United States
Nuclear Regulatory Commission

Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors

Format and Content

February 1996

NUREG - 1537 PART 1

Office of Nuclear Reactor Regulation
Division of Reactor Program Managment

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Commission

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NUREG-1537, PART 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 6, 1996

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR TEST AND RESEARCH REACTORS

SUBJECT: ISSUANCE OF NUREG-1537, "GUIDELINES FOR PREPARING AND REVIEWING APPLICATIONS FOR THE LICENSING OF NON-POWER REACTORS"

The U.S. Nuclear Regulatory Commission (NRC) has issued NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." Part 1 of NUREG-1537 contains format and content guidance for non-power reactor (NPR) applicants and licensees and Part 2 contains a standard review plan and acceptance criteria for NRC NPR reviewers.

The format and content guide suggests a uniform format for presenting information in NPR applications, helps ensure completeness of information provided, assists the Commission staff and others in locating information, and aids in increasing the efficiency of the review process. The format and content guide represents a format for NPR applications that is acceptable to the NRC staff. Conformance with the format and content, however, is not required.

The standard review plan ensures the quality and uniformity of the staff reviews, makes information about regulatory matters concerning NPRs widely available, and improves the understanding of the staff review process by interested members of the NPR community and the public.

The document covers all aspects of NPR licensing. The document can be used for the construction permit and the initial operating license, license renewal, license amendment, decommissioning and license termination, and highly enriched to low-enriched uranium core conversions. There is also an appendix to the format and content guide that lists selected regulations that are applicable to NPRs.

The document chapters were released in draft form for public comment as the staff completed them. The staff evaluated comments from interested parties and a number of them were incorporated into the document. The comments did not result in any major changes to the document, but the staff used them to clarify the documents. The draft documents were changed in response to comments and the documents were edited to make the writing style consistent between chapters. The staff responded to everyone who commented on the draft documents by sending commenters the staff analysis of their comments and pointing out any changes made to the text as a result of the comments. Comments and NRC responses were placed in the Public Document Room.

The document is in looseleaf form to be put in three-hole binders for ease of use. We plan to amend the document as necessary to keep its content current. Your comments on NUREG-1537 are encouraged. They should be sent to--

Director, Non-Power Reactors and Decommissioning
Project Directorate
United States Nuclear Regulatory Commission
M. S. O-11 B-20
Washington, DC 20555-0001

Comments will be considered in future revisions of the document. Questions concerning this project should be directed to the project manager for this effort, Alexander Adams, Jr., at 301-415-1127.

Seymour H. Weiss, Director
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

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UNITED STATES
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Seymour H. Weiss, Director
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

ABSTRACT

NUREG-1537, Part 1 gives guidance to non-power reactor licensees and applicants on the format and content of applications to the Nuclear Regulatory Commission for licensing actions. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEA	Atomic Energy Act of 1954, as amended
AEC	Atomic Energy Commission
AGN	Aerojet-General Nucleonics
ALARA	as low as is reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASLAB	Atomic Safety and Licensing Appeal Board
ATR	advanced test reactor
BNCT	boron neutron capture therapy
BSF	bulk shielding facility
CAM	continuous air monitor(ing)
CFR	*Code of Federal Regulations*
CP	construction permit
CP-5	Chicago Pile #5
CP-11	Chicago Pile #11
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DOT	Department of Transportation
DP	decommissioning plan
EA	environmental assessment
EC	effluent concentration
ECCS	emergency core cooling system
ER	environmental report
ESF	engineered safety feature
EPRI	Electric Power Research Institute
EPZ	emergency preparedness zone
FDA	Food and Drug Administration
FLIP	fuel lifetime improvement program
FR	*Federal Register*
FSAR	final safety analysis report
GA	General Atomics
HEU	highly enriched uranium
HVAC	heating, ventilation, and air conditioning

IAEA	International Atomic Energy Agency
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronic Engineers
INEL	Idaho National Engineering Laboratory
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LOFA	loss-of-flow accident
LSSS	limiting safety system setting
LWR	light-water reactor
MHA	maximum hypothetical accident
MIT	Massachusetts Institute of Technology
MPC	maximum permissible concentration
MTR	materials testing reactor
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUMARC	Nuclear Management and Resources Council
ORNL	Oak Ridge National Laboratory
ORRR	Oak Ridge Research Reactor
OSHA	Occupational Safety and Health Administration
OWR	Omega West Reactor
PSAR	preliminary safety analysis report
PULSTAR	pulsed training assembled reactor
QA	quality assurance
RAM	remote area monitor
RCS	reactor control system
RG	regulatory guide
RO	reactor operator
RPS	reactor protection system
SAR	safety analysis report
SI	Internation System of Units
SL	safety limit
SNM	special nuclear material
SNAP	system for nuclear auxiliary power
SPERT	special power excursion reactor test
SRO	senior reactor operator

TLD	thermmoluminescence dosimeter
TRIGA	training reactor, isotope [production], General Atomics
TS	technical specification
USGS	U.S. Geological Survey
UTM	Universal Transverse Mercator

INTRODUCTION

Background

This document describes acceptable format and content of the safety analysis report (SAR) to be submitted to the U.S. Nuclear Regulatory Commission (NRC) by an applicant or licensee of a non-power reactor for a new license, license renewal, or license amendment. A companion document, NUREG-1537, Part 2 (Standard Review Plan), gives criteria to assist NRC staff reviewers in effecting comparable, complete, and consistent reviews of licensing applications for non-power reactors. Applicants could peruse the Standard Review Plan to gain further insight into the review process for finding non-power reactor applications acceptable.

NRC published several documents that give guidance that is applicable to commercial power reactors. In 1972, to help commercial power applicants prepare SARs for operating licenses for power reactors, NRC clarified the format and content of SARs for light-water reactors (LWRs) by issuing Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," with revisions in 1973, 1975, and 1978. In 1975, NRC issued NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," to assist the NRC staff in effecting comparable, complete, and consistent reviews of SARs for nuclear power plants. In 1982, the staff completely revised the earlier Standard Review Plan (NUREG-75/087) and published the revision as NUREG-0800. In 1987, the staff revised NUREG-0800.

These documents were developed specifically for LWR nuclear power plants. Applicants who would use these documents to prepare SARs for non-power reactor facilities and NRC staff who would use them to review these SARS may find it very cumbersome because of the great differences in complexity and hazards between non-power reactors and nuclear power plants. Thus, the NRC staff started this program to document guidance applicable to non-power reactors. The guidance herein is based on the *Code of Federal Regulations*, Title 10, Section 50.34 (10 CFR 50.34) which describes the information to be supplied in a SAR.

All reactors, both power and non-power, are licensed to operate as utilization facilities under Title 10 in accordance with the *Atomic Energy Act* of 1954, as amended (AEA or Act). The AEA was written to promote the development and use of atomic energy for peaceful purposes and to control and limit its radiological hazards to the public. These purposes are expressed in paragraph 104 of the Act for non-power reactors, which states that utilization facilities for research and development should be regulated to the minimum extent consistent with protecting the health and safety of the public and promoting the common defense and

security. These concepts are promulgated in 10 CFR 50.40 and 50.41, and in other parts of Title 10 that deal with non-power reactors. The licensed thermal power levels of non-power reactors are several orders of magnitude lower than current power reactors. Therefore, the accumulated inventory of radioactive fission products in the fuel (in core) of non-power reactors is proportionally less than power reactors and requires less stringent and less prescriptive measures to give equivalent protection to the health and safety of the public. Thus, even though many of the regulations of Title 10 apply to both power and non-power reactors, the regulations may be implemented in a different way for each category of reactor and are intended to be consistent with protecting the health and safety of the public. Because the potential hazards may also vary widely among non-power reactors, regulations also may be implemented in a different way within the non-power reactor category.

Sections 50.20 through 50.22 of Title 10 specify two classes of reactor licenses to be issued to applicants by the NRC: Class 104 (medical therapy and research and development facilities) and Class 103 (commercial and industrial facilities). These classes derive from definitions in the AEA. Non-power reactors are designed and operated for medical therapy, research, development, and education. Non-power reactors consist of testing facilities (also called "test reactors" in some regulations) which are defined in 10 CFR 50.2, and research reactors, which are defined in 10 CFR 170.3.

Currently, all non-power reactors are licensed as Class 104 facilities. However, NRC recognizes that a non-power reactor for commercial purposes could be licensed as a Class 103 facility, and thus, 10 CFR 50.22 contains criteria for judging if a non-power reactor is a Class 103 facility.

A Class 104 non-power reactor can be licensed as a Class 104a facility for conducting medical therapy or as a Class 104c facility for conducting research and development. One non-power reactor is licensed as both a Class 104a and 104c facility. All other non-power reactors are licensed as Class 104c facilities.

Most of the design, operation, and safety considerations for non-power reactors apply to both test and research reactors. All non-power reactor applicants should be guided by the format and content for licensing applications in this document. Test reactors are subject to additional requirements, such as preparation of an environmental impact statement, conduct of licensing hearings, and review by the Advisory Committee on Reactor Safeguards (ACRS).

The issue of what standards to use in evaluating accidents at a research reactor was discussed in an Atomic Safety and Licensing Appeal Board (ASLAB) decision issued May 18, 1972, for the research reactor at Columbia University in New York City. ASLAB stated that "as a general proposition, the Appeal Board does not

consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The staff has not found it necessary to conform to that recommendation to develop separate criteria for the evaluation of research reactor accidents, since most research reactor accidents evaluated to date have been within the 10 CFR Part 20 criteria.

The principal safety issues that differentiate test reactors from research reactors are the reactor site requirements and the doses to the public that could result from a serious accident. For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 (10 CFR 20.1 through 20.602 and Appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and Appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results for research reactors are less than 5 rem whole body and less than 30 rem thyroid for occupational exposure, and less than 0.5 rem whole body and less than 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analyses that exceed the 10 CFR Part 20 dose limits discussed above.

If the facility conforms to the definition of a test reactor, the doses should be compared with 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses given in 10 CFR Part 100 are reference values. Any further references to 10 CFR Part 100 in this document apply to test reactors only.

The hazards from non-power reactors, compared with power reactors, range from small to insignificant. After licensing almost 150 non-power reactors, the NRC staff has developed guidelines and criteria for use in concluding that a facility, function, or procedure provides reasonable assurance that the public will not receive a radiation dose that exceeds regulatory limits.

The regulations in 10 CFR 2.105(c) for the initial licensing of a research reactor facility do not preclude a joint application for a construction permit and the initial operating license. If well planned, the final facility design and the final SAR descriptions, analyses, and conclusions will not differ significantly from those in the initial application, and a one-step licensing procedure can be undertaken. To initiate this process, the applicant should request both a construction permit and an operating license to be issued when construction and operating readiness are acceptable to NRC. The applicant should submit only one SAR that is complete,

appropriate, and acceptable for both permits. This will enable NRC to publish a joint notice of intent in the *Federal Register* at the construction permit stage that includes issuance of the operating license when appropriate. The joint application and joint notice procedure streamlines the licensing process. If a final SAR documenting changes during construction is submitted, it must demonstrate that the facility design and the safety conclusions of the previous SAR documentation are unchanged.

The design information in an SAR should reflect the current state of the facility design, or the current as-built system at the time of the submittal. If certain information noted herein is not yet available because the design has not progressed sufficiently, the SAR should contain (1) the criteria and design bases used to develop the required information, (2) the concepts and alternatives under consideration, and (3) the schedule for completing the design and submitting the missing information. The SAR for a new facility should describe the current design of the facility in sufficient detail to enable the reviewers to determine whether or not the facility can be constructed and operated in accordance with applicable regulations.

The licensing process conforms to the legislative requirement for minimum regulation stated in Section 104 of the AEA. A license for facility operation constitutes the legal agreement between the licensee and NRC, and both parties must adhere to it rigorously. Quite often, because of applicant choice, the licensing process leads to two or more facilities with the same type of fuel and the same intrinsic safety limits being licensed for operation at maximum power levels differing by at least a factor of 10. The resultant difference between licensed operating conditions and safety limits may vary by at least an order of magnitude. However, each facility is obligated to adhere to its own license conditions.

Document Structure

Parts 1 and 2 of this document are complementary; titles and numbers of sections correspond to the SAR sections.

The structure of the document is summarized below. The general requirements of the safety analysis are presented along with information on the purpose, applicability, and use of this document.

Chapter 1 summarizes the principal design bases and considerations, general descriptions of the reactor facility that illustrate the anticipated operations, and the design safety considerations, including the limiting potential accidents. This chapter should summarize the detailed information found in subsequent chapters of the SAR.

Chapter 2 describes the bases for the site selection and describes the applicable site characteristics, including geography, demography, meteorology, hydrology, geology, seismology, and interaction with nearby installations and facilities.

Chapter 3 describes the design bases and facility structures, systems, and components, and the responses to environmental factors on the reactor site (e.g., floods).

Chapter 4 describes the design bases and the functional characteristics of the reactor core and its components. In this chapter, the safety considerations and features of the reactor are discussed.

Chapter 5 lists the design bases and describes the functions of the reactor coolant and associated systems at the facility, including the primary and secondary systems as applicable, and coolant makeup and purification systems. The chapter also describes provisions for adequate heat removal while the reactor is operating and while it is shut down.

Chapter 6 lists the design bases and describes the functions of engineered safety features (ESFs) that may be required to mitigate consequences of postulated accidents at the facility. This includes design-basis accidents and a maximum hypothetical accident (MHA). The MHA, which assumes an incredible failure that can lead to fuel cladding or to a fueled experiment containment breach, is used to bound credible accidents in the accident analysis.

Chapter 7 lists the design bases and describes the functions of the instrumentation and control systems and subsystems at the facility, placing emphasis on safety-related systems and safe reactor shutdown.

Chapter 8 lists the design bases and describes the functions of the normal and emergency (if applicable) electrical power systems at the facility.

Chapter 9 lists the design bases and describes the functions of such auxiliary systems at the facility as heating, ventilation, air exhaust, air conditioning, service water, compressed air, and fuel handling and storage.

Chapter 10 lists the design bases and describes the functions of experimental facilities. Non-power reactors are designed with irradiation capabilities for research, education, and technological development. This chapter discusses the characteristics of experiment and irradiation facilities on the basis of the proposed experimental programs.

Chapter 11 lists the design bases and describes the functions of the radiation protection and the radioactive waste management programs at the facility. This

chapter also describes the control of byproduct materials produced in the reactor and utilized under the 10 CFR Part 50 reactor operating license. The description of the radiation protection program should include health physics procedures, monitoring programs for personnel exposures and effluent releases, and assessment and control of radiation doses, both to workers and the public. The program to maintain radiation exposures and releases as low as is reasonably achievable (ALARA) includes the control and disposal of radiological waste from reactor operations and from experimental programs.

Chapter 12 lists the bases and describes the functions of plans and procedures for the conduct of facility operations. These include discussions of the management structure, personnel training and evaluation, provisions for safety review and auditing of operations by the safety committees, and other required functions, such as reporting, security planning, emergency planning, and planning for reactor startup.

Chapter 13 lists the bases, scenarios, and analyses of accidents at the reactor facility, and describes an MHA, which may include a fission product release, and radiological consequences to the operational staff, reactor users, the public, and the environment. The function of ESFs is discussed in the accident analysis, as applicable.

Chapter 14 presents the technical specifications, which state the operating limits and conditions and other requirements for the facility to acceptably ensure protection of the health and safety of the public.

Chapter 15 concerns financial qualifications of the non-power reactor applicant for initial construction, continuing operations, and decommissioning.

Chapter 16 discusses other license considerations, such as prior reactor use and the use of the reactor for medical therapy. Issues not discussed elsewhere in the SAR are covered in this chapter.

Chapter 17 gives guidance on decommissioning. This includes the development of a decommissioning plan and the preparation of an amendment request to amend a license from operating to possession-only status.

Chapter 18, which discusses the conversion of the reactor from highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, includes topics covered in Chapters 1 to 17 as related to HEU to LEU conversions.

Appendix A lists regulations in selected parts of Title 10 that apply to research and test reactors.

General Requirements

Section 50.34 of Title 10 of the *Code of Federal Regulations* requires each applicant for a license to submit an SAR in the application. An SAR is clearly required for initial application for a license. Although no regulations apply specifically to an SAR submitted for renewing a non-power reactor license, the NRC staff can most effectively evaluate an application to renew a facility license from an up-to-date SAR.

The SAR performs the following important functions:

- Gives a complete description of the facility.
- Documents the design bases of the facility.
- Demonstrates and documents that the facility is designed and can be operated in a manner consistent with applicable regulations so that the health and safety of the public, the facility staff and users, and the environment are protected.
- Documents the limits, restrictions, administrative controls, and planned conduct of operations of the facility.
- Includes technical specifications based on the SAR. (The technical specifications express an agreement between NRC and the applicant on how the facility will be managed and operated to ensure the protection of the health and safety of facility personnel and the public, as well as protection of the environment.)

