status of structures, systems, and components to prevent inadvertent use.

15. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

a. The measures to control nonconforming materials, parts, or components to prevent their inadvertent use or installation.

b. The methods for identification, documentation, segregation, and disposition of nonconforming items and notification to affected organizations.

16. CORRECTIVE ACTION

a. The corrective action measures established to assure that conditions adverse to quality are promptly identified and corrected.

b. Measures established to assure that the causes of significant conditions adverse to quality are determined and corrective action is taken to preclude repetition.

17. QUALITY ASSURANCE RECORDS

a. The program for the maintenance of records to furnish evidence of activities affecting quality.

b. Measures established for identification, retrieval, and retention of records.

18. AUDITS

a. The system of audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the QA program.

b. Measures established for documenting responsibilities and procedures for auditing, the required frequency of audits, documenting and reviewing audit results, and designating management levels to review and assess audit results.

II. ACCEPTANCE CRITERIA

The applicant and his principal contractors must establish a QA program for design and construction in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." To fulfill the requirements of Section 17.1 of the Standard Format (Ref. 10), this program must be described in the SAR to the extent of demonstrating how each criteria of Appendix B will be met. The detailed acceptance criteria used by the QAB in its evaluation of this program are listed in the following eighteen subsections. If the QA program meets these acceptance criteria, the program is considered in compliance with NRC regulations and is acceptable.

The Organization (17.1.1) elements responsible for the QA program are acceptable if:

1. The responsibility for the QA program is retained and exercised by the applicant.

2. The QA/QC functions, performed by the applicant's QA organization or delegated to other organizations, are identified and described, providing controls to assure all elements of Appendix B will be implemented.

3. Clear and effective lines of communication between the QA organizations of the applicant and his principal contractors are established to assure proper direction of the QA program and resolution of QA problems.

4. Organization charts identify the "onsite" and "offsite" organizational elements which function under the control of the QA program [such as Design Engineering, Procurement, Manufacturing, Construction, Inspecting (QC), Testing, and QA] and demonstrate adequate control over quality aspects within and between organizations.

5. The interface relationships and QA responsibilities of each organizational element identified in item 4 above are described and demonstrate assignment of responsibilities for requirements of Appendix B.

6. A high level of management is responsible for establishing the corporate or company QA policies, goals, and objectives and this management level maintains a continuing involvment in QA matters. Communication through any intermediate levels of management between this position and the Manager (or Director) of QA must be shown to be effective.

7. The applicant designates a position, to be filled by a qualified individual, to retain overall authority and responsibility for the QA program.

8. The authority and independence of the individual responsible for managing the QA program are such that he can direct and control the organization's QA/QC program,

can effectively assure the conformance to quality requirements, and is independent of undue influences and responsibilities for schedules and costs. An acceptable organizational structure would have this individual report to at least the same organizational level as the highest line manager directly responsible for performing activities affecting quality.

9. Positions or groups responsible for defining and controlling the content of the QA program and related manuals and the management level responsible for final review and approval have appropriate organizational position and authority.

10. The person responsible for directing and managing the QA program at the construction site has appropriate organizational position, responsibilities, and authority to exercise proper control over these functions. This individual is free of non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.

11. The qualification requirements for the principal QA/QC management positions demonstrate competence commensurate with the responsibilities of these positions.

12. Verification of conformance to established requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified.

13. Persons and organizations performing QA/QC functions have direct access to management levels which will assure accomplishment of quality-affecting activities. These personnel have sufficient authority and organizational freedom to perform their QA/QC functions effectively and without reservation. They can:
    a. Identify quality problems.
    b. Initiate, recommend, or provide solutions through designated channels.
    c. Verify implementation of solutions.

14. Designated QA individuals have the responsibility and authority, delineated in writing, to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material.

The <u>Quality Assurance Program</u> (17.1.2) description is acceptable if:

1. Measures are provided by the applicant and his principal contractors that demonstrate <u>how</u> their QA program meets 10 CFR Part 50, Appendix B criteria.

2. Management (i.e., above or outside the QA organization) regularly assesses the scope, status, implementation, and effectiveness of the QA program to assure that the program is adequate and complies with 10 CFR Part 50, Appendix B criteria.

3. Measures are provided by the applicant to assure that trained, qualified personnel within his organization are assigned to determine that functions delegated to his principal contractors are being properly accomplished.

4. A brief summary of the Company's corporate QA policies, goals, and objectives is given and a meaningful channel for transmittal of these policies, goals, and objectives down through the levels of management is established.

5. The QA program procedures are derived from QA policies, goals, and objectives.

6. QA/QC responsibilities are designated for the implementation of the major activities contained in the QA manuals.

7. Provisions are established to control the distribution of the QA manuals and revisions thereto.

8. Provisions are established for communicating to all responsible organizations and individuals that qualtiy policies, QA manuals, and procedures are mandatory requirements are procedurally controlled.

9. A listing of the QA procedures plus a matrix of these procedures cross referenced to each criterion of Appendix B to 10 CFR Part 50 demonstrates that Appendix B provisions are fully implemented by documented procedures.

10. The safety-related structures, systems, and components controlled by the QA program are identified.

11. The applicant reviews and documents agreement with the QA program provisions of his principal contractors to the extent that he can be assured that Appendix B will be implemented.

12. Provisions are established for the resolution of disputes involving quality, arising from a difference of opinion between QA/QC personnel and other department (engineering, procurement, manufacturing, etc.) personnel.

13. An indoctrination and training program is established such that:
    a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.
    b. Personnel performing quality-affecting activities are trained and qualified in the principles and techniques of the activity being performed.
    c. The scope, the objective, and the method of implementing the indoctrination and training program are documented.
    d. Proficiency of personnel performing quality-affecting activities is maintained by retraining, reexamining, and/or recertifying.

14. Quality-related activities initiated prior to the submittal of the PSAR, such as design, procurement, preparation of the PSAR, and safety-related site preparation activities are identified and controlled under a QA program which satisfies the requirements of 10 CFR Part 50, Appendix B and the guidance contained in the Regulatory Guides and ANSI standards listed in Section V of this Standard Review Plan.

15. Quality-related activities are performed with specified equipment and under suitable environmental conditions, and prerequisites have been satisfied prior to inspection and test.

16. The applicant and his principal contractors demonstrate that their QA programs comply with the Regulatory Guides (current issue) and ANSI standards listed in Section V of this Standard Review Plan or describe acceptable alternatives in equivalent detail. Additional Regulatory Guides dealing with QA whose implementation dates are prior to the SAR submittal date shall be similarly addressed.

17. A summary description of advanced planning demonstrates control of quality-related activities including management and technical interfaces between the constructor, A/E, NSSS vendor, and utility during the phaseout of design and construction and during preoperational testing and plant turnover.

18. A commitment is made to control the preoperational test program, formulated after PSAR submittal and prior to FSAR submittal, in accordance with the QA program.

19. Provisions are provided for keeping the QA program, described in the SAR, current.

20. Regulatory Guide 1.28 (Ref. 2) is complied with or acceptable alternatives are provided.

Activities related to Design Control (17.1.3) are acceptable if:

1. Measures are established to carry out design activities in a planned, controlled, and orderly manner.

2. Measures are established to correctly translate the applicable regulatory requirements and design bases into specifications, drawings, written procedures, and instructions.

3. Quality standards are specified in the design documents, and deviations and changes from these quality standards are controlled.

4. Suitable design controls are applied to such activities as reactor physics; seismic, stress, thermal, hydraulic, radiation, and accident analyses; compatibility of materials; and accessibility for inservice inspection, maintenance, and repair.