The SAR contains the formal documentation for a facility, presenting basic information about the design bases, and the considerations and reasoning used to support the applicant's conclusion that the facility can be operated safely. The descriptions and discussions therein also support the assumptions and methods of analysis of potential accidents, including the MHA, and the design of any ESFs used to mitigate accident consequences.

The SAR is the basic document that gives the NRC justification for licensing the facility. It gives information for understanding the design bases for the 10 CFR 50.59 change process, for training reactor operators, for preparing reactor operator licensing examinations, and for preparing for NRC inspections. For these reasons, and for others, it is important that the SAR remain an accurate, current description of the facility. Even though regulations do not require the licensee for a non-power reactor to periodically update the SAR (as is required in 10 CFR 50.71(e) for power reactors), the NRC staff encourages non-power reactor

licensees to maintain current SARs on file at NRC after initial licensing or license renewal by submitting replacement pages along with applications for license amendment and along with the annual report that summarizes changes made without prior NRC approval under 10 CFR 50.59.

An applicant for license renewal should address all applicable topics in this document by submitting an updated, complete SAR to account for any facility changes and any new regulatory requirements. Updating the SAR as described above will not completely eliminate the need to update and rewrite sections of the SAR at license renewal; it can, however, reduce the amount of resources needed to update the SAR. Licensees should review the license renewal requirements of 10 CFR 2.109 at least a year before the expiration date of a facility operating license and should contact NRC for any additional guidance that may be needed.

Purpose of the Format and Content Guide

This guide will help the applicant ensure the completeness and uniformity of the information submitted, assist the NRC staff and others in locating the information, and aid in reducing review time.

Applicability of the Format and Content Guide

The NRC staff recommends this guide for license applications for new non-power reactors and for license renewal applications for existing non-power reactors. This document also gives guidance to licensees preparing SARs for other licensing actions, such as license amendments. This guide should help licensees prepare complete packages and, thus, should reduce potential delays caused by NRC requests for additional information. Applications for license amendments should be written in accordance with applicable sections of the guide; however, a complete revision of the existing SAR should not be required in support of such an application. For license amendment requests, the corresponding sections of the SAR should be amended and submitted to NRC along with the amendment application. A complete revision of the SAR is strongly encouraged for license renewal. This format can be applied to all NRC-regulated non-power reactors. However, license applicants for non-power reactors with power levels above several tens of megawatts or with novel design features should contact the NRC staff to determine if additional guidance is needed.

The NRC staff recognizes that not every suggestion given here will be applicable to every non-power reactor. This problem is inherent to writing a single guidance document for reactors ranging from an AGN design with 0.1-watt thermal power to the heavy-water tank test reactor at the National Institute of Standards and Technology at 20 megawatts thermal power. As applicants consult this document, they will identify guidance that they believe is not applicable to their particular

reactor design. Applicants should carefully consider what guidance is applicable to their reactor design. The applicant does not need to discuss the reasoning for not providing all of the information suggested in this document. However, the applicant should be able to justify such deletion upon request of the NRC reviewer.

Use of the Standard Format and Content Guide

Although applicants are not required to consult this guide in preparing the application for a license or license amendment, the NRC staff strongly encourages its use because all applications will be reviewed and evaluated on the basis of their technical content and completeness. Upon receiving an application, the NRC staff will review and evaluate the SAR against the standard review plan (NUREG-1537, Part 2) to determine if the SAR contains the information necessary to form the bases for the staff findings required for the issuance or renewal of an operating license or granting of a license amendment.

Physical Specifications of the Application

Style and Composition

The applicant should

- Clearly and concisely state the technical bases to support the adequacy of designs or design methods.
- Include a contents page.
- Include the topics and headings at least to the level of headings with three digits (e.g., 2.4.2).
- Place an index of key items as back matter, if desired.
- Add appendices for supplemental information not explicitly discussed in this guide. Such information could comprise summaries of the responses to NRC regulatory guides or proposed regulations, and supplemental information on calculational methods or design approaches.
- Avoid duplicating information. Similar information may be requested in various parts because it is relevant to more than one portion of the facility or analysis. However, this information should be presented in the principal area of the document that deals with the topic and thereafter appropriately referenced in the other parts of the safety analyses. For example, where piping and instrumentation diagrams for the same system are needed in

more than one place in the document, the safety analyses may reference the first location, provided all necessary information is presented.

- The number of significant figures in numerical values should reflect the accuracy or precision to which the number is known. Where possible, estimate limits of error or uncertainty.

- Include equations for technical detail in safety analyses. Equations should use standard technical conventions and symbols. All symbols should be defined

- Specify measurements in the units used in the design of the facility. If the facility was designed in English units, the measurements should be given in English units first, followed by the SI [International System of Units (or metric)] numerical equivalent in parentheses. Drawings and diagrams in English units need not be changed to add SI units. If the facility was designed in SI units, only the SI units need be given.

- Use abbreviations in a consistent manner throughout the safety analyses and in a manner consistent with generally accepted use. Any abbreviations, symbols, or special terms unique to the facility or not in general use should be defined the first time they are used in the safety analyses.

- Submit three signed and notarized copies of the application in accordance with 10 CFR 50.30, 10 CFR 50.4, and Generic Letter 84-18. Part 170 of Title 10 of the *Code of Federal Regulations* presents information on licensing and amendment fees.

Graphic Presentations

The applicant should use graphic presentations, such as drawings, maps, diagrams, sketches, and tables to convey information more clearly or conveniently. All information should be clearly legible and reproducible and all symbols should be defined. Locate graphic presentations where the information they contain is primarily discussed.

References

The applicant should list documents referenced under a heading or topic at the end of the chapter or place them as footnotes on the page on which they are discussed. If the former, cite references in the text parenthetically by author and date or by reference numbers. If proprietary documents are referenced, cite a non-proprietary summary of the document if available (see 10 CFR 2 790).

Printing Specifications

The applicant should be guided by the following recommendations:

- Paper size should be 8-1/2 x 11 inches for text pages and for most drawings and graphics. If a larger page is needed, the finished copy should not exceed 8-1/2 x 11 inches when folded.

- Maintain a margin of no less than 1 inch on the top, bottom, and binding side of all pages submitted.

- Number pages with the digits corresponding to the chapter number followed by a hyphen and a sequential number (e.g., the third page of the discussion under Chapter 4, "Reactor," should be numbered 4-3).

- Use paper and ink suitable in composition, paper color, and ink density for microfilming or photocopying.

- Text pages may be single or double spaced using a suitable type face and style for microfilming or photocopying.

- The document may be mechanically or photographically reproduced. All pages of text may be printed on both sides of the paper. However, each major section (contents, chapter, appendix, etc.) should start on a right-hand page.

- Pages should be punched for standard three-hole loose-leaf binders.

Procedures for Updating or Revising Pages

The applicant should update or revise this guide by replacing pages. It is recommended that the applicant highlight the changed portion on each page by placing a change indicator mark consisting of a bold vertical line drawn in the margin opposite the binding margin (i.e., the outside margin). The line should be the same length as the portion changed. Replacement pages may be added in response to NRC staff requests for additional information.

All changed pages should show the date of change and a revision or change number. A guide with instructions for inserting, exchanging, or removing pages should accompany the changed pages. If affected, revised contents pages should be submitted.

Other Forms of Presentation

Other forms of presentation may be used. However, under 10 CFR 50.4(c), the applicant should contact the Information and Records Management Branch, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, to obtain specifications and copy requirements before submitting materials other than paper.

Revisions

The format and content guide should be revised and updated periodically as needed to clarify the content, to correct errors, and to incorporate modifications. The revision number and publication date should be printed at the bottom of each revised page. The revision numbers and dates need not be the same for all sections because individual sections will be replaced with a newly revised section only as needed. A list of affected pages will indicate the revision numbers for the current sections. As necessary, the staff will make corresponding changes to the standard review plan using these methods.

Contributors

This document was prepared by A. Adams, Jr., Senior Project Manager, Non-Power Reactors and Decommissioning Project Directorate, Division of Project Support, Office of Nuclear Reactor Regulation, U.S Nuclear Regulatory Commission Major contributors to the document include the project manager, S. Weiss, and M. Mendonca and T. Michaels, also of NRC; S. Bryan, W. Carpenter, R. Carter, D. Ebert, R. Garner, P. Napper, and P. Wheatley of the Idaho National Engineering Laboratory (INEL) under contract to NRC; and J. Hyder, J. Teel, and C. Thomas, Jr., of Los Alamos National Laboratory under contract to INEL. Comments and suggestions for improving this document should be sent to the Director, Non-Power Reactors and Decommissioning Project Directorate, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Notices of errors or omissions should be sent to the same address.

References

U.S. Nuclear Regulatory Commission, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," 1972.

U.S. Nuclear Regulatory Commission, Generic Letter 84-18, "Filing of Applications for Licenses and Amendments," July 6, 1984.

U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," 1987.

U.S. Nuclear Regulatory Commission, NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," Part 2, February 1996.

1 THE FACILITY

In this chapter of the safety analysis report (SAR), the applicant should present an introduction to the SAR and the facility. The introduction should state the purpose of the SAR and briefly describe the application. Chapter 1 should contain the following topics:

- introduction
- summary and conclusions of principal safety considerations
- general description of the facility
- shared facilities and equipment
- comparison with similar facilities
- summary of operations
- compliance with the *Nuclear Waste Policy Act* of 1982
- facility modifications and history

1.1 Introduction

The applicant should state its name and description (e.g., university, government agency, research institute, or company name) and should briefly state the purpose and intended use of the facility, the geographical location of the facility, the reactor type and power level, including principal inherent or passive safety features, and any unique design features. These topics should be covered in full and referenced to later chapters of the SAR.

1.2 Summary and Conclusions on Principal Safety Considerations

The applicant should state safety criteria, the principal safety considerations, and the resulting conclusions, including brief discussions of the following:

- consequences from the operation and use of the non-power reactor, and the methods used to ensure the safety of the reactor

- safety considerations that influenced the selection of the facility site, the type of reactor and fuel, the reactor thermal power level, the type of building housing the reactor, and any special factors

- any inherent or passive safety features designed to contribute to facility safety, protection of the health and safety of the public and staff, and protection of the environment

- design features and design bases for any systems and components that promote safe operation and shutdown of the facility

- potential accidents at the facility, including the maximum hypothetical accident, and any design features that prevent accidents or mitigate the potential consequences

These discussions need only be a general overview, with reference to the chapters in which detailed analyses appear.

1.3 General Description of the Facility

The applicant should briefly describe the reactor facility as follows: (1) geographical location; (2) principal characteristics of the site; (3) principal design criteria, operating characteristics, and safety systems; (4) any engineered safety features; (5) instrumentation, control, and electrical systems; (6) reactor coolant and other auxiliary systems; (7) radioactive waste management provisions (or system) and radiation protection; and (8) experimental facilities and capabilities. The general arrangement of major structures and equipment should be indicated with plan and elevation drawings. Safety features of the facility that are likely to be of special interest should be briefly identified. Such items as unusual site characteristics, the containment building, novel designs of the reactor, or unique experimental facilities should be highlighted. The information and discussions in this section in no way should substitute for the complete discussion and analysis found in (and referenced to) subsequent chapters of the SAR.

1.4 Shared Facilities and Equipment

The applicant should briefly describe the following:

- Systems and equipment that are shared with facilities not covered by the SAR or the operating license. Examples of shared facilities and equipment could be water purification systems; electrical supplies; heating, ventilation, and air conditioning systems; and the building that houses the reactor room.

- Any other reactor, subcritical assembly, irradiation facilities, or hot cell located within the confinement or containment structures, or the restricted area to which this SAR applies.

- Any safety barriers and any special isolation provisions for the shared facilities and equipment.

Complete descriptions and any safety implications that result from sharing facilities or systems should be evaluated in and referenced to the appropriate chapter of the SAR.

1.5 Comparison With Similar Facilities

The applicant should describe briefly the principal similarities to other facilities, particularly those either licensed by the U.S. Nuclear Regulatory Commission (NRC) or designed and operated by the U.S. Department of Energy (DOE). Comparisons should be made of the principal design parameters, reactor safety systems, engineered safety features, and instrumentation and control systems. The operating history of these facilities should be referenced briefly to demonstrate the safety and reliability of the design. Design features, operations experience, and tests and experiments from similar facilities could be referenced and used to support analyses in appropriate chapters of the SAR.

1.6 Summary of Operations

The applicant should briefly discuss reactor operations, experimental programs, and the mission of the reactor. The actual or proposed operations are important for estimating parameters such as total operating time, power level, pulsed or steady-state operation, and the amount and type of radioactive byproduct materials produced. If the facility licensee is applying for license renewal, this section should reflect current and proposed operational plans. If safety considerations analyzed in later chapters of the SAR limit the operating schedule of the reactor, that fact should be noted here.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

The applicant should briefly discuss how it meets the requirements of Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982 for disposal of high-level radioactive wastes and spent nuclear fuel. This discussion should include the contract arranged with DOE for return of the material. A copy of the cover letter for the contract between the applicant and DOE should be included in an appendix to the SAR.

1.8 Facility Modifications and History

This section of the SAR applies primarily to existing facilities that are applying for license renewal. This section has limited applicability to an application for initial construction permit and operating license. The applicant should present a brief history of the facility, including the dates of significant events, issuance of the construction permit and operating license, and initial criticality. If the licensee has experience with other research reactors, a brief description of this information should be presented. The SAR should indicate if the facility has not undergone significant or safety-related physical or operational modifications since it was

initially licensed, or since the last renewal was issued. The SAR should reflect any significant modifications made to the non-power reactor, programs, or schedules. The modifications should be discussed briefly in this section in chronological order, including the number and date of the license amendment. Changes performed under the provisions of Section 50.59 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.59) that affect the SAR descriptions should be provided. If applicable, technical specifications changes should also be given.

2 SITE CHARACTERISTICS

In this chapter of the SAR, the applicant should discuss and describe the geographical, geological, seismological, hydrological, and meteorological characteristics of the site and vicinity in conjunction with present and projected population distributions, industrial facilities and land use, and site activities and controls. In this presentation, the applicant should define the site characteristics for use in design and analyses discussions in other chapters of the SAR, e.g., Chapter 3, "Design of Structures, Systems and Components"; Chapter 11, "Radiation Protection Program and Waste Management"; and Chapter 13, "Accident Analyses." In 10 CFR 100.10, the staff gives factors to consider in selecting sites for test reactors and related reactor design.

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The reactor should be located by latitude and longitude to the nearest second and by Universal Transverse Mercator Coordinates [Zone Number, Northing, and Easting, as found on U.S. Geological Survey (USGS) topographical maps] to the nearest 100 meters. The State and county or other political subdivision in which the site is located should be identified, including the location on a campus, if applicable, as well as the location of the site with respect to prominent natural and manmade features such as highways, rivers, lakes, reservoirs, and mountains.

2.1.1.2 Boundary and Zone Area Maps

Maps of the site area of suitable scale (with explanatory text as necessary) should be included to show the boundaries and zones associated with the facility. Boundaries and zones are defined in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.7 (1977), "Research Reactor Site Evaluation," Section 2, and in the documents in the bibliography related to emergency preparedness at non-power reactors (NRC Regulatory Guide 2.6, NRC NUREG-0849, ANSI/ANS 15.16). The maps should clearly show the following:

- the general area in which the reactor will be located and sufficient secondary detailed maps to show the location of the reactor facility and adjacent surroundings

- location of the area directly under the NRC facility operating license

- location of the operations boundary
- location of the site boundary
- location of the rural zone up to a distance of 8 kilometers from the reactor
- location of the urban zone up to a distance of 8 kilometers from the reactor
- location of emergency preparedness zones (EPZs), as applicable
- true north
- highways, railways, and waterways that traverse or are in close proximity to the site
- the general topography of the area near the reactor that could affect diffusion and dispersion of airborne effluents, including buildings at least as tall as the reactor building and any stacks or other air-exhaust facilities

2.1.2 Population Distribution

Population data presented should be based on the most recently available (last decade or later) census data. Information on population distributions should be in suitable form to use in dose analyses in Chapters 11 and 13, in which potential doses down to a small percentage of 10 CFR Parts 20 or 100 may be applicable.

On a map of suitable scale that identifies places of significant population grouping (such as cities and towns) within an 8-kilometer radius, concentric circles should be drawn, with the reactor at the center point, at distances of 1, 2, 4, 6, and 8 kilometers. The population in each area at the time of application and a projection of the population in five years and at the end of the license period should be given. The basis for population projections should be described. Information should be given about the direction and distance of the nearest permanent residence to the reactor and any reactor effluent exhaust points. Any part-time, transient, or seasonal occupation of buildings should be described, such as classrooms or dormitories on a university campus, giving best estimates of occupation times and numbers of occupants.

2.2 Nearby Industrial, Transportation, and Military Facilities

In this section, the applicant should establish whether the effects of potential accidents in the vicinity of the reactor from present and projected industrial,

transportation, and military installations and operations should be used in the safety analyses and should establish the reactor facility design parameters related to accidents selected. The applicant should consider all facilities and activities within 8 kilometers of the reactor. Facilities and activities at greater distances should be included as appropriate to their significance of accident impact on the facility.

2.2.1 Locations and Routes

The applicant should submit maps showing the location and distance from the reactor of all significant manufacturing plants; chemical plants; refineries; storage facilities; mining and quarrying operations; military bases; missile sites; transportation routes (air, land, and water); transportation facilities (docks, subways, highways, railways, and rail yards, anchorages, airports); oil and gas pipelines, drilling operations, and wells; and underground facilities used for such purposes as fuel storage and storm-water runoff. These maps should show any other facilities that, because of the products manufactured, stored, or transported there, may require consideration with respect to possible adverse effects on the reactor. Any military firing or bombing ranges and any nearby aircraft flight, holding, and landing patterns should be indicated on the maps.

The maps should be clearly legible and of suitable scale to enable easy location of the facilities and routes in relation to the reactor. All symbols and notations used to depict the location of the facilities and routes should be identified in legends or tables. Topographic features should appear on the maps in sufficient detail to illustrate the information presented and to support analyses of potential impacts on the reactor facility.

2.2.2 Air Traffic

Factors such as frequency and type of aircraft movement, flight patterns, local meteorology, and topography should be considered for the following sites:

- sites located within 8 kilometers of an existing or projected commercial or military airport

- sites located between 8 and 16 kilometers from an existing or projected commercial or military airport with more than approximately 200 d^2 (where d is the distance in kilometers from the airport to the reactor site) commercial or military aircraft movements per year.

Special consideration should be given when siting the facility within the trajectory of a runway of any airport. The analysis should demonstrate that there is a low potential that any aircraft, including general aviation aircraft, could affect the

reactor or that the consequences from any aircraft-associated accident are already bounded or considered in the accident analysis.

2.2.3 Analysis of Potential Accidents at Facilities

For each facility identified in Section 2.2.1, the applicant should provide an analysis of possible effects on the reactor for postulated accidents or other events that could occur at the facility. If a facility cannot affect the reactor, the applicant should make a statement to that effect and give a basis for this statement.

2.3 Meteorology

In this section, the applicant should describe the meteorology of the site and its surrounding areas. Sufficient data on average and extreme conditions should be included to permit an independent evaluation by the reviewer.

2.3.1 General and Local Climate

The general climate of the region should be described with respect to types of air masses, synoptic features (high- and low-pressure systems and frontal systems), general and prevailing air-flow patterns (wind direction and speed), temperature and humidity, precipitation (rain, snow, and sleet), and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions. References should indicate the climatic atlases and regional climatic summaries used.

Historical seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes, waterspouts, thunderstorms, lightning, and hail, should be stated. The applicant should give the known and maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. The applicant should estimate the 100-year return wind speed. The applicant should also estimate the weight of the 100-year return period snowpack and the weight of the 48-hour probable maximum precipitation for the site vicinity, if applicable, as specified by the USGS. Using these estimates for Chapter 3, the applicant should calculate the design loads on the roof of the reactor building, and compare them with local building codes for similar types of structures.

2.3.2 Site Meteorology

In addition to discussing potential meteorological effects on the reactor facility, the applicant should give sufficient information to support the dispersion analyses of airborne releases from the facility. The applicant may need to evaluate potential radiological effects in both the restricted and unrestricted areas in the reactor

vicinity from routine releases during normal operations and from postulated releases resulting from accidents. The analyses of potential doses from normal and accident releases should be placed in Chapters 11 and 13, respectively. The meteorological information used for both long-term and short-term dispersion calculations, along with a description of the technical bases of the dispersion model should be summarized. The continuing onsite measurements program or an alternative source of meteorological information (e.g., National Weather Service station) should be described; and plans for access to meteorological information during the license period should be described. Description of the meteorological program should include measurements made, locations and elevations of measurements, description of instruments and their performance specifications, and calibrations, type of data output, and data analysis procedures.

2.4 Hydrology

In this section, the applicant should give sufficient information to allow an independent hydrologic engineering review to be made of all hydrologically related design bases, performance requirements, and bases for operation of structures, systems, and components important to safety.

Sufficient information should also be given about the water table, groundwater, and surface water features at the reactor site to support analyses and evaluations in Chapters 11 and 13 of consequences of uncontrolled release of radioactive material from pool leakage or failure, neutron activation of soils in the vicinity of the reactor, or deposition and migration of airborne radioactive material released to the unrestricted area.

The effect of potential floods on sites along streams, rivers, and lakes should be analyzed. Effects and consequences of a probable maximum flood, seiche, surge, standing water, drainage or seismically induced flood (such as might be caused by dam failure) should be considered. Hazards of tsunami, river blockage, diversion in the river system, or distant or locally generated "sea waves" should be described to establish the suitability of a site. The detail and extent of the considerations should be commensurate with the potential consequences to the reactor and to the public, the environment, and the facility staff.

2.5 Geology, Seismology, and Geotechnical Engineering

In this section, the applicant should detail the seismic and geologic characteristics of the site and the region surrounding the site. The degree of detail and extent of the considerations should be commensurate with the potential consequences of seismological disturbance, both to the reactor facility and to the public from radioactive releases.

2.5.1 Regional Geology

The applicant should discuss all geologic and seismic hazards within the region that could affect the facility, and relate them to the regional physiography, tectonic structures and tectonic provinces, geomorphology, stratigraphy, lithology, and geologic and structural history and geochronology.

2.5.2 Site Geology

The applicant should discuss in detail the structural geology at the facility site, including the relationship of site structure to regional tectonics, and should pay particular attention to specific structural units of significance to the site such as folds, faults, synclines, anticlines, domes, and basins. The applicant should also discuss the geologic history of the site and should relate it to the geologic history of the region.

2.5.3 Seismicity

The applicant should list all historically reported earthquakes that could have reasonably affected the region surrounding the site. The list should include all earthquakes of modified Mercalli intensity greater than IV or magnitude (Richter) greater than 3.0 that have been reported in all tectonic provinces, any part of which is within 200 kilometers of the site.

2.5.4 Maximum Earthquake Potential

The applicant should note the largest historic earthquake associated with each geologic structure or tectonic province. If the earthquakes are associated with a geologic structure, the applicant should evaluate the largest earthquake that could occur on that structure on the basis of such considerations as the nature of faulting, fault length, fault displacement, and earthquake history. If the earthquakes are associated with a tectonic province, the applicant should identify the largest historical earthquakes within the province and, whenever reasonable, should estimate the return period for the earthquakes. Also, isoseismal maps for the earthquakes should be presented.

2.5.5 Vibratory Ground Motion

The applicant should proceed from discussions of the regional seismicity, geologic structures, and tectonic activity to a determination of the relation between seismicity and geologic structures. The earthquake-generating potential of tectonic provinces and any active structures should be identified. Finally, the applicant should assess the ground motion at the site from the maximum potential

earthquakes associated with each tectonic province or geologic structure and should consider any site-amplification effects. Using the results, the applicant should establish the vibratory ground motion design spectrum.

2.5.6 Surface Faulting

The applicant should discuss any potential for surface faulting at the site, and should list all historically reported earthquakes that can be reasonably associated with faults, any part of which is within 8 kilometers of the site.

2.5.7 Liquefaction Potential

The applicant should discuss soil structure. If the foundation materials at the site adjacent to and under safety-related structures are saturated soils or soils that have a potential for becoming saturated, the applicant should prepare an appropriate state-of-the-art analysis of the potential for liquefaction at the site. The applicant should also determine the method of analysis on the basis of actual site conditions, the properties of the reactor facilities, and the earthquake and seismic design requirement for the protection of the public.

2.6 Bibliography

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," 1977.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, "Emergency Planning for Research Reactors," 1982.

International Atomic Energy Agency, IAEA-TECDOC-348, "Earthquake Resistant Design of Nuclear Facilities With Limited Radioactive Inventory," 1985.

International Atomic Energy Agency, IAEA-TECDOC-403, "Siting of Research Reactors," 1987.

U.S. Nuclear Regulatory Commission, NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," 1983.

U.S. Nuclear Regulatory Commission, NUREG/CR-2260, "Technical Basis for R.G. 1.145 Atmospheric Dispersion Models," 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, Rev. 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," 1982.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," 1983.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

In this chapter of the SAR, the applicant should identify and describe the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public. The material presented should emphasize the safety and protective functions and related design features that help provide defense in depth against uncontrolled release of radioactive material. The bases for the design criteria for some of the systems discussed in this chapter may be developed in other chapters and should be appropriately cross referenced. For example, confinement or containment design criteria may be summarized here and discussed in detail in Chapter 6, "Engineered Safety Features."

3.1 Design Criteria

In this section the applicant should identify the structures, systems, and components; modes of operation; location; type(s) of actuation; relative importance in the control of radioactive material and radiation; applicable design criteria; and chapter and section in the SAR where these criteria are applied in the design of specific structures, systems, and components.

The design criteria should include applicable standards, guides, and codes, for example, American National Standards Institute/American Nuclear Society (ANSI/ANS) standards (see references); NRC regulatory guides (see Division 2 regulatory guides and "Other Regulatory Guides of Possible Interest to Division 2 Recipients," which is a list attached to the Division 2 regulatory guides table of contents); and national, State, and local building, plumbing, and electrical codes.

In this section the applicant should specify the design criteria for the facility structures, systems, and components. The description of the actual design should be in the section or chapter that corresponds to the specific structure, system, or component. The design criteria should be both specific and general. The amount of detail given should be related to the safety function of the structure, system, or component. For example, general design criteria should include the following:

- Design for the complete range of normal expected reactor operating conditions (e.g., reactor power levels from cold subcritical conditions to maximum allowed power level, associated radiation and system temperature conditions, and allowed fuel storage and reactor operating configurations).

- Design to cope with anticipated transients and potential accidents, including those discussed in Chapter 13, "Accident Analyses," of the SAR.

Anticipated transients and potential accidents should include malfunction of any control function or other malfunction of a structure, system, or component, experiment malfunction, single operator error, testing and surveillance activity, reactor startup and shutdown, and power-level change. These design criteria should be based on a systematic examination of the most limiting transients and accidents to identify the needed facility structures, systems, and components. They should ensure that each needed structure, system, or component stays within acceptable operational and safety limits for conservative assumptions of initial conditions, operating history of the facility for the proposed license term, and required operating characteristics. The most limiting conditions of each type should be analyzed in detail in Chapter 13 of the SAR.

- Design redundancy for reactor protective and safety features, so that any single failure of any active component will not prevent safe reactor shutdown or result in unsafe conditions as verified by Chapter 13 analyses.

- Design to facilitate inspection, testing, and maintenance of the structures, systems, and components whose integrity and reliability are important to safe reactor shutdown and to the protection of the public, reactor facility personnel, and environment.

- Provisions to avoid or mitigate the consequences of fires, explosions, and other potential manmade or natural conditions.

- Quality standards commensurate with the safety function and the potential risks. For example, fuel fabrication may be consistent with the applicable guidance in ANSI/ANS 15.2–1990.

- Analyses and designs for meteorological, hydrological, and seismic effects.

- The bases for technical specifications necessary to ensure the availability and operability of required structures, systems, and components.

3.2 Meteorological Damage

In this section the applicant should describe the design for the protection from meteorological conditions of facility structures (e.g., buildings and cooling towers), systems (e.g., ventilation systems), and components that are assumed to be operable in the SAR. The design criteria should be based on data given in Chapter 2, "Site Characteristics," on such factors as historical data on maximum wind velocity, vertical velocity profiles, gust factors, applied loads, recurrence intervals, tornado loadings, and snow and ice loads. The applicant may refer to local building codes, standards, or other criteria to ensure that significant

meteorological damage to the facility is very unlikely. Further, the design criteria should provide reasonable assurance that potential meteorological damage would not significantly affect designed structures, systems, and components (i.e., they would continue to perform necessary operational and safety functions). An example would be consideration of adverse wind conditions that affect ventilation systems. The bases for appropriate technical specification surveillances to verify capability and reliability of the design features should be given.

3.3 Water Damage

In this section the applicant should specifically describe the proposed site and facility designs to protect against water damage of the structures, systems, and components assumed to function in the SAR. This should include (1) the impact on structures resulting from the force or submergence of flooding, (2) the impact on systems resulting from instrumentation and control electrical or mechanical malfunction due to water, and (3) the impact on equipment, such as fans, motors, and valves, resulting from degradation of the electromechanical function due to water. This section should be based on historical data on the site with regard to potential flooding and other hydrological conditions discussed in detail in Chapter 2. Design criteria for structures and systems that are based on information on precipitation rates, ground water, accumulation of standing water, and drainage rates should be included. The impacts of watersheds, flood plains, drainage easements, and fluid supplies or conduits on reactor operation, safe shutdown, and control of radioactive material should also be included. Maps and other information in Chapter 2 to describe these features and characteristics around the facility should be used.

The applicant should use local building codes or other applicable standards to ensure that significant water damage to the facility is very unlikely. Facility design features (e.g., elevations, sumps, pumps, watertight doors, berms, and drains) may be used to avoid or mitigate water damage to structures, systems, and components important to safety. The applicant should show that the design features are sufficient to avoid significant water damage to the facility during the projected reactor license term.

The bases for any technical specifications required to ensure operability of structures, systems, and components that ensure safe reactor shutdown should be given.

3.4 Seismic Damage

In this section the applicant should specify and describe the structures, systems, and components that are required to maintain the necessary safety function if a seismic event should occur, as well as the required facility seismic design criteria.

The seismic characteristics of the site should be summarized in Chapter 2. Seismic design for non-power reactors should, at a minimum, be consistent with local building codes and other applicable standards.

The reactor facility seismic design should provide reasonable assurance that the reactor could be shut down and maintained in a safe condition or that the consequences of accidents would be within the acceptable limits. For most NRC-licensed non-power reactors, this may involve analysis to show that the conditions of the SAR for safe reactor shutdown remain valid for the potential seismic events (e.g., reactor fuel fission product barrier and shutdown capability would not be impaired). The applicant can also show that the radiological consequences of a potential seismic event are bounded by the accident analyses in Chapter 13.

Such criteria for acceptable seismic performance have been established in ANSI/ANS 15.7. With regard to seismic design, Section 3.2(2) of ANSI/ANS 15.7 states, "(R)eactor safety related structures and systems shall be seismically designed such that any seismic event cannot cause an accident which will lead to dose commitments in excess of those specified in 3.1." For NRC-licensed non-power reactors, "any seismic event" should be the maximum historical intensity earthquake in accordance with the guidance on the design-basis earthquake in Section 3.1.2.1 of International Atomic Energy Agency document IAEA-TECDOC-403. This IAEA document gives additional guidance and references IAEA-TECDOC-348, which contains guidance on the seismic design of structures, systems, and components.

With regard to the allowed dose commitments for seismic events and designs in Section 3.1 of ANSI/ANS 15.7, the terms "site boundary," "rural zone," and "urban boundary" are used. For most NRC-licensed non-power reactors, "rural zone" should not be used and "urban boundary" should be assumed to begin at the "site boundary." However, given applicable site characteristics and emergency preparedness requirements, the criteria as specified in Section 3.1 of ANSI/ANS 15.7 could be used.

The above guidance is applicable to research reactors licensed by NRC. For test reactors the requirements of 10 CFR Part 100 must be applied. The guidance and criteria of 10 CFR Part 100 are complete and are adequate for assessing test reactors.

To verify that seismic design functions are met, the applicant should give the bases for technical specifications necessary to ensure operability, testing, and inspection of associated systems, including instrumentation and control portions, if applicable.

3.5 Systems and Components

In this section the applicant should give the design bases for the systems and components required to function for safe reactor operation and shutdown. For non-power reactors, this section should include, at a minimum, the fuel system, control rod scram systems, other protective and safety systems, and the electromechanical systems and components associated with emergency core cooling systems, reactor room ventilation, confinement or containment systems, and other systems that may be required to prevent uncontrolled release of radioactive material. The design criteria should include the conditions that are important for reliable operation of the systems and components (e.g., dynamic and static loads, number of cyclic loads, vibration, wear, friction, strength of materials, and effects of radiation and temperature). The specific application of these design criteria should generally be given in other chapters of the SAR. For example, if this chapter establishes that a design criterion for the control rods is that it drop by the force of gravity, Chapter 4, "Reactor Description," should describe the electromechanical and reactor dynamic design bases to accomplish this insertion within a specified time, normally 1 second.