5. Designs are reviewed to assure that (1) design characteristics can be controlled, inspected, and tested and (2) inspection and test criteria are identified.

6. Internal and external design interface controls are established. These controls include the review, approval, release, distribution, and revision of documents involving design interfaces with participating design organizations.

7. Proper selection and accomplishment of design verification or checking processes such as by design reviews, alternate calculations, or qualification testing are performed. When a test program is used to verify the adequacy of a design, a qualification test of a prototype unit under adverse design conditions shall be used.

8. Individuals or groups responsible for design verification are other than the original designer and the designer's immediate supervisor.

9. Design and specification changes, including field changes, are subject to the same design controls and approvals that were applicable to the original design unless the applicant designates another qualified responsible organization.

10. Errors and deficiencies in the design, including the design process, that could adversely affect safety-related structures, systems, and components are documented; and corrective action is taken to preclude repetition.

11. Materials, parts, and equipment which are standard, commercial (off the shelf) or which have been previously approved for a different application are reviewed for suitability prior to selection.

12. The positions or groups responsible for design reviews and other design verification activities and their authority and responsibility are identified and controlled by written procedures.

13. Measures are established for the selection of suitable materials, parts, equipment, and processes for safety-related structures, systems, and components which include the use of valid industry standards and specifications.

14. Regulatory Guide 1.64 (Ref. 9) is complied with or acceptable alternatives are presented.

Activities related to <u>Procurement Document Control</u> (17.1.4) are acceptable if:

1. Procedures are established that clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of procurement documents.

2. A review and concurrence of the adequacy of quality requirements stated in procurement documents is performed by qualified personnel. This review should determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

3. The review and approval of procurement documents are documented prior to release and available for verification.

4. Procurement documents identify the applicable 10 CFR Part 50, Appendix B requirements which must be complied with and described in the supplier's QA program. This QA program or portions thereof shall be reviewed and concurred with by qualified personnel in QA prior to initiation of activities affected by the program.

5. Procurement documents contain or reference the design basis technical requirements including the applicable regulatory requirements, material and component identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions.

6. Procurement documents identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and chemical and physical test results of material) to be prepared, maintained, and submitted to the purchaser for review and approval.

7. Procurement documents identify those records to be retained, controlled, and maintained by the supplier, and those delivered to the purchaser prior to use or installation of the hardware.

8. Procurement documents contain the procuring agency's right of access to supplier's facilities and records for source inspection and audit.

9. Changes and revisions to procurement documents are subject to at least the same review and approval as the original document.

10. Procurement documents for spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

11. The requirements and guidelines of ANSI N45.2.13 (Ref. 16) are complied with or acceptable alternatives are provided.

Activities related to <u>Instructions, Procedures, and Drawings</u> (17.1.5) are acceptable if:

1. Activities affecting quality are prescribed and accomplished in accordance with documented instructions, procedures, or drawings.

2. Provisions are established which clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, and drawings.

3. Methods for complying with each of the 18 criteria of 10 CFR Part 50, Appendix B are specified in instructions, procedures, and drawings.

4. Instructions, procedures, and drawings include quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples) acceptance criteria to verify that important activities have been satisfactorily accomplished.

5. The QA organization reviews and concurs with inspection plans; test, calibration, and special process procedures; drawings and specifications; and changes thereto or acceptable alternatives are described.

Activities related to Document Control (17.1.6) are acceptable if:

1. The review, approval, and issue of documents (such as listed in item 8 below) and changes thereto, prior to release, are procedurally controlled to assure they are adequate and the quality requirements are stated.

2. Provisions are established which identify those individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto.

3. Changes to documents are reviewed and approved by the same organizations that performed the original review and approval or by other qualified responsible organizations delegated by the applicant.

4. Approved changes are included in instructions, procedures, drawings, and other documents prior to implementation of the change.

5. Obsolete or superseded documents are controlled to prevent inadvertent use.

6. Documents are available at the location where the activity will be performed prior to commencing the work.

7. A master list or equivalent is established to identify the current revision number of instructions, procedures, specifications, drawings, and procurement documents. This list is updated and distributed to predetermined, responsible personnel to preclude use of superseded documents.

8. The documents that are controlled under this subsection are identified in the PSAR. As a minimum this should include;

   a. Design specifications.
   b. Design, manufacturing, construction, and installation drawings.
   c. Procurement documents.
   d. QA manuals.
   e. PSAR and related design criteria documents.
   f. Manufacturing, inspection, and testing instructions.
   g. Test procedures.
   h. Design change requests.
   i. Nonconformance reports.

Activities related to Control of Purchased Material, Equipment, and Services (17.1.7) are acceptable if:

1. Qualified personnel evaluate the supplier's capability to provide acceptable quality services and products before the award of the procurement order or contract. The QA and engineering groups participate in the evaluation of those suppliers providing critical components.

2. The evaluation of suppliers is based on one or more of the following:

   a. The supplier's capability to comply with the elements of 10 CFR Part 50, Appendix B that are applicable to the type of material, equipment, or service being procured.
   b. A review of previous records and performance of suppliers who have provided similar articles of the type being procured.
   c. A survey of the supplier's facilities and QA program to determine his capability to supply a product which meets the design, manufacturing, and quality requirements.

3. The results of supplier evaluations are documented and filed.

4. Surveillance of suppliers during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed in accordance with written procedures to assure conformance to the purchase order requirements. These procedures provide for:

   a. Instructions that specify the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions.
   b. Audits and surveillance which assure that the supplier complies with the quality requirements. Surveillance is performed on those items where verification of procurement requirements cannot be determined upon receipt.

5. The supplier furnishes the following records as a minimum to the purchaser:

   a. Documentation that identifies the purchased material or equipment and the specific procurement requirements (e.g., codes, standards, and specifications) met by the items.
   b. Documentation that identifies any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair."

   The review and acceptance of these documents shall be described in the purchaser's QA program and as a minimum shall be undertaken by a responsible QA individual.

6. Supplier's certificates of conformance are periodically evaluated by audits, independent inspections, or tests to assure they are valid.

7. Receiving inspection of the supplier-furnished material, equipment, and services is performed to assure:

a. The material, component, or equipment is properly identified and corresponds with the identification on receiving documentation.

b. Material, components, equipments, and acceptance records are inspected and judged acceptable in accordance with predetermined inspection instructions, prior to installation or use.

c. Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment are available at the nuclear power plant prior to installation or use.

d. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

8. The effectiveness of the control of quality by suppliers is assessed by the applicant at intervals consistent with the importance, complexity, and quantity of the item.

9. Spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

10. The requirements and guidelines of ANSI N45.2.13 (Ref. 16) are complied with or acceptable alternatives are provided.

Activities related to Identification and Control of Materials, Parts, and Components (17.1.8) are acceptable if:

1. Procedures are established to identify and control materials, parts, and components including partially fabricated subassemblies.

2. Identification requirements are determined during generation of specifications and design drawings.

3. The identification and control procedures assure that identification is maintained either on the item or on records traceable to the item to preclude use of incorrect or defective items.

4. Identification of materials and parts important to the function of safety-related structures, systems, and components can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports.

5. The location and the method of identification do not affect the fit, function, or quality of the item being identified.

6. Correct identification of material, parts, and components is verified and documented prior to release for fabrication, assembling, shipping, and installation.

Activities related to Control of Special Processes (17.1.9) are acceptable if:

1. Special processes such as welding, heat treating, nondestructive testing, and cleaning are procedurally controlled.

2. Procedures, equipment, and personnel connected with special processes are qualified in accordance with applicable codes, standards, and specifications.

3. Special processes are performed by qualified personnel and accomplished in accordance with written process sheets or equivalent with recorded evidence of verification.

4. Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

5. Regulatory Guides 1.37 (Ref. 4), 1.39 (Ref. 6), and 1.54 (Ref. 7) are complied with or acceptable alternatives are provided.

Activities related to Inspection (17.1.10) are acceptable if:

1. An inspection program which verifies conformance of quality-affecting activities with requirements is established, documented, and accomplished in accordance with written controlled procedures.

2. Inspection personnel are independent from the individuals performing the activity being inspected.

3. Inspection procedures, instructions, and check lists provide for the following:

    a. Identification of characteristics and activities to be inspected.
    b. Identification of the individuals or groups responsible for performing the inspection operation.
    c. Acceptance and rejection criteria.
    d. A description of the method of inspection.
    e. Recording evidence of completing and verifying a manufacturing, inspection, or test operation.
    f. Recording inspector or data recorder and the results of the inspection operation.

4. Inspection procedures or instructions are used with necessary drawings and specifications when performing inspection operations.

5. Inspectors are qualified in accordance with applicable codes, standards, and company training programs; and their qualifications and certifications are kept current.

6. Modifications, repairs, and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives.

7. Provisions are established that identify mandatory inspection hold points for witness by an inspector.

8. The individuals or groups who perform receiving and process verification inspections at the construction site are identified and shown to have sufficient independence and qualifications.

9. Provisions are established for indirect control by monitoring processing methods, equipment, and personnel if direct inspection is not possible.

10. Regulatory Guides 1.30 (Ref. 3), 1.58 (Ref. 8), 1.94 (Ref. 13) and the requirements and guidelines of ANSI N45.2.8 (Ref. 14), and N45.2.13 (Ref. 16) are complied with or acceptable alternatives are provided.

Activities related to <u>Test Control</u> (17.1.11) are acceptable if:

1. A test program to demonstrate that the item will perform satisfactorily in service is established, documented, and accomplished in accordance with written controlled procedures.

2. Modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptable alternatives.

3. Written test procedures incorporate or reference:

    a. The requirements and acceptance limits contained in applicable design and procurement documents.
    b. Instructions for performing the test.
    c. Test prerequisites such as:
        . Calibrated instrumentation.
        . Adequate and appropriate equipment.
        . Trained, qualified, and licensed or certified personnel.
        . Completeness of item to be tested.
        . Suitable and controlled environmental conditions.
        . Provisions for data collection and storage.
    d. Mandatory inspection hold points for witness by owner, contractor, or inspector.
    e. Acceptance and rejection criteria.
    f. Methods of documenting or recording test data and results.

4. Test results are documented, evaluated, and their acceptability determined by a qualified, responsible individual or group.

5. Regulatory Guides 1.30 (Ref. 3), 1.58 (Ref. 8), and 1.94 (Ref. 13) and the requirements and guidelines of ANSI N45.2.8 (Ref. 14) are complied with or acceptable alternatives are provided.

Activities related to <u>Control of Measuring and Test Equipment</u> (17.1.12) are acceptable if:

1. Provisions, contained in procedures, describe the calibration technique and frequency, maintenance, and control of the measuring and test equipment (instruments, tools,

gages, fixtures, reference and transfer standards, and nondestructive test equipment) which is used in the measurement, inspection, and monitoring of safety-related components, systems, and structures.

2. Measuring and test equipment is identified and traceable to the calibration test data.

3. Measuring and test equipment is labeled or tagged to indicate date of the next calibration.

4. Measuring and test instruments are calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement.

5. Measures are taken and documented to determine the validity of previous inspections performed when measuring and test equipment is found to be out of calibration.

6. Calibrating standards have an uncertainty (error) requirement of no more than 1/4th of the tolerance of the equipment being calibrated. A greater uncertainty may be acceptable when limited by the "state-of-the-art."

7. The complete status of all items under the calibration system is recorded and maintained.

8. Reference and transfer standards are traceable to nationally recognized standards; or, where national standards do not exist, provisions are established to document the basis for calibration.

Activities related to Handling, Storage, and Shipping (17.1.13) are acceptable if:

1. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by qualified individuals in accordance with predetermined work and inspection instructions.

2. Procedures are prepared which control the cleaning, handling, storage, packaging, shipping, and preservation of materials, components, and systems in accordance with design and specification requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

3. Regulatory Guide 1.38 (Ref. 5) is complied with or acceptable alternatives are provided.

Activities related to Inspection, Test, and Operating Status (17.1.14) are acceptable if:

1. Identification of the inspection, test, and operating status of structures, systems, and components is known throughout manufacturing and installation.

2. The application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps are procedurally controlled.

3. Bypassing of required inspections, tests, and other critical operations is procedurally controlled under the cognizance of the QA organization.

4. The status of nonconforming, inoperative, or malfunctioning structures, systems, or components is identified to prevent inadvertent use.

Activities related to Nonconforming Materials, Parts, or Components (17.1.15) are acceptable if:

1. The identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components, or services are procedurally controlled.

2. Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition.

3. Provisions are established identifying those individuals or groups delegated the responsibility and authority for the disposition and approval of nonconforming items.

4. Nonconforming items are segregated from acceptable items and identified as discrepant until properly dispositioned.

5. Acceptability of rework or repair of materials, parts, components, systems, and structures is verified by reinspecting and retesting the item as originally inspected and tested or by a method which is at least equal to the original inspection and testing method. Inspection, testing, rework, and repair procedures are documented.

6. Nonconformance reports dispositioned "accept as is" or "repair" are made part of the inspection records and forwarded with the hardware to the utility for review and assessment.

7. Nonconformance reports are periodically analyzed to show quality trends, and the results are reported to management for review and assessment.

Activities related to Corrective Action (17.1.16) are acceptable if:

1. Evaluation of conditions adverse to quality (such as nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment) is conducted to determine the need for corrective action in accordance with established procedures.

2. Corrective action is initiated following the determination of a condition adverse to quality to preclude recurrence.

3. Follow-up reviews are conducted to verify proper implementation of corrective actions and to close out the corrective action documentation.

4. Significant conditions adverse to quality, the cause of the conditions, and the corrective action taken are reported to cognizant levels of management for review and assessment.

Activities related to Quality Assurance Records (17.1.17) are acceptable if:

1. Sufficient records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.

2. QA records include operating logs; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports; nonconformance reports; and corrective action reports.

3. Records are identifiable and retrievable.

4. Requirements and responsibilities for record transmittals, retention (such as duration, location, fire protection, and assigned responsibilities), and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents.

5. Inspection and test records contain the following where applicable:

    a. A description of the type of observation.
    b. Evidence of completing and verifying a manufacturing, inspection, or test operation.
    c. The date and results of the inspection or test.
    d. Information related to conditions adverse to quality.
    e. Inspector or data recorder identification.
    f. Evidence as to the acceptability of the results.

6. Record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flooding, theft, and deterioration by environmental conditions such as temperature or humidity.

7. Regulatory Guide 1.88 (Ref. 12) is complied with or acceptable alternatives are provided.

Activities related to Audits (17.1.18) are acceptable if:

1. Audits are performed in accordance with preestablished written procedures or check lists and conducted by trained personnel not having direct responsibilities in the areas being audited.

2. Audit results are documented and then reviewed with management having responsibility in the area audited.

3. Responsible management takes the necessary action to correct the deficiencies revealed by the audit.

4. Deficient areas are reaudited on a timely basis to verify implementation of corrective actions which minimize recurrence of deficiencies.

5. Audits include an objective evaluation of quality-related practices, procedures, and instructions and the effectiveness of implementation.