3.6 References

American National Standards Institute, ANSI N323, "Radiation Protection Instrumentation Test and Calibration," ANS, LaGrange Park, Illinois, 1978.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.2, "Quality Control for Plate-Type Uranium-Aluminum Fuel Elements," ANS, LaGrange Park, Illinois, 1990.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," ANS, LaGrange Park, Illinois, 1977.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1986.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Controls at Research Reactors," ANS, LaGrange Park, Illinois, 1993.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.15, "Criteria for Reactor Safety Systems for Research Reactors," ANS, LaGrange Park, Illinois, 1978.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.17, "Fire Protection Program Criteria for Research Reactors," ANS, LaGrange Park, Illinois, 1981.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.20, "Criteria for the Reactor Control and Safety Systems of Research Reactors," (draft), ANS, LaGrange Park, Illinois.

International Atomic Energy Agency, IAEA-TECDOC-348, "Earthquake Resistant Design of Nuclear Facilities With Limited Radioactive Inventory," Vienna, Austria, 1985

International Atomic Energy Agency, IAEA-TECDOC-403, "Siting for Research Reactors," Vienna, Austria, 1987.

4 REACTOR DESCRIPTION

In this chapter of the SAR, the applicant should discuss and describe the principal features, operating characteristics, and parameters of the reactor. The analysis in this chapter should support the conclusion that the reactor is conservatively designed for safe operation and shutdown under all credible operating conditions. Information in this chapter of the SAR should provide the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many technical specifications.

4.1 Summary Description

In this section the applicant should briefly summarize the design and functional characteristics of the reactor. The applicant should present the principal safety considerations in the selection of the reactor type as well as the design principles for the components and systems that address those considerations. This section should contain summary tables of important reactor parameters and sufficient drawings and schematic diagrams to explain and illustrate the main reactor design features.

The applicant should briefly address the following features of the reactor:

- thermal power level
- fuel type and enrichment
- pool or tank type
- forced and/or natural-convection cooling
- type of coolant, moderator, and reflector
- principal features for experimental programs
- pulsing or steady power
- novel concepts requiring substantial new development

4.2 Reactor Core

In this section the applicant should present all design information and analyses necessary to demonstrate that the core can be safely operated. The major core components to be described are fuel, neutron moderator, neutron reflector, control elements, neutron startup source, incore cooling components, and any incore experimental facilities. The source or basis of the information presented should be given.

4.2.1 Reactor Fuel

In this section the applicant should describe the reactor fuel system. Included should be the design features selected to ensure that the fuel and cladding can withstand all credible environmental and irradiation conditions during their life cycle at the reactor site. The discussions should address the incore fuel operating conditions. Handling, transport, and storage of fuel should be discussed in Chapter 9, "Auxiliary Systems," of the SAR. Drawings and tables of design specifications and operating characteristics of the fuel should be presented.

Most non-power reactors contain heterogenous fuel elements consisting of rods, plates, or pins, which are addressed in the following sections. Homogeneous fuels should be described and analyzed in a comparable way. Information should be current; supported by referenced tests, measurements, and operating experience; and compared with additional applicant experience where applicable. The information should include the following:

- Chemical composition, enrichment, uranium loading, and important metallurgical features of the fissile material in the basic fuel unit. The information should indicate dispersion, alloy, cermet, sintering, and such special properties as burnable poisons or neutron moderators.

- Description of the basic fuel unit, including plates, rods, pins, or pellets. This information should include dimensions, fabrication methods, and cladding or encapsulation methods. Special features, such as moderators or reflectors, external geometrical designs to enhance cooling capability, and inherent safety or feedback provisions should be discussed.

- Material and structural information such as dimensions, spacings, fabrication methods, compatibility of materials, and specifications with tolerances. All types of fuel elements to be used should be described, including full elements, partial elements, control rod elements, instrumented elements, and special elements for experimental facilities. Features that ensure accurate and secure positioning and adequate cool nt flow should be described.

- Information on material parameters that could affect fuel integrity, such as melting, softening, or blistering temperatures; corrosion; erosion; and mechanical factors, such as swelling, bending, twisting, compression, and shearing.

- Physical properties with significance in regard to safety and fuel integrity that are important for the thermal-hydraulic analyses, such as heat capacity, thermal conductivity, gas evolution or diffusion, occluded or encapsulated

void volume, fuel burnup limits, capability to retain fission products, swelling resistance, and buildup of oxides.

- If the reactor is designed for pulsing, any special attributes of the fuel that contribute to pulsing safety.

- A brief history of the fuel type, with references to the fuel development program, including summaries of performance tests, qualification, and operating history. A brief history of the actual fuel elements to be used, including fabrication, previous irradiation conditions, storage environments, surveillance procedures, and qualification tests.

- Mechanical forces and stresses, hydraulic forces, thermal changes and temperature gradients, internal pressures including that from fission products and gas evolution, and radiation effects including the maximum fission densities and fission rates that the fuel units and elements are designed to accommodate.

Limits on operating conditions for the fuel should be supported by information and analyses. These limits are specified to ensure that the integrity of the fuel elements and their cladding or fission product barrier will not be impaired. They should form the design bases for this and other chapters of the SAR, for the reactor safety limits, and for other fuel-related technical specifications.

4.2.2 Control Rods

In this section the applicant should give information on the control rods, including all rods or control elements that are designed to change reactivity during reactor operation. The physical, kinetic, and electromechanical features demonstrating that the rods can fulfill their control and safety functions should be described. Results of computing control rod reactivity worths may be presented in this section, but details of the calculation of reactivity effects should appear in Section 4.5, "Nuclear Design," of the SAR. The information in this section should include the following:

- The number and types of rods (e.g., shim, safety, regulating, transient), their designed locations in the core, and their designed reactivity worths. The considerations and bases for redundancy and diversity should be provided. Limits on core configuration should be discussed.

- The structural and geometric description, including the shape, size, materials, cladding, fabrication methods, and specifications with tolerances for the rods. This should include the type and concentration of neutron absorber, or emitter, if applicable. Also, calculations of changes in

reactivity worth due to burnup and assessment of radiation damage, heating effects, and chemical compatibility with the coolant and other core components should be given. If the control rods have followers, the design, composition, and reactivity effects of the follower should be discussed.

- The design of mechanical supports for the active component, the method of indicating and ensuring reproducible positioning in the core, and the drive mechanism of each type of rod. This information should include the source of motive power, usually electrical, and the systems ensuring scram capability. For a reactor designed for pulsing, the transient rod should be described in detail, including its drive mechanisms and the methods for calibration, pulse reproducibility, and prevention of inadvertent pulsing.

- The kinetic behavior of the rods, showing either the positive or negative rate of reactivity change, in the normal drive and scram modes of operation. This information should be supplied for all rods, including transient rods in a reactor designed for pulsing. The applicant should show that the control rod design conforms with the shutdown margin requirements.

- The scram logic and circuitry, interlocks and inhibits on rod withdrawal, trip release and insertion times, and trip or scram initiation systems should be summarized here and described in detail in Chapter 7, "Instrumentation and Control Systems."

- Special features of the control rods, their core locations, power sources, drive or release mechanisms designed to ensure operability and capability to provide safe reactor operation and shutdown under all conditions during which operation is required in the safety analysis if there is a single failure or malfunction in the control system itself. Such features may include mechanisms to limit the speed of rod movement.

- Technical specification requirements for the control rods and their justification. These are the limiting conditions for operation, surveillance requirements, and design features as discussed in Chapter 14, "Technical Specifications," of this format and content guide.

4.2.3 Neutron Moderator and Reflector

In this section the applicant should discuss the materials and systems designed to moderate the neutrons within the fuel region and reflect leakage neutrons back into the fuel region. The information should include the materials, geometries, designs for changes or replacement, provisions for cooling, radiation damage

considerations, and provisions for experimental facilities or special uses. Multiple-use systems and features such as moderator coolant, fuel moderator, and reflector shield should be described. If moderators or reflectors are encapsulated to prevent contact with coolant, the effect of failure of the encapsulation should be analyzed. It should be possible to operate the reactor safely until failed encapsulations are repaired or replaced. If reactor operations cannot be safely continued, the reactor should be placed and maintained in a safe condition until encapsulations are repaired or replaced. Technical specification requirements should be proposed and justified for the moderator and reflector in accordance with the guidance in Chapter 14 of this format and content guide. The nuclear design of the moderator and reflector should be discussed in Section 4.5 of the SAR.

4.2.4 Neutron Startup Source

In this section the applicant should present design information about the neutron startup source and its holder. The applicant should show that the source will produce the necessary neutrons to allow a monitored startup with the reactor instrumentation. The information should include the neutron strength and spectrum, source type and materials, its burnup and decay lifetime, and its regeneration characteristics. Other necessary information includes the material and geometry of the holder, the method of positioning the source in the core, and the core locations in which the source is designed to be used. Utilization information and such limitations as radiation heating or damage and chemical compatibility with coolant and other core components should be discussed. Any technical specification limits on the source, such as the maximum power level the reactor can be run with the source in place (for plutonium-beryllium sources and other source types that can act as fuel), or surveillance requirements to ensure source integrity should be proposed and justified in this section of the SAR in accordance with the guidance in Chapter 14 of this format and content guide.

4.2.5 Core Support Structure

In this section the applicant should present design information about the mechanical structures that support and position the core and its components. The information should include the following:

- The design considerations that ensure that all necessary loads and hydraulic forces can be conservatively supported with and without the buoyant forces of the reactor water.

- The methods by which core components are accurately and reproducibly positioned and secured, including specification tolerances, as well as features of the grid plate such as fuel holddown grids, fuel element spacers, and control rod guides and supports.

- The materials of construction, including considerations for radiation damage, corrosion, erosion, chemical compatibility with coolant and core components, potential effects on reactivity, induced radioactivities, and maintenance.

- Design features of the core support structure that accommodate other systems and components such as radiation shields, beam ports or other experimental facilities, coolant pipes, coolant plenums or deflectors, and nuclear detectors.

- For a movable core support, design information describing the motive power system, the system for ensuring position, and interlocks that prevent or control motion while the reactor is critical, while forced cooling is required, or while other activities that prohibit core support movement are to be conducted, if such a system is required (e.g., experimental facility operations).

- Technical specifications that control important design features, limiting conditions for operation, and surveillances as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.3 Reactor Tank or Pool

The cores of most non-power reactors are immersed in water within a tank or pool. In this section the applicant should present all information about the tank or pool necessary to ensure its integrity. The information should include the following:

- Design and considerations to ensure that no hydrodynamic, hydrostatic, mechanical, chemical, and radiation forces or stresses could cause failure or loss of integrity of the tank during its projected lifetime over the range of design characteristics.

- Design and dimensions to ensure sufficient shielding water to protect personnel and components, as well as sufficient depth to ensure necessary coolant flow and pressures. (Also see Sections 4.4 and 4.6 and Chapter 11, "Radiation Protection Program and Waste Management," of this format and content guide.)

- Designs and description of materials, including dimensions, supporting structures, chemical compatibility with the coolant and other reactor system components, radiation fields and any consequences of radiation

damage, protection from corrosion in inaccessible regions, and capability to replace components, if necessary.

- Locations of penetrations and attachment methods for other components and pipes. The relationships of these penetrations to core and water surface elevations should be discussed. Safety-related features that prevent loss of coolant should be discussed and related to Sections 4.4 and 4.6 and to the loss-of-coolant-accident scenarios analyzed in Chapter 13, "Accident Analyses," as applicable.

- Planned methods for assessing radiation damage, chemical damage, or deterioration during the projected lifetime. In this section the applicant should assess the possibility of uncontrolled leakage of contaminated primary coolant and should discuss preventive and protective features.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.4 Biological Shield

In this section the applicant should present information about the principal biological shielding designed for the reactor. The information should include the following:

- The design bases for the radiation shields (e.g., water, concrete, or lead), including the projected reactor power levels and related source terms and the criteria for determining the required protection factors for all applicable nuclear radiation activity. Information about conformance with the regulations for radiation exposure and the facility ALARA (as low as is reasonably achievable) program should appear in Chapter 11. The design basis should include the designed reactor power levels, the associated radiation source terms, and other radiation sources within the pool or tank that require shielding.

- The design details and the methods used to achieve the design bases. The applicant should discuss the protection of personnel and equipment functions. The information should specify the general size and shape of the shields and the methods used to ensure structural strength, rigidity, and functional integrity. The applicant should discuss the distribution of shielding factors between liquid (water) and solid (concrete, lead, etc.) materials. If loss of shield integrity could cause a loss-of-coolant accident, the features to prevent the loss of integrity should be described.

- The materials used and their shielding coefficients and factors, including a detailed list of constituents and their nuclear and shielding properties. The applicant should discuss radiation damage and heating or material dissociation during the projected lifetime of the reactor; induced radioactivity in structural components; potential radiation leakage or streaming at penetrations, interfaces, and other voids; shielding at experimental facilities; and shielding for facilities that store fuel and other radioactive materials within the reactor pool or tank.

- The assumptions and methods used to calculate the shielding factors, including references to and justification of the methods. Detailed results of the shielding calculations should give both neutron and gamma-ray dose rates at all locations that could be occupied. The applicant should calculate shield penetrations and voids, such as beam ports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.

- Methods used to prevent neutron irradiation and activation of ground water or soils surrounding the reactor shield that could enter the unrestricted environment. The applicant should estimate the maximum activity should such activation occur and describe remedial actions.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

Regulatory Guide 2.1, "Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors" is given as Appendix 4.1.

4.5 Nuclear Design

In this section the applicant should give information on the nuclear parameters and characteristics of the reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle of allowed cores and burnup as discussed in the safety analysis. The descriptions, analyses, and results should address all safety issues in the design and operation of the reactor and should support the conclusion that the reactor can be built and operated without unacceptable risk to the health and safety of the public. A detailed description of the analytical methods used in the nuclear design should be given. Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Code descriptions should

include methods of obtaining parameters such as cross sections. Estimates of the accuracy of the analytical methods should be included. Tables and figures should be used as necessary to present information clearly.

4.5.1 Normal Operating Conditions

In this section the applicant should present information on the core geometry and configurations. Operating core configurations should be compact, with no vacancy in which fuel could be inserted within the core periphery. The limiting core configuration for a reactor is the core that would yield the highest power density using the fuel specified for the reactor. All other core configurations should be demonstrated to be encompassed by the safety analysis of the limiting core configuration. Further information on power density limitations should be given in Sections 4.5.3 and 4.6. The information in the SAR should include the following:

- The number, types, and locations of all core components on the grid plate, including fuel, control rods, neutron reflectors, moderators, incore experimental components, and core-associated cooling components. If this information appears elsewhere in the SAR, the section where it is located should be referenced.

- Descriptions of planned core configurations during the life of the reactor, showing how a compact core is ensured.

- Discussions and analyses of the reactor operating characteristics. The applicant should give in detail the effects of changes in configuration and fuel burnup. If applicable, the applicant should analyze safety-related considerations for all requested operating modes (e.g., steady power and pulsing).

- Changes in core reactivity with fuel burnup, plutonium buildup, and poisons, both fission products and those added by design, if applicable.

- Analyses of the reactor kinetic behavior and the design requirements and dynamic features of the control rods that allow controlled operation for all possible reactor conditions.

- Analyses of the basic reactor criticality physics, including the interacting effects of fuel, neutron moderators and reflectors, control rods, and incore or in-reflector components such as experimental facilities.

- Discussion of the safety considerations for different core configurations, including a limiting core configuration that would yield the highest power densities and fuel temperatures achievable with the planned fuel.

- The individual reactivity worths of fuel elements, reflector components, incore and in-reflector components, experimental components, and control rods in allowed positions. If experimental facilities or components could be voided or flooded, the reactivity effects and safety considerations should be included.

- The calculated core reactivities for all core configurations. including the limiting configuration that would yield the highest possible power density.

- Discussion of the administrative and physical constraints to prevent inadvertent addition of positive reactivity.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.5.2 Reactor Core Physics Parameters

In this section the applicant should discuss the core physics parameters and show the methods and analyses used to determine them. The information should include the following:

- Analysis methods and values for neutron lifetime and effective delayed neutron fraction. The applicant should describe the effects of reactor operating characteristics and fuel burnup.

- Analysis methods, values, and signs for coefficients of reactivity (e.g., fuel and moderator temperature, void, and power). The applicant should describe the effects of reactor operating characteristics and fuel burnup. This analysis, along with the analysis in Chapter 13, should show that reactivity coefficients are sufficiently negative to prevent or mitigate damaging reactor transients.

- The axial and radial distributions of neutron flux densities, justifications for the methods used, and comparisons with applicable measurements. The applicant should describe changes in flux densities with power level, fuel burnup, core configurations, and control rod positions. The information on neutron flux density should include peak-to-average values for thermal-hydraulic analyses. The applicant should validate these calculations by

comparing them with experimental measurements and other validated calculations.

- Technical specifications that control important design features, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.5.3 Operating Limits

The applicant should present the following information on reactor operating limits:

- Reactivity conditions, excess reactivity, and negative reactivity for combinations of control rods inserted that are analyzed for the limiting core and operating cores during the life of the reactor. The applicant should discuss operational and safety considerations for excess reactivity.