6. Audits include the objective evaluation of work areas, activities, processes, and items, and the review of documents and records.

7. Audits to assure that procedures and activities are meaningful and comply with the overall QA program are performed by:

   a. The QA organization, to provide a comprehensive independent verification and evaluation of quality-related procedures and activities.
   b. The applicant and his principal contractors, to verify and evaluate their suppliers' QA programs, procedures, and activities.

8. Provisions are established requiring that audits be performed in those areas where the requirements of Appendix B to 10 CFR Part 50 are being implemented. Areas which are often neglected include those activities associated with:

   a. The determination of site features which affect plant safety (e.g., core sampling, site and foundation preparation, and meteorology).
   b. The preparation, review, approval, and control of early procurements.
   c. Indoctrination and training programs.
   d. Interface control among the applicant and the principal contractors.

9. Audits are regularly scheduled on the basis of the status and safety importance of the activities being performed and are initiated early enough to assure effective quality assurance during the design, procurement, and contracting activities.

10. Audit data are analyzed and the reports, which indicate quality trends and the effectiveness of the QA program, are reported to management for review and assessment.

11. The requirements and guidelines of ANSI N45.2.12 (Ref. 15) are complied with or acceptable alternatives are provided.

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 acceptability of the QA program proposed for the design and construction phase is determined by the following review procedures:

1. The PSAR is reviewed in detail to determine if each of the criteria of Appendix B is addressed within the QA program description.

2. The measures described to implement Appendix B are evaluated for:

   a. Technical acceptability (i.e., do they meet the regulations and regulatory guides?).
   b. Workability (i.e., do they seem to fit into a plan of action that can be implemented?).
   c. Management support (i.e., do QA program measures have adequate review, approval, and endorsement of management?).

   This evaluation is based primarily on the acceptance criteria contained in Part II of this Standard Review Plan.

3. The duties, responsibility, and authority of personnel performing QA functions are reviewed to assure they provide sufficient independence to effectively perform these functions.

4. Through meetings with the applicant and his principal contractors and by review of the Office of Inspection and Enforcement inspection reports, a judgment is made of the applicant's capability to carry out his QA responsibilities.

5. Satisfaction with program commitments, organization arrangements, and capabilities to fulfill QA requirements will lead to the conclusion of acceptability expressed in Part IV of this Standard Review Plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review is sufficiently complete and adequate to support conclusions of the following type, which are to be included in the staff's Safety Evaluation Report:

> "Our review of the applicant's quality assurance (QA) program description for the design and construction phase has established and verified that all applicable requirements of Appendix B to 10 CFR Part 50 are included in the QA program requirements.

Further, this review established that the QA organizations are structured such that they can effectively carry out their responsibilities related to quality without undue influence from other groups.

"Based on our detailed review and evaluation of the QA program description contained in the PSAR for (nuclear facility), we conclude that:

1. The QA organizations within (Utility, A/E, NSSS Vendor, and Constructor) are provided sufficient independence from cost and schedule, when opposed to safety considerations, authority to effectively carry out the QA programs, and sufficient access to management at a level necessary to perform their QA functions.

2. The QA program describes adequate QA procedures, requirements, and controls demonstrating compliance with the requirements of Appendix B to 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (endorses N45.2).

3. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (endorses N45.2.4).

4. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (endorses N45.2.1).

5. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (endorses N45.2.2).

6. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants" (endorses N45.2.3).

7. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants" (endorses N101.4).

8. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (endorses N45.2.6).

9. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (endorses N45.2.11).

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

11. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions" (endorses N45.2.10).

12. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (endorses N45.2.9).

13. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (endorses N45.2.5).

14. ANSI N45.2.8 (Draft 3, Rev. 3 - April 1974), "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants," American National Standards Institute.

15. ANSI N45.2.12 (Draft 3, Rev. 4 - February 1974), "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," American National Standards Institute.

16. ANSI N45.2.13 (Draft 2, Rev. 4 - April 1974), "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," American National Standards Institute.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

REVIEW RESPONSIBILITIES

Primary - Quality Assurance Branch (QAB)

Secondary - None

I. AREAS OF REVIEW

QAB will review and evaluate the operational quality assurance (QA) program description of the applicant. This operating license (OL)-stage review will cover both the "offsite" and "onsite" QA controls to be applied to those activities that may affect the quality of safety-related items during the operation, maintenance, and modification of a nuclear power plant. A detailed OL-stage review will cover the QA controls to be applied to those activities (e.g., designing, constructing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, maintaining, modifying, operating, inspecting, and testing) that may affect the quality of safety-related structures, systems, and components.

This review will extend to the determination of how the applicable requirements of the eighteen criteria of Appendix B to 10 CFR Part 50 will be satisfied by the proposed QA program.

The areas of review are as follows:

1. ORGANIZATION
   a. Organizational descriptions of the lines, interrelationships, and areas of responsibility and authority for all organizations performing quality-related activities. Organization charts should show the lines of authority.
   b. The "offsite" and "onsite" organizational locations and the freedom and authority of the individuals or groups assigned the responsibility for checking, auditing, inspecting, or verifying that an activity has been correctly performed.
   c. The applicant's mechanism for maintaining responsiblity for the QA program, including verification of the adequacy and implementation of the suppliers' QA programs, even for those cases where the applicant has delegated to other organizations the work of establishing and implementing the quality assurance program or any part thereof.
   d. Qualification requirements for the principal management positions in QA/QC organizations.

2. QUALITY ASSURANCE PROGRAM

a. Identification of the structures, systems, and components covered by the QA program.

b. Documentation of policies, procedures, or instructions implementing the program to Appendix B requirements.

c. The measures provided to assure that suppliers[1/] meet the requirements of Appendix B.

d. Indoctrination or training programs for personnel performing quality-related and QA activities.

e. Regulatory Guides and standards met by the QA program.

f. Preoperational testing phase controls.

3. DESIGN CONTROL

a. The design control measures for:
- Translation of design bases into design documents.
- Specification of appropriate quality standards in design documents and control of deviations from these standards.
- Accessibility for inservice inspection, maintenance and repair.
- The selection and review of suitability of application of materials, parts, equipment, and processes.
- The identification and control of design interfaces and coordination among participating design organizations.
- Verifying or checking the adequacy of design, such as by design reviews, alternate calculational methods, or testing programs.
- Assuring that design changes, including field changes, be subject to the same measures applied to the original design.

b. The organization structure, authority, and responsibilities of those positions or groups responsible for design and design verification activities.

4. PROCUREMENT DOCUMENT CONTROL

a. The procurement document control measures which assure that applicable regulatory requirements, design bases, and other QA program requirements are included or referenced in procurement documents.

5. INSTRUCTIONS, PROCEDURES, AND DRAWINGS

a. The means for assuring that activities affecting quality will be prescribed by and accomplished in accordance with documented instructions, procedures, or drawings.

b. Provisions for inclusion of quantitative and qualitative acceptance criteria in instructions, procedures, and drawings.

[1/] Supplier refers to the A/E, the NSSS vendor, the constructor, construction manager when other than the constructor, consultants performing quality-related services, and those contractors, subcontractors, and vendors providing safety-related structures, systems, components, and services.

6. DOCUMENT CONTROL

   a. Document control measures which assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed.
   b. Control measures for obsolete or superseded documents to prevent inadvertent use.

7. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

   a. Provisions for evaluation and selection of suppliers.
   b. The measures for the control of purchased material, equipment, and services including provisions for assessing the adequacy of quality furnished by suppliers; for surveillance at the supplier source; and for receiving inspection.
   c. Control measures taken to assure that documentary evidence of the conformance of material and equipment to procurement requirements is available at the plant site prior to installation or use.
   d. Assessment by the utility of the effectiveness of the control of quality by suppliers.

8. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

   a. Provisions to identify and control materials, parts, and components.
   b. Measures which assure traceability of each item identified.
   c. Control measures to assure that incorrect or defective items will not be used.

9. CONTROL OF SPECIAL PROCESSES

   a. Control measures to assure adequate performance of special processes such as welding, heat treating, nondestructive testing, and cleaning.
   b. Provisions for qualification requirements of procedures, equipment, and personnel connected with special processes.
   c. Measures to assure that special processes are performed by qualified personnel using qualified procedures.

10. INSPECTION

   a. Organization of the individuals or groups performing inspections, including their independence from the group performing the activity being inspected.
   b. The program for the inspection of activities affecting quality including the items and activities to be covered.
   c. Provisions for preparation and use of inspection procedures, instructions, and check lists.
   d. Provisions for mandatory inspection hold points.

11. TEST CONTROL

   a. The test program which demonstrates that structures, systems, and components will perform satisfactorily in service.
   b. Prerequisites to be provided in the written test procedures.
   c. Provisions for documenting and evaluating test results.

12. CONTROL OF MEASURING AND TEST EQUIPMENT

   a. The measures to assure that tools, gages, instruments, and other measuring and testing devices are properly controlled, calibrated, and adjusted at specified intervals.

   b. Provisions for the identification of measuring and test equipment and identification of the corresponding calibration data.

13. HANDLING, STORAGE, AND SHIPPING

   a. The measures employed to control handling, storage, shipping, cleaning, and preservation of items in accordance with work and inspection instructions to prevent damage, loss, or deterioration by environmental conditions such as temperature or humidity.

14. INSPECTION, TEST, AND OPERATING STATUS

   a. The measures to indicate the inspection and test status of items to prevent inadvertent bypassing of such inspections and tests.

   b. The measures for indicating the operating status of structures, systems, and components to prevent inadvertent use.

15. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

   a. The measures to control nonconforming materials, parts, or components to prevent their inadvertent use or installation.

   b. The methods for identification, documentation, segregation, and disposition of nonconforming items and notification to affected organizations.

16. CORRECTIVE ACTION

   a. The corrective action measures established to assure that conditions adverse to quality are identified and corrected.

   b. Measures established to assure that the causes of significant conditions adverse to quality are determined and corrective action is taken to preclude repetition.

17. QUALITY ASSURANCE RECORDS

   a. The program for the maintenance of records to furnish evidence of activities affecting quality.

   b. Measures established for identification, retrieval and retention of records.

18. AUDITS

   a. The system of audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the QA program.

   b. Measures established for documenting responsibilities and procedures for auditing, the required frequency of audits, documenting and reviewing audit results, and designating management levels to review and assess audit results.

II. ACCEPTANCE CRITERIA

The applicant must establish a QA program for the operations phase, which includes operation, maintenance, and modification of the nuclear power plant, in accordance with 10 CFR Part 50,

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." To fulfill the requirements of Section 17.2 of the Standard Format (Ref. 12), this program must be described in the SAR to the extent of demonstrating how each criteria of Appendix B will be met.

The detailed acceptance criteria used by the QAB in its evaluation of this program are listed in the following 18 subsections. If the QA program for the operations phase meets these acceptance criteria, the program is considered in compliance with NRC regulations and is acceptable.

The <u>Organization</u> (17.2.1) elements responsible for the QA program are acceptable if:

1. The responsibility for the QA program is retained and exercised by the applicant.

2. The QA/QC functions, performed by the applicant's QA organization or delegated to other organizations, are identified and described, providing controls to assure all elements of Appendix B will be implemented.

3. Clear and effective lines of communication between the QA organizations of the applicant are established to assure proper direction of the QA program and resolution of QA problems.

4. Organization charts identify the "onsite" and "offsite" organizational elements which function under the control of the QA program (such as design engineering, procurement, manufacturing, construction, inspecting, testing, and QA/QC).

5. The QA responsibilities of each organizational element identified in item 4 above are described and demonstrate assignment of responsibilities for requirements of Appendix B.

6. A high level of management is responsible for establishing the corporate or company QA policies, goals, and objectives and this management level maintains a continuing involvement in QA matters. Communication through any intermediate levels of management between this position and the Manager (or Director) of QA must be shown to be effective.

7. The applicant designates a position to be filled by a qualified individual to retain overall authority and responsibility for the QA program.

8. The authority and independence of the individual responsible for managing the QA program are such that he can direct and control the organization's QA/QC program, can effectively assure the conformance to quality requirements, and is independent of undue influences and responsibilities for schedules and costs. An acceptable organization structure would have this individual report to at least the same organizational level as the highest line manager directly responsible for performing quality-affecting activities. In the case where the "onsite" QA organization reports directly to the plant superintendent, the "offsite" manager maintains an overview of onsite QA activities, through audits, surveillance, inspection, etc., as further delineated in subsequent sections.

9. Positions and groups responsible for defining the content and changes to the QA program and manual(s) and the management level responsible for the final review and approval of the QA program and manual(s) are identified.

10. The person responsible for the QA program at the plant site has appropriate organizational position, responsibilities, and authority to exercise proper control over these functions. This individual can give full attention to assuring that the QA program at the plant site is being effectively implemented. In the case where the "onsite" QA organization reports to the plant superintendent, a formal line of communication is established between the "onsite" QA organization and the "offsite" QA organization.

11. The qualification requirements for the principal QA/QC management positions demonstrate competence commensurate with the responsibilities of these positions.

12. Verification of conformance to established requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified.

13. Personnel performing QA/QC functions have direct access to management levels which will assure accomplishment of quality-affecting activities. These personnel have sufficient authority and organization freedom to perform their QA/QC functions effectively and without reservation. They can:
    a. Identify quality problems.
    b. Initiate, recommend, or provide solutions through designated channels.
    c. Verify implementation of solutions.

14. Designated QA individuals have the responsibility and authority, delineated in writing, to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material.

15. Regulatory Guide 1.28 (Ref. 4) and the requirements and guidelines of ANSI 18.7 (Ref. 16) are complied with or acceptable alternatives are provided.

The Quality Assurance Program (17.2.2) description is acceptable if:

1. Measures are provided by the applicant and, when applicable, his contractors that demonstrate how their QA program meets 10 CFR Part 50, Appendix B criteria throughout the operations phase, which includes preoperational testing, operation, maintenance, modification, and refueling.

2. Management (i.e., above or outside the QA organization) regularly assesses the scope, status, implementation, and effectiveness of the QA program to assure that the program is adequate and complies with 10 CFR Part 50, Appendix B criteria.

3. Measures are provided by the applicant to assure that trained, qualified personnel within his organization are assigned to determine that functions delegated to his principal contractors are being properly accomplished.

4. A brief summary of the corporate QA policies, goals, and objectives is given and a meaningful channel for transmittal of these policies, goals, and objectives down through the levels of management is established.

5. The QA program procedures are derived from QA policies, goals, and objectives.

6. QA/QC responsibilities are designated for the implementation of major activities contained in the QA manual.

7. Provisions are established to control the distribution of the QA manuals and revisions thereto.

8. Provisions are established for communicating to all responsible organizations and individuals that quality policies, QA manuals, and procedures are mandatory requirements which must be implemented and enforced.

9. A listing of QA procedures plus a matrix of these procedures cross referenced to each criterion of Appendix B to 10 CFR Part 50 demonstrates that Appendix B provisions are procedurally controlled.

10. The safety-related structures, systems, and components controlled by the QA program are identified.

11. The applicant reviews and documents agreement with the QA program provisions of his suppliers to the extent that he can be assured that Appendix B will be implemented.

12. Provisions are established for the resolution of disputes involving quality, arising from a difference of opinion between QA/QC personnel and other department (engineering, procurement, manufacturing, etc.) personnel.

13. An indoctrination and training program is established for those personnel, both "offsite" and "onsite," performing quality-affecting activities such that they are knowledgeable in the QA procedures and requirements and proficient in implementing these procedures. The indoctrination and training program shall assure that:
    a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.
    b. Personnel performing quality-affecting activities are trained and qualified in the principles and techniques of the activity being performed.
    c. The scope, the objective, and the method of implementing the indoctrination and training program are documented.
    d. Proficiency of personnel performing quality-affecting activities is maintained by retraining, reexamining, and/or recertifying.
    e. Methods are provided for documenting training sessions describing content; who attended; when attended; and the results of the training session.

14. Quality-related activities are performed with specified equipment under suitable environmental conditions, and prerequisites have been satisfied prior to inspection and test.

15. The applicant demonstrates that his QA program complies with the Regulatory Guides (current issue) and ANSI Standards listed in Section V of this SRP or describes acceptable alternatives to the controls contained in these documents in equivalent detail. Additional Regulatory Guides dealing with QA whose implementation dates (as identified in the Regulatory Guide) are prior to FSAR submittal shall be similarly addressed.

16. A summary description of advance planning demonstrates the control of quality-related activities including management and technical interfaces between the constructor, architect-engineer (A/E), nuclear steam supply system (NSSS) vendor, and utility during the phaseout of design and construction and during preoperational testing and plant turnover.

17. Controls are provided which assure that appropriate Appendix B requirements will be applied to the preoperational test program.

18. Provisions are provided describing how changes to the SAR QA program are identified and incorporated into the FSAR.

19. Regulatory Guides 1.8 (Ref. 3) and 1.28 are complied with or acceptable alternatives are provided.

Activities related to Design Control (17.2.3) are acceptable if:

1. Measures are established to carry out design activities in a planned, controlled, and orderly manner.

2. Measures are established to correctly translate the applicable regulatory requirements and design bases into specifications, drawings, written procedures, and instructions.

3. Quality standards are specified in the design documents, and deviations and changes from these quality standards are controlled.

4. Suitable design controls are applied to such activities as reactor physics; seismic, stress, thermal, hydraulic, radiation, and accident analyses; compatibility of materials; and accessibility for inservice inspection, maintenance, and repair.

5. Designs are reviewed to assure that (1) design characteristics can be controlled, inspected and tested, and (2) inspection and test criteria are identified.

6. Internal and external design interface controls are established. These controls include the review, approval, release, distribution, and revision of documents involving design interfaces with participating organizations.

7. Proper selection and accomplishment of design verification or checking processes such as by design reviews, alternate calculations, or qualification testing are performed. When a test program is used to verify the adequacy of a design, a qualification test of a prototype unit under adverse design conditions should be used.

8. Individuals or groups responsible for design verification are other than the original designer and the designer's immediate supervisor.

9. Design and specification changes, including those originating "onsite," are subject to the same design controls and approvals that were applicable to the original design unless the applicant designates another qualified responsible organization.

10. Errors and deficiencies in the design process including the design that could adversely affect safety-related structures, systems, and components are documented, and corrective action is taken to preclude repetition.

11. Materials, parts, and equipment which are standard, commercial (off the shelf), or which have been previously approved for a different application are reviewed for suitability prior to selection.

12. The positions or groups responsible for design reviews and other design verification activities and their authority and responsibility are identified and controlled by written procedures.

13. Measures are established for the selection of suitable materials, parts, equipment, and processes for safety-related structures, systems, and components which include the use of valid industry standards and specifications.

14. Regulatory Guide 1.64 (Ref. 11) is complied with or acceptable alternatives are provided.

Activities related to Procurement Document Control (17.2.4) are acceptable if:

1. Procedures are established that clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of procurement documents.

2. A review and concurrence of the adequacy of quality requirements stated in procurement documents is performed by qualified personnel. This review should determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

3. The review and approval of procurement documents are documented prior to release and are available for verification.

4. Procurement documents identify the applicable 10 CFR Part 50, Appendix B requirements which must be complied with and described in the supplier's QA program. This QA program or portions thereof shall be reviewed and concurred with by qualified personnel in QA prior to initiation of activities affected by the program.

5. Procurement documents contain or reference the design basis technical requirements including the applicable requlatory requirements, material and component identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions.

6. Procurement documents identify the documentation (e.g., drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications, and chemical and physical test results of material) to be prepared, maintained, and submitted, to the purchaser for review and approval.

7. Procurement documents identify those records to be retained, controlled, and maintained by the supplier and those delivered to the purchaser prior to use or installation of the hardware.

8. Procurement documents contain the procuring agency's right of access to supplier's facilities and records for source inspection and audits.

9. Changes and revisions to procurement documents are subject to at least the same review and approval as the original document.

10. Procurement documents for spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

11. The requirements and guidelines of ANSI N45.2.13 (Ref. 19) are complied with or acceptable alternatives are provided.

Activities related to Instructions, Procedures, and Drawings (17.2.5) are acceptable if:

1. Activities affecting quality are prescribed and accomplished in accordance with documented instructions, procedures, or drawings.

2. Provisions are established which clearly delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, and drawings.

3. Methods for complying with each of the 18 criteria of 10 CFR Part 50, Appendix B are specified in instructions, procedures, and drawings.

4. Instructions, procedures, and drawings include quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples) acceptance criteria to verify that important activities have been satisfactorily accomplished.

5. The "offsite" or "onsite" QA organization reviews and concurs with inspection plans; test, calibration, special process, maintenance, modification and repair procedures; drawings and specifications; and changes thereto or acceptable alternatives are provided. In the case where the "onsite" QA organization reports to the plant superintendent and reviews and concurs in the procedures, the "offsite" QA organization reviews these procedures thru surveillance and audit to assure they meet QA program requirements.

Activities related to Document Control (17.2.6) are acceptable if:

1. The review, approval, and issue of documents (such as listed in Item 8 below) and changes thereto, prior to release, are procedurally controlled to assure they are adequate and the quality requirements are stated.

2. Provisions are established which identify those individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto.

3. Changes to documents are reviewed and approved by the same organizations that performed the original review and approval or by other qualified responsible organizations delegated by the applicant.

4. Approved changes are included in instructions, procedures, drawings, and other documents prior to implementation of the change.

5. Obsolete or superseded documents are controlled to prevent inadvertent use.

6. Documents are available at the location where the activity will be performed prior to commencing the work.

7. A master list or equivalent is established to identify the current revision number of instructions, procedures, specifications, drawings, and procurement documents. This list is updated and distributed to predetermined, responsible personnel to preclude the use of superseded documents.

8. The documents that are controlled under this subsection are identified. As a minimum this should include:
   a. Design specifications.
   b. Design, manufacturing, construction, and installation drawings.
   c. Procurement documents.
   d. QA manual and maintenance, modification and operating procedures.
   e. Final safety analysis report (FSAR).
   f. Manufacturing, inspection, and testing instructions.
   g. Test procedures.

h. Design change requests.
i. Nonconformance reports.

Activities related to Control of Purchased Material, Equipment, and Services (17.2.7) are acceptable if:

1. Qualified personnel evaluate the suppliers' capability to provide acceptable quality services and products before the award of the procurement order or contract. The QA and engineering groups participate in the evaluation of those suppliers providing critical components.