- The excess reactivity based on reactor temperature coefficients, poisons, and experiment worths. The applicant should justify the upper limit on excess reactivity to ensure safe reactor operation and shutdown.

- The amount of negative reactivity that must be available by control rod action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe shutdown state. The analyses should assume that the most reactive control rod is fully withdrawn (one stuck rod), non-scrammable control rods are at their most reactive position, and normal electrical power is unavailable to the reactor. The applicant should discuss how shutdown margin will be verified. The analyses should include all relevant uncertainties and error limits.

- The limiting core configuration that is possible with the planned fuel in this reactor. The limit should be imposed by the maximum neutron flux density and thermal power density compatible with coolant availability. The safety limits and limiting safety system settings for the reactor should be derived from this core configuration. The detailed analyses should be included in Section 4.6. Normal operating conditions and credible events, such as a stuck control rod, should be considered.

- A transient analysis assuming that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region. This analysis can also be based on a credible failure of a movable experiment. It should show that the reactor is not damaged and fuel integrity is not lost.

- The redundancy and diversity of control rods necessary to ensure reactor control for the considerations noted above.

- Technical specifications for safety limits, limiting safety system settings, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of this format and content guide. The applicant should justify these technical specifications in this section of the SAR.

4.6 Thermal-Hydraulic Design

In this section the applicant should present the information and analyses necessary to show that sufficient cooling capacity exists to prevent fuel overheating and loss of integrity for all anticipated reactor operating conditions, including pulsing, if applicable. The applicant should address the coolant flow conditions for which the reactor is designed and licensed, forced or natural-convection flow, or both. A detailed description of the analytical methods used in the thermal-hydraulic design should be provided. Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Estimates of the accuracy of the analytical methods should be included. The information should include the following:

- The coolant hydraulic characteristics of the core, including flow rates, pressures, pressure changes at channel exits and entrances, and frictional and buoyant forces. The applicant should address individual heated channels as well as the core as a whole for all flow conditions in the primary coolant system. The transition from forced to natural-convection flow for all forced-flow reactors should be calculated, and the applicant should prepare calculations for an event during which normal electrical power is lost.

- The thermal power density distribution in the basic fuel units and heat fluxes into the coolant of each channel and along the channel, derived from the fuel loading and neutron flux characteristics discussed above.

- Calculations and the thermal-hydraulic methodology for the transfer of heat to the coolant. The applicant should take into account uncertainties in thermal-hydraulic and nuclear parameters and such engineering factors as plate thickness, gap width, and the buildup of cladding oxides. The calculations should be based on fuel measurements and procurement specifications, as well as operating history and conditions. The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use.

- The calculations and experimental measurements to determine the coolant conditions ensuring that fuel and cladding integrity are not lost. The applicant should calculate at least the limiting core configuration. Operating conditions should include steady fission power, shutdown decay heat, planned pulses, and transients analyzed in Chapter 13. The applicant should take into account operational and fuel characteristics from the beginning to the end of fuel life.

- For the core geometry and the coolant thermal-hydraulic characteristics, a discussion to establish the fuel heat removal conditions that ensure fuel integrity such as fuel surface saturation temperature, onset of nucleate boiling, departure from nucleate boiling and/or flow instability. The discussion should show correlations among these factors and justify their use in deriving safety limits and limiting safety system settings for the technical specifications.

- The design bases for the primary coolant system, emergency core cooling system, and other systems designed to maintain fuel integrity, which should also be discussed in Chapter 5, "Reactor Coolant Systems." The analyses here and in Chapter 13 should describe loss-of-coolant scenarios for forced-flow reactors. Natural-convection cooling that removes decay heat to ensure thermal stability should also be discussed. Flow blockages should be analyzed in Chapter 13.

- Detailed analyses for a pulsing reactor containing descriptions of the core configurations; the bases of the feedback coefficients; the calculational model and assumptions; the thermal-hydraulic evolution during a pulse; core, transient rod, and fuel characteristics that determine the shape and magnitude of a pulse; and the safety considerations that establish limits to pulse sizes. Any changes in fuel parameters resulting from steady-power operation that could affect pulse characteristics should be analyzed. These changes could include burnup, hydrogen migration, cladding oxidation, and decrease in burnable poison, as applicable. The analyses should form the bases for technical specifications that limit reactor operating conditions, process variables, and pulse rod reactivity worths.

Appendix 4.1

Regulatory Guide 2.1

Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors

U.S. ATOMIC ENERGY COMMISSION

May 1973

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 2.1

SHIELD TEST PROGRAM FOR EVALUATION OF INSTALLED BIOLOGICAL SHIELDING IN RESEARCH AND TRAINING REACTORS

A. INTRODUCTION

Subdivision (b)(6) (iii) of section 50.34, "Contents of applications technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires an applicant for a license to include in his final safety analysis report plans for preoperational testing and initial operation. This regulatory guide describes a shield test program that is generally acceptable for evaluation of installed biological shielding in research and training reactors

B. DISCUSSION

Subcommittee ANS 6, Shielding, of the American Nuclear Society Standards Committee has developed a standard that describes an operational shield test program which may be used in evaluating the installed biological shielding in research and training reactors. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria, and its Secretariat. It was subsequently approved and designated ANSI N18.9-1972 by the American National Standards Institute (ANSI) on September 15, 1972

C. REGULATORY POSITION

The requirements and guidelines contained in ANSI N18.9-1972, "Program for Testing Biological Shielding in Nuclear Reactor Plants,"[1] approved September 15, 1972, are generally acceptable and, with due consideration for the unique characteristics of each research and training reactor, provide an adequate basis for conducting a shield test program during preoperational and startup testing for evaluation of installed biological shielding in research and training reactors subject to the following.

1. Section 3.2.4 of ANSI 18.9-1972 defines accessible areas, controlled areas, and unlimited access areas. Section 3.2.5 defines Maximum Permissible Dose rate. Nothing in these paragraphs should imply that exposures need not be controlled to the requirements of 10 CFR Part 20, "Standards for Protection Against Radiation."

2. Section 5.2 of ANSI 18.9-1972 states that procedures for implementing the minimum shield test program shall be prepared. These procedures should be designed so that exposures to personnel performing the test program are as low as practicable. These procedures should also be designed so that safety hazards to personnel performing the shield test program are properly identified For example, gas monitoring should be required where gases or vapors could affect the accessibility of an area.

3 Section 6 of ANSI N18.9-1972 specifies tests that should be conducted for evaluation of installed biological shielding. This section further specifies use of survey meters when conducting the required tests. The shield test program should also include provisions for gamma and neutron film mapping of critical areas where personnel exposure may occur due to streaming, cracks, or gaps in the shielding too small to detect by survey meters, e.g areas in the vicinity of beam holes, irradiation ports, or shielding areas directly aligned with the core

4 Section 9 of ANSI N18.9-1972 states that instruments used in carrying out the minimum shield test program shall have been calibrated prior to use in the test program and immediately after each survey The shield test program should also include provisions for

[1] Copies may be obtained from American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521

USAEC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the AEC Regulatory staff of implementing specific parts of the Commission's regulations to delineate techniques used by the staff in evaluating specific problems or postulated accidents or to provide guidance to applicants Regulatory Guides are not substitutes for regulations and compliance with them is not required Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission

Published guides will be revised periodically as appropriate to accommodate comments and to reflect new information or experience

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1 Power Reactors
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8 Occupational Health
9 Antitrust Review
10 General

calibrating all radiation survey monitors (both portable and installed) against a source emitting radiation of approximately the same type and intensity as that expected to be measured during the survey.

5. Sections 9.2 and 9.3 of ANSI 18.9-1972 provide requirements for survey instruments. In addition to these requirements, a survey instrument's range should be consistent with the actual dose range expected. For measurements conducted while a reactor is operating in the pulsed mode, appropriate instrumentation, such as film packets, which will properly respond to and measure radiation during the pulsed mode of operation should be provided.

5 REACTOR COOLANT SYSTEMS

In this chapter of the SAR, the applicant should give the design bases, descriptions, and functional analyses of the reactor coolant systems. The principal purpose of the coolant system is to safely remove the fission and decay heat from the fuel and dissipate it to the environment. The discussions should include all significant heat sources in the reactor and should show how the heat is safely removed and transferred to the environment.

The coolant in the primary systems of most non-power reactors serves more functions than just efficient removal of heat. It can act as a radiation shield for the reactor, fuel storage facilities, and, in some designs, experimental facilities and experiments. In open-pool reactors, the coolant is the only vertical shielding. In many designs the reactor coolant also acts as a core moderator and reflector. Because of these many functions of the reactor coolant, the design of the reactor coolant systems is based on selecting among interdependent parameters, including thermal power level, research capability, available fuel type, reactor core physics requirements, and radiation shielding.

Some non-power reactors are licensed to operate at such low power levels that no significant temperature increases will occur during normal operation. Such reactors may not require an engineered coolant system for heat removal. For those reactors, the applicant should, in Chapter 4, "Reactor Description," of the SAR, discuss the disposition of the heat produced, estimate potential temperature increases during operation, and justify why an engineered coolant system for heat removal is not required. In this chapter the applicant should summarize those considerations and conclusions.

For all other non-power reactors, the applicant should describe and discuss in this chapter systems to remove and dispose of the waste heat. The design bases of the reactor coolant systems for the full range of normal operation should be based on ensuring acceptable reactor conditions established in Chapter 4 of the SAR. The design bases of any features of the core cooling system designed to respond to potential accidents or to mitigate the consequences of potential accidents should be derived from the analyses in Chapter 13, "Accident Analyses." These features should be summarized in this chapter and discussed in detail in Chapter 6, "Engineered Safety Features." In this chapter the applicant should discuss and reference the technical specifications where analyses are used as the basis for a requirement.

In this chapter the applicant should describe all auxiliary and subsystems that use and contribute to the heat load of either the primary or secondary coolant system. Any auxiliary systems using coolant from other sources should be discussed in Chapter 9, "Auxiliary Systems."

The primary loops of the coolant systems of most licensed non-power reactors are of two basic types, forced-convection and natural thermal-convection. Facilities using forced-convection cooling also may be licensed to operate in the natural-convection mode. All non-power reactors with engineered coolant systems that do not have active decay heat removal systems should be capable of dissipating decay heat in the natural-convection mode. In this chapter the applicant should describe the complete coolant systems for the allowed modes of operation, as discussed below.

5.1 Summary Description

In this section the applicant should give a brief description of the reactor coolant systems, summarizing the principal features. Information should include the following:

- type of primary coolant: liquid, gas, or solid (conduction to surrounding structures)
- type of primary coolant system: open or closed to the atmosphere
- type of coolant flow in the primary coolant system: forced-convection, natural-convection, or both
- type of secondary coolant system, if one is present, and the method of heat disposal to the environment
- capability to provide sufficient heat removal to support continuous operation at full licensed power
- special or facility-unique features

5.2 Primary Coolant System

The basic requirements and design bases of the primary coolant system are to maintain reactor facility conditions within the range of design conditions and accident analyses assumptions derived from other chapters of the SAR, especially Chapters 4 and 13. The applicant should show the interrelationships among all SAR chapters and the way the designed primary coolant system provides all necessary functions. The following information should be included:

- Design bases and functional requirements of the primary coolant system.
- Schematic and flow diagrams of the system, showing such essential components as the heat source (reactor core), heat sink (heat exchanger),

pumps, piping, valves, control and safety instrumentation, interlocks, and other related subsystems.

- Tables of allowable ranges of important design and operating parameters and specifications for the primary coolant system and its components, including

  - coolant material
  - coolant flow rates
  - inlet and outlet temperatures and pressures throughout the system
  - elevation of components and water levels relative to the reactor core
  - construction materials of components
  - fabrication specifications of safety-related components
  - coolant quality requirements for operation and shutdown conditions, including pH and conductivity at a minimum
  - minimum coolant level

- Discussions and analyses keyed to drawings showing how the system provides the necessary cooling for all heat loads and all potential reactor conditions analyzed in the thermal-hydraulics section of Chapters 4 and 13, including the following:

  - Removal of heat from the fuel by forced-convection or natural-convection cooling, or both for those reactors licensed to operate in both modes. Discussion and analyses of the effect of the size, shape, and structural features of the primary vessel or pool on cooling characteristics; the function of the pool as a heat reservoir; and the effect of water depth on natural thermal convection cooling.

  - Transfer of heat from the primary coolant to a secondary coolant system for all reactor conditions. This discussion should include any heat exchanger design and operating conditions. Some non-power reactors may have only a primary coolant system that functions as a heat reservoir. For such systems, the analyses should include any factors that limit continuous operation, such as pool water temperature, and the proposed technical specifications that

ensure operation within the analyzed limits. Some high-power non-power reactors may require shutdown pumps to circulate coolant through the core to remove decay heat.

- Safe reactor shutdown, including passive or fail-safe transition from forced- to natural-convection cooling and removal of decay heat from the fuel. This discussion should include the loss of offsite electrical power.

- Locations, designs, and functions of such essential components as drains, syphon-breaks, pumps, isolation valves, and check valves. These components ensure that the primary coolant system is operable and that uncontrolled loss or discharge of contaminated coolant from the primary system does not occur. Radiological effects of potential coolant releases should primarily be analyzed in Chapter 11, "Radiological Protection Program and Waste Management."

- Discussion of the control and safety instrumentation, including location and functions of sensors and readout devices. The scram or interlock functions that prevent safety limits from being exceeded should be shown and discussed, including the related technical specifications.

- Description and function of any special features of the primary coolant system, such as removal of the neutron moderator for backup reactor shutdown.

- Brief description and functions of special features or components of the primary coolant system that affect or limit personnel radiation exposures from such radionuclides as nitrogen-16 and argon-41 and from radioactive contaminants and fission products.

- Description of radiation monitors or detectors incorporated into the primary coolant system and discussion of their functions.

- Brief discussion and references to detailed discussions in later sections of auxiliary systems using primary coolant, such as coolant cleanup, makeup water, nitrogen-16 control, emergency core cooling, experiment cooling, experimental facility cooling, and biological or thermal shield cooling. The direct effect of these auxiliary systems on the design and functioning of the primary coolant system should be discussed.

- Brief discussion of radiation shielding provided by the primary coolant. Most non-power reactors are submerged in a pool or tank so that the

primary coolant shields personnel above the pool and at the reactor room floor. The design bases for these shielding functions should be analyzed for anticipated reactor conditions in Chapters 4 and 11 of the SAR and for postulated reactor accidents in Chapter 13. The effect of any special shielding features, such as fuel storage facility shielding and experimental facility shielding (e.g., beam tubes), on the functioning of the primary coolant system should be discussed.

- Discussion of leak detection and allowable leakage limits, if any. Prompt detection of leakage is very important in reactors with heavy water.

- Discussion of normal primary coolant radiation concentration limits, including sampling frequency, isotopes of interest, and actions to be taken if limits are exceeded.

- For reactors that have closed systems, a discussion of allowable hydrogen limits in air spaces that are in contact with the primary coolant.

- Discussion of technical specification requirements for parameters of the primary coolant system, including the bases and surveillance requirements.

5.3 Secondary Coolant System

Many licensed non-power reactors include fuel cooling systems composed of both primary and secondary coolant systems. Some very low powered reactors contain no engineered coolant system, and some may contain a single component (primary) coolant system. Still other low-powered reactors may include a coolant system, with both primary and secondary subsystems, that would not support continuous reactor operation at full licensed power.

In this section the applicant should give information about those non-power reactors that include a secondary coolant system. For the others, the applicant should state that a secondary coolant system is not needed and should justify that conclusion. The following information should be provided:

- The design bases and functional requirements of the secondary coolant system, including whether the system is designed for continuous full-power reactor operation and whether it is shared with other facilities.

- Schematic and flow diagrams of the secondary coolant system, showing such essentials as how the heat exchanger connects the primary coolant system (the heat source) to the secondary coolant system, pumps, piping, valves, control and safety instrumentation, interlocks, and interface with the environment for ultimate release of the heat.

- Tables of the range of important design and operating parameters and specifications of the secondary coolant system, including the following:

  - Coolant material and its source.

  - Coolant flow rates.

  - Type of heat dissipation system, such as cooling tower, refrigerator, radiator, or body of water.

  - Location of heat dissipation system in relation to the reactor and the heat exchanger.

  - Construction materials and fabrication specifications of components. (For older facilities for which complete information may not be available, the applicant should make a best effort to provide this information and should discuss the operating history of components.)

  - Heat dissipation specifications related to environmental factors (e.g., temperature and humidity).

  - Specifications and limitations on coolant quality and corrosion of the secondary coolant system components including the environmental effects of the use of secondary coolant chemicals.