2. The evaluation of suppliers is based on one or more of the following:
   a. The supplier's capability to comply with the elements of 10 CFR Part 50, Appendix B that are applicable to the type of material, equipment, or service being procured.
   b. A review of previous records and performance of suppliers who have provided similar articles of the type being procured.
   c. A survey of the supplier's facilities and QA program to determine his capability to supply a product which meets the design, manufacturing, and quality requirements.

3. Results of supplier evaluations are documented and filed.

4. Surveillance of suppliers during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed in accordance with written procedures to assure conformance to the purchase order requirements. These procedures provide for:
   a. Instructions that specify the characteristics or processes to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions.
   b. Audits and surveillance which assure that the supplier complies with the quality requirements. Surveillance should be performed on those items where verification or procurement requirements cannot be determined upon receipt.

5. The supplier furnishes the following records as a minimum to the purchaser:
   a. Documentation that identifies the purchased material or equipment and the specific procurement requirements (e.g., codes, standards, specifications) met by the items.
   b. Documentation that identifies any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair."

   The review and acceptance of these documents shall be described in the purchaser's QA program, and as a minimum, shall be undertaken by a responsible QA individual.

6. Supplier's certificates of conformance are periodically evaluated by audits, independent inspections, or tests to assure they are valid.

7. Receiving inspection of the supplier-furnished material, equipment, and services is performed to assure:
   a. The material, component, or equipment is properly identified and corresponds with the identification on the receiving documentation.
   b. Material, components or equipment, and acceptance records are inspected and judged acceptable in accordance with predetermined inspection instructions, prior to installation or use.
   c. Inspection records or certificates of conformance attesting to the acceptance of material, components, and equipment are available at the nuclear power plant prior to installation or use.
   d. Items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

8. The effectiveness of the control of quality by suppliers is assessed by the applicant at intervals consistent with the importance, complexity, and quantity of the item.

9. Spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

Activities related to Identification and Control of Materials, Parts, and Components (17.2.8) are acceptable if:

1. Procedures are established to identify and control materials, parts, and components including partially fabricated subassemblies.

2. The identification and control procedures assure that identification is maintained either on the item, or on records traceable to the item, to preclude the use of incorrect or defective items.

3. Identification of materials and parts important to the function of safety-related structures, systems, and components can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical mill test reports.

4. The location and the method of identification do not affect the fit, function, or quality of the item being identified.

5. Correct identification of materials, parts, and components is verified and documented prior to release for fabrication, assembling, shipping, and installation.

Activities related to Control of Special Processes (17.2.9) are acceptable if:

1. Special processes such as welding, heat treating, nondestructive testing, and cleaning are controlled.

2. Procedures, equipment, and personnel connected with special processes are qualified in accordance with applicable codes, standards and specifications.

3. Special processes are performed by qualified personnel and accomplished in accordance with written process sheets, or equivalent with recorded evidence of verification.

4. Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

5. Regulatory Guides 1.37 (Ref. 6), 1.39 (Ref. 8) and 1.54 (Ref. 9) are complied with or acceptable alternatives are provided.

Activities related to Inspection (17.2.10) are acceptable if:

1. An inspection program which verifies conformance of quality affecting activities with requirements is established, documented, and accomplished in accordance with written controlled procedures.

2. Inspection personnel are independent from the individuals performing the activity being inspected.

3. Inspection procedures, instructions, and check lists provide for the following:
   a. Identification of characteristics and activities to be inspected.
   b. Identification of the individuals or groups responsible for performing the inspection operation.
   c. Acceptance and rejection criteria.
   d. A description of the method of inspection.
   e. Recording evidence of completing and verifying a manufacturing, inspection, or test operation.
   f. Recording the inspector or data recorder and the results of the inspection operation.

4. Inspection procedures or instructions are used with necessary drawings and specifications when performing inspection operations.

5. Modifications, repairs, and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives.

6. Provisions are established that identify mandatory inspection hold points for witness by an inspector.

7. Provisions are established for indirect control by monitoring processing methods, equipment, and personnel if direct inspection is not possible.

8. Inspectors are qualified in accordance with applicable codes, standards, and company training programs, and their qualifications and certifications are kept current.

9. Maintenance and modification procedures are reviewed by qualified personnel knowledgeable in QA to determine the need for a) inspection, b) identification of inspection personnel, and c) documenting inspection results.

10. Regulatory Guides 1.30 (Ref. 5), 1.58 (Ref. 10), 1.94 (Ref. 15), and the requirements and guidelines of ANSI N45.2.13 (Ref. 19) are complied with or acceptable alternatives are provided.

Activities related to Test Control (17.2.11) are acceptable if:

1. A test program (proof, preoperational, and operational tests) to demonstrate that the item will perform satisfactorily in service is established, documented, and accomplished in accordance with written controlled procedures.

2. Modifications, repairs, and replacements are tested in accordance with the original design and testing requirements or acceptable alternatives.

3. Written test procedures incorporate or reference:
   a. The requirements and acceptance limits contained in applicable design and procurement documents.
   b. Instructions for performing the test.
   c. Test prerequisites such as:
      - Calibrated instrumentation.
      - Adequate and appropriate equipment.
      - Trained, qualified, and licensed or certified personnel.
      - Completeness of item to be tested.
      - Suitable and controlled environmental conditions.
      - Provisions for data collection and storage.
   d. Mandatory inspection hold points for witness by owner, contractor, or inspector.
   e. Acceptance and rejection criteria.
   f. Methods of documenting or recording test data and results.

4. Test results are documented, evaluated, and their acceptability determined by a qualified, responsible individual or group.

5. Regulatory Guides 1.30 (Ref. 5), 1.58 (Ref. 10), 1.94 (Ref. 15) and the requirements and guidelines of ASNI N45.2.8 (Ref. 17) are complied with or acceptable alternatives are provided.

Activities related to Control of Measuring and Test Equipment (17.2.12) are acceptable if:

1. Provisions contained in procedures establish the calibration technique and frequency, maintenance, and control of all measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) which are used in the measurement, inspection, and monitoring of safety-related components, systems, and structures.

2. Measuring and test equipment is identified and traceable to the calibration test data.

3. Measuring and test equipment is labeled or tagged to indicate the date of the next calibration.

4. Measuring and test instruments are calibrated at specified intervals based on the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement.

5. Measures are taken and documented to determine the validity of previous inspections performed when measuring and test equipment is found to be out of calibration.

6. Calibrating standards have an uncertainty (error) requirement of no more than 1/4 of the tolerance of the equipment being calibrated. A greater uncertainty may be acceptable when limited by the "state-of-the-art."

7. The complete status of all items under the calibration system is recorded and maintained.

8. Reference and transfer standards are traceable to nationally recognized standards; or, where national standards do not exist, provisions are established to document the basis for calibration.

Activities related to Handling, Storage, and Shipping (17.2.13) are acceptable if:

1. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by qualified individuals in accordance with predetermined work and inspection instructions.

2. Procedures are prepared which control the cleaning, handling, storage, packaging, shipping, and preservation of materials, components and systems in accordance with design and specification requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity.

3. Regulatory Guide 1.38 (Ref. 7) is complied with or acceptable alternatives are provided.

Activities related to Inspection, Test, and Operating Status (17.2.14) are acceptable if:

1. Identification of the inspection, test, and operating status of structures, systems, and components is known by affected organizations.

2. The application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps are procedurally controlled.

3. Bypassing of required inspections, tests, and other critical operations is procedurally controlled under the cognizance of the QA organization.

4. The status of operating, nonconforming, inoperative, or malfunctioning structures, systems, or components is identified to prevent inadvertent use.