- Discussion and functional analyses keyed to the drawings showing how the system provides the necessary cooling for all potential reactor conditions. These discussions should address the following:

  - Inlet and outlet temperatures and pressures throughout the system, including the pressure differential between the primary and secondary coolant systems in the heat exchanger. (The applicant should discuss how the pressure in the secondary coolant system is maintained above that in the primary coolant system for all operating conditions, or analyze the radiological effect of leakage of contaminated primary coolant into the secondary coolant system. Isolation of the heat exchanger during shutdown periods is an acceptable method to control potential primary-to-secondary-system leakage if secondary coolant system pressure is lower than primary coolant system pressure only during periods of system shutdown. The applicant does not need to perform an analysis of primary-to-secondary-system leakage if secondary coolant system pressure is lower than primary coolant system pressure for only

short periods for system testing or repair. If the transfer of primary coolant into the secondary coolant system is caused by an abrupt event, such as a tube rupture in the heat exchanger, the analysis should be given in Chapter 13 and summarized here.)

- Control of heat removal from the secondary coolant system necessary to maintain fuel temperatures in the primary coolant system within the limits derived in the thermal-hydraulics analyses in Chapters 4 and 13 of the SAR.

- Removal of heat from the heat exchanger and release to the environment when the primary coolant system operates in all anticipated and licensed modes, including forced-convection flow and natural-convection flow, as applicable.

- Safe reactor shutdown and removal and dissipation of decay heat, including evaluation of the primary coolant system change from forced-convection flow to natural-convection flow if forced-convection flow is an allowed mode of operation.

- Response of the secondary coolant system to the loss of primary coolant with or without an emergency core cooling system.

- Locations, designs, and functions of such essential components as drains, sumps, pumps, makeup water, and check valves that ensure contaminated primary coolant is not inadvertently transferred to the secondary coolant system and released to the environment.

- Discussion of control and safety instrumentation, including locations and functions of sensors and readout devices and interlocks or safety capabilities.

- Descriptions of functions of any radiation monitors or detectors incorporated into the secondary coolant system. Discussion of surveillance to measure secondary coolant activity including frequency, action levels, and action to be taken.

- Brief comments and reference to detailed discussion in other sections of auxiliary cooling systems that transfer heat to the secondary coolant system, such as emergency core cooling system, experiment cooling, or biological or thermal shield cooling.

- Discussion of technical specification requirements, as appropriate, for the secondary coolant system, including the bases and surveillance requirements.

5.4 Primary Coolant Cleanup System

Most licensed non-power reactors contain solid fuel elements immersed in the primary coolant water. Experience has shown that the metal cladding is susceptible to corrosion if the chemical purity of the water is not high. The water purity must be above the usual purity of the potable water supply. Experience has shown that oxide buildup on aluminum-clad fuel operating at high power densities can reduce heat transfer. The rate of buildup depends, among other factors, on the water quality (Griess et al., 1964). This process should be evaluated in Chapter 4 of the SAR and summarized in this chapter if it contributes to determining the requirements for primary coolant purity. To delay or prevent component failure by corrosion, non-power reactors should have a primary coolant cleanup system. The purity of the primary coolant should be maintained as high as reasonably possible for the following reasons:

- to limit the chemical corrosion of fuel cladding, control and safety rod cladding, reactor vessel or pool, and other essential components in the primary coolant system

- to limit the concentrations of particulate and dissolved contaminants that could be made radioactive by neutron irradiation

- to maintain high transparency of the water for observation of submerged operational and utilization components

The applicant should give the following information:

- The design bases and functional requirements of the primary coolant cleanup system. Experience at non-power reactors has shown that with a well-planned water cleanup system and good housekeeping practices, primary coolant quality can be maintained within the following ranges:

  - electrical conductivity ≤ 5 µmho/cm
  - pH between 5.5 and 7.5

  The design bases should be derived from discussions in Chapter 4 of the SAR, and any recommendations of the fuel vendor also should be addressed.

- Schematic drawings and flow diagrams of the primary coolant cleanup loop.

- Table of specifications for the cleanup system demonstrating that it is designed for the volume and throughput of the primary coolant system. (For older facilities for which complete information may not be available, the applicant should make a best effort to provide this information and should discuss the operating history of components.)

- Locations and functions of control and monitoring instrumentation, including sensors, recorders, and meters. The discussion of monitors should include methods for continuously assessing coolant quality and effectiveness of the cleanup system.

- Locations and functional designs of cleanup system components such as branch points, pumps, valves, filters, and demineralizers.

- Discussion of schedules and methods for replacing or regenerating resins and filters and disposing of resultant radioactivity to ensure that radiation exposures do not exceed the limits discussed in Chapter 11 of the SAR.

- Summary of methods for predicting, monitoring, and shielding radioactivity deposited in filters and demineralizers from routine operations. The detailed discussion should be in Chapter 11.

- Summary of methods for predicting and limiting exposures of personnel in the event of inadvertent release of excess radioactivity in the primary coolant system and deposition in filters and demineralizers. The detailed discussion should be in Chapter 13.

- Provisions in the design and operation of the cleanup system to avoid malfunctions that could lead to significant loss of primary coolant or release of contaminated coolant, which could cause radiological exposure of personnel or release to the unrestricted environment to exceed the requirements in 10 CFR Part 20 and the facility ALARA (as low as is reasonably achievable) program guidelines.

- Discussion of technical specification requirements for the primary coolant cleanup system, including the bases and surveillance requirements.

5.5 Primary Coolant Makeup Water System

During operations at non-power reactors with a water-based primary coolant system, primary coolant must be replaced or replenished. Coolant may be lost as a

result of evaporation from open-pool systems, radiolysis, designed leakage as from pump seals, and other operational activities. The non-power reactor design should include a system or a procedure that meets the projected needs for coolant. The makeup water system need not be designed to provide a rapid, total replacement of the primary coolant inventory, but should be able to maintain the minimum acceptable water quantity and quality for reactor operation.

The applicant should provide the following information:

- The design bases for the primary coolant makeup water system that account for all activities that could cause a decrease in the primary coolant. A large loss-of-coolant event should be analyzed in Chapter 13 of the SAR. Although a required emergency core cooling system need not be a part of the makeup water system, it should be discussed in Chapter 6 if it exists.

- Schematic diagrams and functional discussions that show the source of water, the methods of addition to the primary coolant system, and the requirements for pretreatment before addition. Not all non-power reactors need a makeup water system attached to the reactor primary coolant system.

- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks. Methods should be discussed for tracking additions of makeup water to detect significant changes that might indicate leaks or other malfunction of the primary coolant system.

- Discussion of safety systems and administrative controls to ensure that the system or procedures for adding makeup water will not lead to significant loss of primary coolant and will prevent leakage of contaminated coolant into the potable water supply.

- Discussion of technical specification requirements for the primary coolant makeup water system, including the bases and surveillance requirements.

5.6 Nitrogen-16 Control System

When ordinary oxygen is irradiated with neutrons of sufficient energy, nitrogen-16, a high-energy beta and gamma emitter with a 7-second half-life, is formed. In water-cooled reactors operated above a few hundred kilowatts, the radioactivity of this nuclide may require specific systems or procedures for limiting personnel exposure.

In reactors with natural-convection cooling, transport of nitrogen-16 to the pool surface may be delayed by a coolant circulator or diffuser. In reactors with forced-

convection cooling, the coolant carrying the nitrogen-16 out of the core may be passed through a system such as a large shielded and baffled tank. This delay allows the radioactivity to decay significantly before the coolant emerges from the shielding. Another method of radiation control is to shield the entire primary coolant system.

The applicant should analyze the potential for personnel exposure to nitrogen-16 and propose control systems or procedures that include the following:

- Design bases and functional design of the nitrogen-16 control system or procedures. The design bases should be derived from analyses in Chapter 11 of the SAR.

- Schematic drawings and system and component specifications for the nitrogen-16 control system.

- The method used by the nitrogen-16 control system to reduce exposure rates and potential doses in occupied areas. Potential doses with and without the nitrogen-16 controls should be analyzed in Chapter 11 and summarized in this section of the SAR. These potential doses include dose from direct radiation and dose from airborne nitrogen-16.

- The effect of the nitrogen-16 control system on overall reactor safety and operation. For example, diffuser systems in natural convection reactors may affect coolant flow and coolant transparency.

- Other reactor design features affected by the nitrogen-16 control system. For example, a large shielded decay tank may affect coolant flow parameters, pump sizes, access for surveillance or inservice testing, or other factors for the primary coolant system.

- An assessment that the nitrogen-16 control system would not lead to an uncontrolled loss of primary coolant or the release of contaminated primary coolant that exceeds the requirements in 10 CFR Part 20 and the facility ALARA program guidelines. Methods for analyzing radiation exposures as a result of coolant release should be consistent with the analyses in Chapter 11.

- Discussion of any technical specification requirements for the nitrogen-16 control system, including the bases and surveillance requirements.

5.7 Auxiliary Systems Using Primary Coolant

In addition to the systems discussed above that are associated with the primary coolant system, other auxiliary cooling systems or shields in some non-power reactors may require the use of primary coolant and may affect the operation or safety of the reactor. Any function of the primary coolant that is principally shielding should be described in Chapter 4 and summarized in this section and Chapter 11 of the SAR. If the reactor design includes an emergency core cooling system, it should be described and discussed in Chapter 6. The following auxiliary systems that use primary coolant should be discussed in this section:

- experiment cooling
- experimental facility cooling
- experimental facility shielding (e.g., beam tubes)
- biological shield cooling
- thermal shield cooling
- fuel storage cooling and shielding

The applicant should provide the following information about these systems in this section:

- design bases and functional requirements of the auxiliary systems based on discussions elsewhere in the SAR, such as Chapters 4, 9, and 10, "Experimental Facilities and Utilization"

- schematic drawings and flow diagrams that show the source of water, locations of sensors and instruments, and locations of the components cooled or shielded

- tables of the range of important parameters of the systems and specifications of materials and components

- discussion of components to be cooled, the source of heat, the source of the coolant water, heat transfer to the coolant, and coolant heat dissipation

- summary of the shielding requirements and the protection factors provided by the coolant

- discussion of the provisions in the auxiliary system designs to prevent interference with safe reactor shutdown

- discussion of the provisions in the auxiliary system design to prevent the uncontrolled release of primary coolant or radiation exposures that would

exceed the requirements in 10 CFR Part 20 and the facility ALARA program guidelines

- requirements for minimum water quality
- discussion of any technical specification requirements for the auxiliary cooling systems, including the bases and surveillance requirements

5.8 Reference

Griess, J. C., et al., ORNL-3541, "Effect of Heat Flux on the Corrosion of Aluminum by Water," Part IV, Oak Ridge National Laboratory, Tennessee, February 1964.

6 ENGINEERED SAFETY FEATURES

In this chapter the applicant should discuss and describe engineered safety features (ESFs) for a non-power reactor. ESFs are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to the public, the facility staff, and the environment within acceptable values. The concept of ESFs evolved from the defense-in-depth philosophy of multiple layers of design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The need for ESFs is determined by the SAR analyses of accidents that could occur, even though prudent and conservative design of the facility has made the incidence of an accident very unlikely. It is also possible that for a particular design the SAR analyses will show that ESFs are not needed.

Normal operation of a non-power reactor is defined as operation with all process variables and other reactor parameters within allowed conditions of the license, technical specifications, applicable regulatory limits, and design requirements for the system. Accidents at non-power reactor facilities assume failure of a major component such as the reactor coolant system boundary or a reactivity addition event. Licensees analyze a maximum hypothetical accident that assumes an incredible failure that leads to breach of the fuel cladding or a fueled experiment containment. These postulated accidents are compared with acceptance criteria such as the safety limits from the technical specifications or, where there are radiological consequences, to accepted regulatory limits (10 CFR Part 20 or 100). The results of the accident analyses are presented in SAR Chapter 13, "Accident Analyses." ESF systems must be designed to function for the range of conditions from normal operation through accident conditions.

Because most non-power reactors operate at atmospheric pressure, at relatively low power levels, and with conservative safety margins, few credible postulated accidents result in radiological risk to the public. The analyzed accident scenarios that the applicant should present in SAR Chapter 13 include the following:

- loss of coolant
- loss of coolant flow
- insertion of excess reactivity (rapid or ramp)
- loss of fuel cladding integrity or mishandling of fuel
- failure or malfunction of an experiment
- other uncontrolled release of radioactive material
- loss of electric power
- external events such as floods and earthquakes

The SAR accident analyses for many non-power reactors may show that ESFs are not required, even for the maximum hypothetical accident. In other cases the

accident analyses may show that ESFs need to be considered in mitigating the potential release of radioactive material to the environment.

The accident analyses provide the design bases for any required ESF. The ESF design should be as basic and fail safe as practical. Because non-power reactors are conservatively designed, few, if any, accidents should require redundant or diverse ESF systems. However, consideration should be given to adding redundancy and diversity to ESF systems if the reactor is of a higher power level (2 MW or greater thermal power level), if an ESF system would be susceptible to loss of capability to function because of a single failure, or if the radiological consequences to the public of the accident that the ESF is designed to protect against would be very serious if the ESF were to not function.

In addition to the design and functional characteristics of each ESF, the applicant should describe the methods and criteria for testing to demonstrate ESF system operability. The functional requirements, related setpoints, interlocks, and bypasses for each ESF should be described, analyzed, and included in the facility technical specifications. The technical specification surveillance requirements for system components that ensure the integrity and operational capability of the ESFs should be identified and discussed in the SAR.

The discussion should include how the ESFs interact with site utilities, such as electrical power and water and, if applicable, how the transfer between normal and emergency sources of electricity and water is accomplished. The applicant should discuss and demonstrate the need for site utility redundancy or diversity and the specific design features that provide it for each ESF component.

The SAR should include schematic diagrams, showing all components, their interrelationships, and the relationship of each ESF to other reactor systems (e.g., the core cooling system or the reactor room ventilation system). It should include a brief description of the instrumentation and control (I&C) system for each ESF, with detailed descriptions presented in SAR Chapter 7, "Instrumentation and Control Systems." The material presented should show how I&C systems necessary for ESF operation are designed to function in the environment created by the accident.

Typical ESFs that may be required at non-power reactors are (1) the confinement, (2) the containment, and (3) the emergency core cooling system (ECCS). In addition, features required in the facility heating, ventilation, and air conditioning (HVAC) system to mitigate the consequences of accidents should be treated as part of the ESFs of the confinement or containment system. The applicant should discuss any additional ESFs in a comparable way.

Brief definitions and illustrations of the confinement, the containment, and the ECCS follow:

(1) The confinement is an enclosure of the overall facility (e.g., a reactor room) that is designed to limit the exchange of effluents between the enclosure and its external environment to controlled or defined pathways. A confinement should include the capability to maintain sufficient internal negative pressure to ensure inleakage (i.e., prevent uncontrolled leakage outside the confined area), but need not be capable of supporting positive internal pressure or significantly shielding the external environment from internal sources of direct radiation. Air movement in a confinement could be integrated into the HVAC systems, including exhaust stacks or vents to the external environment, filters, blowers, and dampers.

(2) The containment is an enclosure of the facility designed to (a) be at a negative internal pressure to ensure inleakage, (b) control the release of effluents to the environment, and (c) mitigate the consequences of certain analyzed accidents. The containment is designed (a) to be sealed to support a defined pressure differential across it and (b) to have a defined upper limit on leakage from it. Both design conditions are testable. An accident scenario that might require a containment for a non-power reactor would involve positive internal pressures, either static or transient, or the need to shield the external environment from internal sources of direct radiation, or both. Exhaust stacks, vents, particulate filters, activated charcoal filters, or piping may be provided for controlled venting of a containment, and the design should provide for both normal and emergency operational modes. A containment may be designed to be integral with the facility HVAC and liquid waste systems.

Most non-power reactors can be designed, sited, and operated so that a normal building or, at most, a confinement can be used to house the reactor; a containment would not be required. In contrast to a containment, a confinement plus its HVAC system

- usually responds to accidents by reducing and changing the airflow paths to and from the building (a containment seals the building from the environment and significantly reduces releases of radioactive material to the environment)
- has doors with gasket-type seals (airlocks for containments)
- may not have sealing isolation dampers on air penetrations (sealing isolation dampers for containments)

- cannot maintain as high a negative differential pressure as a containment
- is not as leaktight as a containment and the leak rate normally cannot be confirmed through testing
- cannot control the release from an event that results in positive pressure in the reactor building
- usually has less direct radiation shielding capacity than a containment because the walls are thinner
- is less resistant than a containment to challenges placed on the building by the external environment

If the analyses show that a confinement ESF will mitigate the consequences of the most limiting accident scenario to acceptable levels, a containment ESF would not be required, although some licensees have chosen to build containments as an additional design conservatism.

(3) An ECCS is designed to provide a source of coolant to limit fuel damage from decay heat should primary cooling be lost from the reactor core region.

The issue of what standards to use in evaluating accidents at a non-power reactor was discussed in an Atomic Safety and Licensing Appeal Board (ASLAB) decision issued May 18, 1972, for the research reactor at Columbia University in New York City. The ASLAB stated that "as a general proposition, the Appeal Board does not consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The NRC staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents, since most research reactors to date have been able to meet the conservative 10 CFR Part 20 criteria. American National Standards Institute/American Nuclear Society ANSI/ANS-15.7, "Research Reactor Site Evaluation," contains additional information on doses to the public from releases of radioactive material.