Activities related to <u>Nonconforming Materials, Parts, or Components</u> (17.2.15) are acceptable if:

1. The identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components, or services are procedurally controlled.

2. Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconformance, and the inspection requirements; and includes signature approval of the disposition.

3. Provisions are established identifying those individuals or groups delegated the responsibility and authority for the disposition and approval of nonconforming items.

4. Nonconforming items are segregated from acceptable items and identified as discrepant until properly dispositioned.

5. Acceptability of rework or repair of materials, parts, components, systems, and structures is verified by reinspecting and retesting the item as originally inspected and tested or by a method which is equivalent to the original inspection and testing method; and inspection, testing, rework, and repair procedures are documented.

6. Nonconformance reports dispositioned "accept as is" or "repair" are made part of the inspection records and forwarded with the hardware to the utility for review and assessment.

7. Nonconformance reports are periodically analyzed to show quality trends, and the results are reported to management for review and assessment.

Activities related to <u>Corrective Action</u> (17.2.16) are acceptable if:

1. Evaluation of conditions adverse to quality (such as nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment) is conducted to determine need for corrective action in accordance with established procedures.

2. Corrective action is initiated following the determination of a condition adverse to quality to preclude recurrence.

3. Follow-up reviews are conducted to verify proper implementation of corrective actions and to close out the corrective action documentation.

4. Significant conditions adverse to quality, the cause of the conditions, and the corrective action taken are reported to cognizant levels of both "offsite" and "onsite" management including QA, for review and assessment.

Activities related to Quality Assurance Records (17.2.17) are acceptable if:

1. Sufficient records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.

2. QA records include plant history; operating logs; principal maintenance and modification activities; abnormal occurrences; results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports; nonconformance reports, and corrective action reports.

3. Records are identifiable and retrievable.

4. Requirements and responsibilities for record transmittals, retention (such as duration, location, fire protection and assigned responsibilities), and maintenance subsequent to completion of work are consistent with applicable codes, standards, and procurement documents.

5. Inspection and test records contain the following where applicable:
   a. A description of the type of observation.
   b. Evidence of completing and verifying a manufacturing, inspection, or test operation.
   c. The date and results of the inspection or test.
   d. Information related to conditions adverse to quality.
   e. Inspector or data recorder identification.
   f. Evidence as to the acceptability of the results.

6. Record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flooding, theft, and deterioration by environmental conditions such as temperature or humidity.

7. Regulatory Guide 1.88 (Ref. 14) is complied with or acceptable alternatives are provided.

Activities related to Audits (17.2.18) are acceptable if:

1. Audits are performed in accordance with preestablished written procedures or check lists and conducted by trained personnel not having direct responsibilities in the areas being audited. Where the "onsite" QA organization reports to the plant superintendent, the "offsite" QA organization conducts audits sufficient to verify adequacy of activities conducted by the "onsite" QA organization.

2. Audits results are documented and then reviewed with management having responsibility in the areas audited.

3. Responsible management takes the necessary action to correct the deficiencies revealed by the audit.

4. Deficient areas are reaudited on a timely basis to verify implementation of corrective actions which minimize recurrence of the deficiencies.

5. Audits include an objective evaluation of quality-related practices, procedures, and instructions and the effectiveness of implementation.

6. Audits include the objective evaluation of work areas, activities, processes, and items and the review of documents and records.

7. Audits to assure the procedures and activities are meaningful and comply with the overall QA program are performed by:
   a. The QA organization to provide a comprehensive independent verification and evaluation of quality-related procedures and activities. Results of audits performed by the "onsite" QA organization are provided to the "offsite" QA organization for review and assessment.
   b. The applicant and his principal contractors to verify and evaluate their suppliers' QA programs, procedures, and activities.

8. Provisions are established requireing that audits be performed in those areas where the requirements of Appendix B to ;0 CFR Part 50 being implemented. These areas shall include those safety-related activities associated with:
   a. Operation, maintenance and modification.
   b. The preparation, review, approval, and control of designs, specifications, procurement documents, instructions, procedures, and drawings.
   c. Receiving and plant inspections.
   d. Indoctrination and training programs.
   e. The implementation of operating and test procedures.
   f. Calibration of measuring and testing equipment.

9. Audits are regularly scheduled on the basis of the status and safety importance of the activities being performed. Where the "onsite" QA organization reports to the plant superintendent, the "offsite" QA organization reviews and concurs in the schedule and scope of audits performed by the "onsite" QA organization.

10. Audit data are analyzed and the reports, which indicate quality trends and the effectiveness of the QA program, are reported to management for review and assessment.

11. The requirements and guidelines of ANSI N45.2.12 (Ref. 18) are complied with or acceptable alternatives are provided.

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 acceptability of the QA program proposed for the operations phase is determined by the following review procedure:

1. The FSAR is reviewed in detail to determine if each of the criteria of Appendix B is addressed within the QA program description.

2. The measures described to implement Appendix B are evaluated for:
   a. Technical acceptability (i.e., do they meet the regulations and regulatory guides?).
   b. Workability (i.e., do they seem to fit into a plan of action that can be implemented?).
   c. Management support (i.e., do QA program measures have adequate review, approval, and endorsement of management?).

   This evaluation is based primarily on the acceptance criteria contained in Part II of this Standard Review Plan.

3. The duties, responsibility, and authority of personnel performing QA functions are reviewed to assure they provide sufficient independence to effectively perform these functions.

4. Through meetings with the applicant and by review of the Office of Inspection and Enforcement inspection reports, a judgment is made of the applicant's capability to carry out his QA responsibilities.

5. Satisfaction with program commitments, organization arrangements, and capabilities to fulfill QA requirements will lead to the conclusion of acceptability expressed in Part IV of this Standard Review Plan.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that his review is sufficiently complete and adequate to support conclusions of the following type, which are to be included in the staff's Safety Evaluation Report:

"Our review of the applicant's quality assurance (QA) program description for the operations phase has verified that all applicable requirements of Appendix B to 10 CFR Part 50 are included in the QA program requirements. Further, this review has determined that the QA organizations are structured such that they can effectively carry out their responsibilities related to quality without undue influence from other groups.

"Based on our detailed review and evaluation of the QA Program description contained in the FSAR for (nuclear facility), we conclude that:

1. The QA organization of the (utility) is provided sufficient independence from cost and schedule when opposed to safety considerations, authority to effectively carry out the QA programs, and sufficient access to management at a level necessary to perform their QA functions.

2. The QA program describes adequate QA procedures, requirements, and controls demonstrating compliance with the requirements of Appendix B to 10 CFR Part 50."

V. REFERENCES

1. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. 10 CFR Part 55, "Operators' Licenses."

3. Regulatory Guide 1.8, "Personnel Selection and Training" (endorses N18.1).

4. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)" (endorses N45.2).

5. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (endorses N45.2.4).

6. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (endorses N45.2.1).

7. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (endorses N45.2.2).

8. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants" (endorses N45.2.3).

9. Regulatory Guide 1.54, "Quality Assuarance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants" (endorses N101.4).

10. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (endorses N45.2.6).

11. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (endorses N45.2.11).

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

13. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions" (endorses N45.2.10).

14. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (endorses N45.2.9).

15. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (endorses N45.2.5).

16. ANSI N18.7 (latest revision), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," American National Standards Institute.

17. ANSI N45.2.8 (Draft 3, Rev. 3 - April 1974), "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants," American National Standards Institute.

18. ANSI N45.2.12 (Draft 3, Ref. 4 - February 1974), "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," American National Standards Institute.

19. ANSI N45.2.13 (Draft 2, Rev. 4 - April 1974), "Quality Assurance Requirements for Contro of Procurement of Items and Services for Nuclear Power Plants," American National Standards Institute.