The objective of non-power reactor ESFs is to ensure that projected radiological exposures from accidents are kept below the regulatory limits. The regulations defining the limits on releases from non-power reactors during accident conditions depend on whether the non-power reactor (see 10 CFR 50.2) is a test reactor (also

called a testing facility, see 10 CFR 50.2) or a research reactor (see 10 CFR 170.3). For a research reactor, the results of the accident analyses have generally been limited to values from the old 10 CFR Part 20 (10 CFR 20.1 through 20.602 and appendices). The exposure values that the staff has generally found acceptable for research reactors are less than 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. However, in several instances, the staff has accepted very conservative accident analyses with results greater than the 10 CFR Part 20 dose limits discussed above. Research reactors that received their initial operating license after January 1, 1994, must show that exposures meet the requirements of the revised 10 CFR Part 20 (10 CFR 20.1001 through 20.2402 and appendixes). Occupational exposure is discussed in 10 CFR 20.1201, and public exposure is discussed in 10 CFR 20.1301.

If the facility meets the definition of a test reactor, the exposures will be compared with the doses in 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses in 10 CFR Part 100 are reference values. References to 10 CFR Part 100 in this chapter pertain to test reactors only.

6.1 Summary Description

In this section of the SAR, the applicant should briefly describe all of the ESFs in the facility design and summarize the postulated accidents they are designed to mitigate. These summaries should include the design bases and performance criteria and contain enough information for an overall understanding of the functions of the ESFs and the reactor conditions under which the equipment or systems must function.

Simple block diagrams and drawings may be used to show the location, basic function, and relationship of each ESF to the facility. Detailed drawings, schematic diagrams, data, and analyses should be presented in subsequent sections of this chapter for specific ESFs.

6.2 Detailed Descriptions

In this section of the SAR, the applicant should discuss in detail the particular ESFs incorporated into the reactor design. Not all of these ESFs are found in any single design. Other systems in addition to the systems discussed in this section may be considered ESFs. The applicant should discuss these ESFs in a manner similar to the discussions in this section.

6.2.1 Confinement

The applicant should discuss in detail the confinement and the associated HVAC systems that function as ESFs. For the confinement to function as an ESF, the design bases for the consequence-mitigation functions should be derived from the accident analyses in SAR Chapter 13.

Confinements and HVAC systems may also have functions that are not considered functions of ESFs and that need not be addressed in this chapter. Most non-power reactors release small quantities of airborne radioactive material, primarily argon-41 gas, to the environment during normal operations. To protect the health and safety of the public and the staff, it may be necessary to control airflow through the reactor room and release the reactor room air in a controlled manner at a location that allows for dilution and diffusion of the radioactive material before it comes in contact with the public. In some cases, it may also be efficient to use the confinement and HVAC systems to prevent an uncontrolled release to the environment of radioactive effluents resulting from operation. This aspect of the use of ESFs during normal system operation is not considered an ESF function. However, the design bases and detailed discussions of these systems for normal operations to control releases should be given in Chapter 3, "Design of Structures, Systems, and Components," and Chapter 9, "Auxiliary Systems." Diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be discussed in Chapter 11, "Radiation Protection Program and Waste Management."

A radioactive release need not be a rapid or burst-type release. It also includes the leakage and diffusion of airborne radioactivity from a room through cracks or gaps in building structural components. Such releases could be controlled by a system of ducts, louvers, blowers, exhaust vents, or stacks. Non-power reactors should have the capability to quantify releases and calculate potential exposures in both restricted and unrestricted areas. Calculating potential exposures provides the bases for actions to ensure that the public is protected during both normal operation and accident conditions.

If the confinement and HVAC or air (stack) exhaust systems are designed to change state or operating condition in response to a potential accident and, in so doing, mitigate the radiological consequences of the accident, those features should be designated as ESFs and should be described in detail. The discussion of the ESF functions should demonstrate how dispersion or distribution of contaminated air to the environment or occupied spaces other than the reactor room is controlled. The discussion should include the design bases for the location and operating characteristics of the air exhaust stack, if applicable, and the design bases for effluent monitoring systems.

The discussion of mitigative effects should contain a comparison of potential radiological exposures to the facility staff and the public with and without the ESF. Either operational data for an operating facility or results of analyses for a new facility should be presented showing airflow rates, reduction in quantities of airborne radioactive material by filter systems, system isolation, and other parameters that demonstrate the effectiveness of the system.

A schematic diagram of the system should be presented showing the blowers, dampers, filters, other components necessary for operation of the system and flow paths. Automatic and manual trip circuits, bypasses, interlocks, and special I&C systems for the ESF system should be described briefly in this section and in detail in Chapter 7.

In this section the applicant should develop requirements to be specified in the technical specifications for system operability, periodic surveillance, setpoints, and other specific requirements to ensure a functional ESF system during postulated events. Examples include the requirement for operability of the ESFs during reactor operation or other significant events such as fuel movement. Periodic functional testing of damper closure, room isolation, minimum airflow rates, automatic system shutdown and startup, and activation setpoints should be required and specified. See Chapter 14, "Technical Specifications," of this format and content guide, for details on what technical specification requirements should be identified and justified in this section.

6.2.2 Containment

Because the potential risk to the public is generally low from accidents at non-power reactors, few require a containment. However, a containment and associated HVAC system required at a non-power reactor to mitigate the consequences of a postulated accident are considered ESFs. As previously discussed, containments are required as an ESF on the basis of the reactor facility design, operating characteristics, accident scenarios, and location.

A containment for a non-power reactor should be designed to prevent the rapid, uncontrolled release of radioactive material to the environment. A possible scenario for such a release could be an accident in the reactor core that involves a large loss of fuel cladding integrity (multiple plates or pins), the release of fission products into the primary coolant system, and the rapid release of fission products from the primary coolant system into the reactor room. The containment is designed to control the release to the environment of airborne radioactive material released in the reactor room even if the accident is accompanied by a pressure surge or a steam release within the room. The thick walls of the containment may also help mitigate direct radiation exposure during certain accidents. The analyses in Chapter 13 of the SAR should include details of the postulated scenario,

including the assumptions and justification for the initiating event, the progression of the scenario, the consequence-mitigating effects of the containment, and the potential radiological exposures to the most exposed member of the public. The design bases for the containment should include the postulated peak pressures, the duration of the event, the pressure-versus-time envelope, the time during which containment integrity must be maintained while recovery from the event is implemented, limits on leakage or controlled release from the containment to the environment, the quantity of failed fuel, and the quantity and type of released radioactive material.

A radioactive release need not be a rapid or burst-type release. It also includes the leakage and diffusion of airborne radioactivity from a room through cracks or gaps in building structural components. Such releases could be controlled by a system of ducts, louvers, blowers, exhaust vents, or stacks. Non-power reactors should have the capability to quantify releases and calculate potential exposures in both restricted and unrestricted areas. Calculating potential exposures provides the bases for actions to ensure that the public is protected during both normal operation and accident conditions.

The description must include the bases for the protection factors provided by the containment. The goal is that the containment should reduce the consequences to the public, the facility personnel, and the environment to acceptable values as specified above.

In this section the applicant should explain how the design and functional details of the containment meet the design bases and criteria described above. System drawings, component and material specifications, and structural details should be included. The information should demonstrate that the radiation protection factors assumed in the accident analyses are provided. The design bases and the discussions should describe how the containment functions over the range of normal operation and the events that initiate switching to the emergency mode. The discussions should address which reactor operations and evolutions require the containment to be operable, and whether an emergency electrical power source is required to be operable.

To qualify as a containment, the reactor building should be a robust structure with airlocks and all other penetrations sealed (e.g., cable penetrations sealed with epoxy) or sealable (e.g., hydraulic dampers on ventilation penetrations). The building should be capable of maintaining a negative pressure in relation to the atmosphere (e.g., at least -½ inch water) during normal operation and have a measurable leakage rate (e.g., less than 5 percent over 24 hours). The actual performance requirements are determined from the accident analyses in Chapter 13 of the SAR. For example, the normal function of the containment ventilation exhaust system may be divided into two trains—one that ventilates the reactor

room and a second that ventilates areas with high airborne radiation generation such as experimental facilities or fume hoods. The ventilation system is normally equipped with high-efficiency particulate filters, and the accident ventilation system has a separate train(s) equipped with high-efficiency particulate and activated charcoal filters to sorb iodine.

Automatic containment trip circuits, interlocks, special I&Cs, and monitoring requirements for the ESF should be described. The description should detail their relationship and interaction with the I&C systems for normal operation described in SAR Chapter 7.

The discussion should give the technical specifications and their bases to ensure that the containment ESF is operable when required. The technical specifications also should provide for necessary surveillance, testing, and maintenance of the containment components to ensure operability. The technical specifications should define an operable containment ESF and describe the reactor conditions and operations for which the containment shall be operable. See Chapter 14 of this format and content guide for details on what technical specification requirements should be identified and justified in this section.

6.2.3 Emergency Core Cooling System

An ECCS may be required at some non-power reactors to remove decay heat from the fuel to prevent failure or degradation of the cladding if cooling is lost. The applicant should give the analysis of the ECCS if one was identified as needed in the Chapter 13 accident analyses.

A schematic diagram should show the relationships among the major system components such as valves, spray headers, pumps, piping, and any I&C systems. Special ECCS I&C systems should be described briefly in this section and described fully in Chapter 7. In this section, the applicant should discuss any effects of the ECCS design on normal operations and reactor safety. Analyses for non-power reactors should demonstrate that fuel failure will be prevented for postulated-accident scenarios.

If the ECCS is a passive system (e.g., a gravity-feed spray from a storage tank), a complete description with associated analyses and data should show how coolant flow is initiated and why the system is effective. The information should demonstrate that the ECCS will provide core cooling capacity in terms of minimum flow and time of operation for all loss-of-coolant accidents considered.

If the ECCS is an active system that requires sensors and an action or event to initiate operation, descriptions should include details of initiation response times and backup or redundant sensing and control systems. The discussion should

include the source of electrical power, source of coolant, heat sink, or other systems required to operate the ECCS and show how operability and availability are ensured.

The facility design should show how radioactive material, such as emergency coolant, is controlled.

In this section the applicant should also give the bases for technical specifications that ensure that the ECCS is available and operable when required. Technical specifications should include minimum operability requirements and the possible operations and conditions under which the ECCS would be required. Test and surveillance functions and intervals should be stated in the technical specifications to ensure operability of the ECCS. See Chapter 14 of this format and content guide for details on what technical specification requirements should be identified and analyzed in this section.

6.3 References

American National Standards Institute/American Nuclear Society, ANSI/ANS-15.7, "Research Reactor Site Evaluation," ANS, LaGrange Park, Illinois, 1977.

Atomic Safety and Licensing Appeal Board, "In the Matter of Trustees of Columbia University in the City of New York," May 18, 1972.

7 INSTRUMENTATION AND CONTROL SYSTEMS

In this chapter of the SAR the applicant describes and discusses the design and operating characteristics of the instrumentation and control (I&C) systems. Sufficient information should be included to explain the design criteria and bases, and to discuss the functional and safety analyses of the I&C subsystems. The I&C subsystems generally comprise the reactor control system (RCS), process instruments, the reactor protection (safety) system (RPS), instruments to initiate operation of engineered safety features (ESFs), and radiation safety monitoring systems. These systems and their outputs can be consolidated into a control console, along with the devices and circuits that control the operation of the reactor. The guidance in this chapter of the SAR is based on the principle that most non-power reactors can be designed and operated to pose acceptably small or insignificant risk to the public without isolating or separating the RPS from other subsystems. Additional design features, such as separation of systems, may be necessary for high-power test reactors. Applicants who need additional guidance beyond that given in this chapter should contact their project manager.

The non-power RPS should monitor selected reactor operating parameters such as neutron flux; fuel temperature (monitored primarily in TRIGA-type reactors); primary coolant flow, temperature, and level; and radiation intensity. The RPS is designed to ensure reactor and personnel safety by limiting parameters to operate within analyzed operating ranges. The RPS can also give the ESF actuation system information for the operation of ESFs when the instruments indicate that abnormal or accident conditions could occur. The RCS may monitor many of the same parameters as the RPS and give information for automatic or manual control of the reactor operating conditions (e.g., reactor power, by inserting or withdrawing control rods). The reactor facility instruments present operating parameter and system status information to the operator for monitoring reactor operation and for deciding on manual control actions to be taken. Instrument systems are the means through which automatic or operator control actions are transmitted for execution by the RCS. Radiation instruments show radiation levels in selected areas in the reactor facility and could give data to the RPS, give information to help in the control of personnel radiation exposure, or monitor the release of radioactive material from the reactor and the reactor building.

In this chapter, the applicant should discuss the functional requirements, design criteria and bases, system descriptions, system performance analyses, and the bases of technical specification limiting safety system settings (LSSSs), limiting conditions of operation (LCOs), and surveillance requirements for the I&C systems for non-power reactors.

7.1 Summary Description

In this section of the SAR, the applicant should briefly describe the I&C systems of the reactor, including block, logic, and flow diagrams showing major components and subsystems, and connections among them. The applicant should summarize the technical aspects, safety, philosophy, and objectives of the I&C system design and should discuss such factors as redundancy, diversity, and isolation of functions. The information should include:

- *Type of instruments*—System instruments should be described by type [e.g., hardwired analog, computerized digital that uses stored programs (software), or combinations of these]. If a combination is used, the applicant should clearly note which portions or functions are analog and which are computerized digital, and how they relate to each other. The applicant could refer to existing systems reviewed and approved by NRC that are similar to the described system.

- *Classification of systems*—I&C systems and equipment should be classified into categories by function performed (e.g., the RCS, RPS, ESF actuation system, control console and display instrument systems, and radiation protection instruments).

The general description of each category of I&C subsystem should include the following, as applicable:

- For the RCS, a brief discussion of each major subsystem such as manual control system, automatic control system, control rod drive systems, bypass and interlock systems, and any integrated experiment I&C systems.

- For the RPS, the types of parameters monitored, both nuclear and non-nuclear, the number of channels designed to monitor each parameter, the actuating logic that determines the need for actions to change reactor conditions and that takes these actions, and number and type of reactivity control devices.

- For the ESF actuation system, a discussion of the subsystems that detect the need for operation and that initiate operation including identification of the parameters monitored or the source of input information and the number of channels designed to monitor, process, and act on the information.

- For the control console and display instruments, a discussion of the parameter display systems and equipment by which the operator can observe and control the operation of the reactor and important subsystems

- For radiation protection instruments, a brief discussion of area and effluent radiation detection systems that monitor, alarm, or provide input to other subsystems of potentially hazardous radiation levels. The applicant should address radiation systems that monitor effluent streams from the reactor facility, state the type of effluent (such as airborne or liquid), and list alarms or signals to other subsystems.

- A summary of the human-machine interface principles used in the location of instrumentation and controls.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

In this section of the SAR, the applicant should discuss the criteria for developing the design bases for the I&C systems. The basis for evaluating the reliability and performance of the I&C systems should be included. All systems and components of the I&C systems should be designed, constructed, and tested to quality standards commensurate with the safety importance of the functions to be performed. Where generally recognized codes and standards are used, they should be named and evaluated for applicability, adequacy, and sufficiency. They should be supplemented or modified as needed in keeping with the safety importance of the function to be performed. Evaluations and modifications of the standards should be described in the SAR. A set of generally applicable criteria for use as a guide is given below. Criteria that are used should be clearly stated and should be shown to provide the appropriate level of reliability, safety, and performance capability. The applicability of these criteria should be determined from the operating analyses in Chapter 4, "Reactor Description," and accident analyses in Chapter 13, "Accident Analyses," of the SAR.

- Systems and components (including I&C systems) determined by the analyses in the SAR to be important to the safe operation or shutdown of the reactor should be designed to be in accordance with local building and siting codes, and should be able to withstand the effects of natural phenomena without loss of capability to perform their safety function (see Chapter 3, "Design of Structures, Systems, and Components," for additional information).

- I&C systems and components determined in the SAR analyses to be important to the safe operation or shutdown of the reactor should be designed, located, and protected so that the effects of fires or explosions would not prevent them from performing their safety functions.

- I&C systems and components determined in the SAR to be important to the safe operation or shutdown of the reactor should be designed to function reliably under anticipated environmental conditions (e.g., temperature, pressure, humidity, and corrosive atmospheres) for the full range of reactor operation, during maintenance, while testing, and under postulated accident conditions, if the systems and components are assumed to function in the accident analysis.

- The RPS should be designed to automatically initiate the operation of systems or give clear warning to the operator to ensure that specified reactor design limits are not exceeded as a result of measured parameters indicating the onset of potential abnormal conditions. The ESF actuation system should be designed to automatically initiate operation to mitigate the consequences of abnormal conditions or accidents.

- I&C systems should be designed to have functional reliability, including redundancy and diversity, commensurate with the safety functions to be performed and the consequences of failure of the system to perform the safety function. For example, an I&C system for a non-power reactor should be designed to perform its protective function after experiencing a single random active failure within the system.

- I&C systems should be designed to fail into a safe state on loss of electrical power or exposure to extreme adverse environments.

- I&C systems should be designed so that a single failure will not prevent the safe shutdown of the reactor.

7.2.2 Design-Basis Requirements

I&C system design requirements for non-power reactors are generally derived from the results of analyses of normal operating conditions and of accidents and transients that could occur. This section provides guidance on the factors to consider in developing the analyses and the design bases. Design bases for the I&C system, subsystems, and components should include the following, as applicable:

- The function or purpose of systems or instruments considering which reactor parameters are monitored or controlled.

- The range of values that monitored variables may exhibit for normal operation, shutdown conditions, and for postulated accidents.

- Safety or control functions and any unique or facility-specific functions performed by the I&C system or subsystems.

- Specification of alarm, trip, and actuation setpoints derived from accident or other operational analyses of the instrumented system or function.

- Any special requirements such as redundancy, diversity, quality assurance, and environmental requirements derived from the results of analyses of the full range of operating conditions and postulated accidents.

- The specification of precision and accuracy requirements for the instruments, control subsystems, or components.

- The specification of number and type of channels required to monitor variables.

- The system operational and support requirements such as those for electrical, mechanical, structural, cooling, heating, and signal input.

- The requirements that controls and instruments be grouped and located so that operators can easily reach and manipulate the controls while readily observing on meters and displays the results of their actions (operator interface requirements).

- For digital computer systems, in addition to the foregoing, the applicable guidelines from IEEE 7-4.3.2–1993, "IEEE Standard Criteria for Digital Computers Systems in Safety Systems of Nuclear Power Generating Stations," for the design, application, and evaluation of digital computer hardware and software and ANSI/ANS-10.4–1987, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," for evaluating the verification and validation programs for software for use in the I&C system. Regulatory Guide (RG) 1.152, which discusses the use of digital computers in nuclear power plant safety systems is attached as Appendix 7.1. However, neither of these standards was uniquely developed for non-power reactors and may contain sections and requirements that do not apply to a particular situation. Furthermore, the technology and safety principles on which computerized I&C systems are based are changing. Until additional guidance is available, applicants should request current requirements from NRC. The testing programs for computer systems help to verify that the software will not cause unintended effects under some combinations of circumstances or conditions, or some malfunctions. ANSI/ANS Standard 15.15–1978, "Criteria for the Reactor Safety Systems of Research Reactors," and draft ANSI/ANS Standard 15.20, "Criteria for the Control

and Safety Systems for Research Reactors," are general guides for the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable. The staff reviewed and accepted a digital control system developed by General Atomics. NRC licensees for several TRIGA reactors have installed this system (see Amendment No. 19 to Facility Operating License No. R-84, Docket No. 50-170 for the Armed Forces Radiobiology Research Institute TRIGA reactor, July 23, 1990. Amendment No. 29 to Facility Operating License No. R-38 for the General Atomics TRIGA Mark I reactor is attached to this chapter as Appendix 7.2). The staff reviewed and accepted a digital control system developed by Atomic Energy Canada Limited. An NRC licensee for a TRIGA reactor installed this system (Amendment No. 30 to Facility Operating License No. R-2 for the Penn State Breazeale Reactor is attached to this chapter as Appendix 7.3)

- Consult NRC Generic Letter 95-02 for I&C systems that are being upgraded to systems with digital technology; it is attached to this chapter as Appendix 7.4.

7.2.3 System Description

The system description in the SAR should include equipment and major components as well as block, logic, and schematic diagrams. The applicant should also submit hardware and software descriptions and software flow diagrams for digital computer systems. The applicant should describe how the system operational and support requirements will be met and how the operator interface requirements will be met. The description should also address the methodology and acceptance criteria used to establish and calibrate the trip or actuation setpoints, or interlock functions.

7.2.4 System Performance Analysis

The applicant should conduct a performance analysis of the proposed I&C system to ensure the design criteria and design bases are met and license requirements for the performance of the system are specified. The system performance analysis should encompass the following:

- The SAR should describe the operation of the I&C system and present the analysis of how the system design meets the design criteria and design bases. The discussion should include accuracy, reliability, adequacy and timeliness of I&C system action, trip setpoint drift, quality of components and, if required by the analyses, redundancy, independence, and how a single failure affects both its ability to perform its safety function and the effect on operation or safe shutdown of the reactor.

- Technical specification LSSSs, LCOs, and surveillance requirements for the I&C system should be established. These parameters and requirements should include system operability tests, trip or actuation setpoint checks, trip or actuation setpoint calibrations, and any system response-time tests that are required. Surveillance intervals should be specified and the bases for the intervals, including operating experience, engineering judgment, or vendor recommendation. should be discussed.

7.2.5 Conclusion

The applicant should summarize in this section of the SAR why the system design is sufficient and suitable for performing the functions stated in the design bases.

7.3 Reactor Control System

The RCS performs several functions, such as maintaining the reactor in a shutdown state, reactor startup, changing power levels, maintaining operation at a set power level, and shutting down the reactor. In non-power reactor designs that allow pulsing such as the TRIGA design), the RCS can rapidly insert reactivity into the reactor core to produce a predetermined high-power pulse of short duration, or to achieve a rapid increase in reactor power in a "square wave." The RCS may be discussed using such subsystems as nuclear instruments, process instruments, control elements, and interlocks. In describing each subsystem in the SAR, the applicant should include design considerations and technical specification requirements.

In the nuclear instrument system, nuclear instruments monitor the neutron flux from the subcritical source multiplication range, through the critical range, and through the intermediate flux range to full power. Neutron flux instruments also should determine the startup rate and, in some designs, reactor period information. Linear and log neutron flux channels should be used to monitor the core neutron flux while control rods are withdrawn or inserted to increase or decrease reactor power. At least one linear neutron flux channel should be calibrated to reactor thermal power.

The process instruments are designed to measure and display such parameters as coolant flow, temperature, or level; fuel temperature; or air flow parameters within or from the reactor room. In some designs, this information may also be sent to the RPS.

The typical non-power reactor has an automatic control (servo) system that controls the reactor power about a point set by the operator. Most servo control systems compare the output of a linear neutron flux channel against an adjustable voltage representing the desired power level, and automatically change the position

of a regulating rod in the core to change the neutron flux density to reduce the difference between the two voltages until the actual reactor power level is very nearly equal to the desired power level. This process can be performed by analog control equipment or by software in a digital computer system.

Reactors with pulsing capabilities have a transient rod that, on command, is rapidly ejected out of the core to a pre-programmed distance. This action rapidly inserts a known amount of excess reactivity into the core that pulses the core power to very high levels for very short intervals. The system can also be used to form a square-wave power increase to a predetermined steady-state power level.

The RCS for non-power reactors should have a set of equipment protection interlocks and inhibits that prohibit or restrict operation of the reactor unless certain conditions are met. For example, there should be an interlock that prohibits control rod motion unless the neutron flux in the core produces a neutron count rate sufficient to help ensure that nuclear instruments are responding to neutrons. There may be additional equipment protection interlocks to ensure, for example, that there is sufficient coolant flow, shielding is intact, ventilation air is flowing, coolant level is sufficient, and required neutron instruments and recorders are functional. There may also be personnel protection interlocks to prevent reactor operation if certain radiation fields are excessive. Control rods may be run back to automatically reduce the reactor power when certain specified reactor conditions approach a predetermined limit, but total reactor shutdown (scram) is not warranted.

Experimental facilities may be interlocked with the RCS to prevent reactor operation if the experimental facility is not in the correct configuration. If experiments conducted in non-power reactors could interact with the core to change reactivity or otherwise modify the reactor operating conditions, data to the RCS or RPS from the experiment instruments may be needed to detect reactivity changes. All experiments should be carefully considered for interaction with the I&C system when the safety analysis for the experiment is performed. The analysis should consider any interaction with the RCS or RPS. Where such interactions are warranted, they should meet the standards used for the design of the systems to which the experimental facilities will be connected.

The applicant should include the following for each RCS subsystem:

- Discuss the design criteria for the RCS as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.

- Discuss the design bases information specified in Section 7.2.2 and any additional design bases of facility-specific subsystems.

- Describe the system as specified in Section 7.2.3, including any additional system descriptive material specific to subsystem design and implementation not covered in Section 7.2.

- Analyze the operation and performance of the system as specified in Section 7.2.4 including analyses and results of any features or aspects specific to the facility design and implementation not specified in Section 7.2. Include the bases of any technical specifications and surveillance tests with intervals specific to the design and operation of the systems. Address the specific design features of the RCS, such as the following:

  - Detector channels directly monitor the neutron flux density for presentation of reactor power level and power rate-of-change.

  - The RCS gives a continuous indication of the neutron flux density from subcritical multiplication source level through the full licensed power range. If multiple detector channels are used, this continuous indication should overlap a minimum of one decade during detector changeover.

  - The RCS has a reactor period channel that covers subcritical neutron source multiplication from the approach to critical, through critical, and into the licensed power range. Depending on the analysis in the SAR, some reactors may not have this channel.
  - The RCS protects against a failure or operation in a mode that could prevent the RPS from performing its intended safety function.

  - The system and equipment are designed to assume a safe state on loss of electrical power.

  - The RCS has at least two channels of reactor power indication through the licensed power range.

  - The startup and operating power detector channels can discriminate against strong gamma radiation, such as that present after long periods of operation at full power, to ensure that indicated changes in neutron flux density are reliable.

  - The reactor power indication of at least one channel should remain reliable for some predetermined range above the licensed power level. For reactors with power level as a safety limit, the instrumentation should be able to indicate if the safety limit was exceeded. For other reactor types, at least one channel should be

able to indicate if the power level was exceeded, which is the basis for limiting licensed power level.

- All control rod positions should be indicated at the control console throughout their travel and should indicate when they are at an "in" or "out" limit.

- The control rod drive speed in "manual" and "automatic" modes of operation should be limited to that analyzed and allowed for controlling the rate of change of reactivity.

- While in "automatic" reactor control mode, the RCS should indicate being placed in or removed from automatic control.

- In a reactor designed for pulsing, the movement of the transient rod should be limited in accordance with reactivity amounts and rates derived from the SAR analysis.

- In a reactor designed for pulsing, the system should indicate the position of the transient rod, when this rod is fully inserted, and when set in position to initiate a pulse, and should provide interlocks to ensure the position of the rod.

- Bypasses of interlocks should be under the direct control of the reactor operator and should be indicated in the control room.

- The RCS and the reactor reactivity control system should meet the requirements of minimum shutdown margin considering the stuck rod criteria.

- The applicant should discuss the conclusions about capability and suitability of the RCS requested in Section 7.2.5.

7.4 Reactor Protection System

The RPS is designed to detect the need to place the reactor in a subcritical, safe shutdown condition (scram) when any of the monitored parameters exceeds the limit as determined in the SAR. Upon detecting the need, the RPS should promptly and automatically place the reactor in a subcritical, safe-shutdown condition (scram) and maintain it there. This prevents or mitigates unintended operation in regions where risks of the following types could occur: fuel damage from overpower or loss of cooling events, uncontrolled release of radioactive materials to the unrestricted environment, or overexposure of personnel to radiation. Parameters monitored for this purpose could include core neutron flux,

fuel temperature, core coolant flow and temperature, coolant level, area radiation levels, and air concentration, or release, of radioactive materials.

Non-power reactors can be designed and operated so that postulated accidents pose risks to the facility or the public that are not significant or that are within applicable regulatory limits. If justified by the accident analyses of Chapter 13, the RPS need not be separate and independent of the RCS. The applicant for such reactors may perform an analysis to determine whether certain RPS-monitored parameters or interlocks should be required to be redundant, diverse, or single-failure-proof. Two examples of these parameters are the reactor pool level or area radiation exposure rates. Therefore, the RPS and its subsystems should be designed in accordance with the guidance in Section 7.2, and the SAR should include the following information:

- Discussion of design criteria for the RPS as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.

- Safety and system design bases information as specified in Section 7.2.2. Any supplemental facility-specific design bases not specified in the general system requirements should be included.

- System descriptions consistent with that specified in Section 7.2.3, along with any subsystem description that is facility specific and that may not be identified in the general system requirements.

- Analyses of the operation and performance of the RPS similar to that specified in Section 7.2.4. This should include analysis of any features, aspects, or technical specifications including surveillance tests that may be reactor specific and not identified in the general system requirements. These analyses should be based on postulated credible accidents, transients, and other events that could require RPS intervention, and should include all of the applicable features noted in Section 7.3 for the RCS. The analyses should include quantitative performance of all scrams, runbacks, interlocks, and ESF initiators. The specific design features of the RPS that should be addressed include the following:

  - Independent redundant or diverse reactor power level trips.

  - A log power level channel with a reactor period or rate-of-flux change output with a rate or period channel set to scram in accordance with the analysis (certain reactor designs do not require the period scram design feature because they are designed to accommodate rapid additions of reactivity). The log channel and a

linear flux monitoring channel should accurately sense neutrons even in the presence of intense high gamma radiation.

- Neutron flux (power) monitor channels covering the range from subcritical source multiplication to well beyond the licensed maximum power level.

- A startup channel measuring neutrons at subcritical with a minimum count rate interlock to ensure operation and to prevent control or safety rod withdrawal unless the neutron count rate is at least some predetermined minimum such as 2 counts per second. This interlock may not be needed in reactor designs that use photoneutrons for startup. The applicant should justify not needing the interlock in this case. The detector is capable of detecting neutrons in a high gamma field and can be verified so that subcritical neutron multiplication can be determined and all reactivity changes can be monitored until the startup channel indication is overlapped by the log or linear channel power indication.

- RPS scram time as established in the accident analysis, and any other requirements to ensure operability.

- The scram circuit is designed for the protective action to go to completion once it is initiated. The circuit cannot be reset until all released rods are fully inserted.

- Each scram channel has a separate set of contacts or other bi-stable component to trip the RPS system.

- The manual scram switch is located where the operator has ready access, such as near the rod drive controls.

- Upon receipt of a scram signal, the RPS will annunciate the scram and signify the circuits that are in a tripped state.

- The RPS shall always be capable of shutting down the reactor at least to the shutdown margin defined in the technical specifications.

- Conclusions about capability, operability, and suitability of the RPS requested in Section 7.2.5.

7.5 Engineered Safety Features Actuation Systems

If ESFs are required by the accident analyses in Chapter 13, their actuation systems should be described in this section. The ESF actuation system senses the need for and initiates the operation of ESF systems (1) to prevent or mitigate the consequences of damage to fission product barriers such as fuel, cladding, or fueled experiments caused by overpower or loss-of-cooling events or (2) to gain control of any radioactive material released by accidents.

Each active ESF should be automatically initiated by a subsystem of the ESF actuation system. Examples of such systems include those to actuate an active emergency core cooling system (ECCS), containment or confinement system, containment or confinement air cleanup and filtration system, or any other ESF that is designed to perform a mitigative function. Most non-power reactors do not have an active ECCS because they are designed to rely on passive ECCS or natural coolant circulation to provide sufficient core cooling to prevent loss of fuel integrity. Certain non-power reactors may not be required by the accident analyses to have containment or confinement ESF systems or a containment or confinement air cleanup and filtration ESF system. When such systems are required, their actuation systems should be described in this section, in coordination with the information in Chapter 6, "Engineered Safety Features," of the SAR.

Certain parameters should be monitored to determine the need to initiate the operation of ESFs. These parameters should be determined by the accident analyses, and may include fuel temperature, core coolant flow and temperature, coolant level, area radiation, and radioactivity of airborne materials. ESF actuation systems need not be designed to be redundant or diverse, or to be able to survive a single failure and still perform the safety function unless the accident analysis requires these features.

The applicant should describe the ESF actuation system in sufficient detail to describe the functions required of the ESF and the operation of the system. The SAR should include the following information for each required ESF actuation system:

- Design criteria for the ESF actuation system as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.

- Design bases information for the ESF actuation system as specified in Section 7.2.2 and any additional facility-specific design bases not specified in the general system requirements.

- System description of each ESF actuation system similar to that specified in Section 7.2.3. The description should include:
  - any additional facility-specific system design
  - features of the individual initiation and actuation systems which provide for them to function in concert to prevent or mitigate the consequences of postulated accidents
- System performance analysis:
  - an analysis of the operation and performance of each ESF actuation system similar to that specified in the general system requirements of Section 7.2.4, including analysis of the designs of any facility-specific features or aspects
  - a discussion of an analysis of the operation and performance of the individual systems which allow them to function in concert to prevent or mitigate the consequences of postulated accidents
  - the bases of any technical specifications, including surveillance tests and intervals specific to the design and operation of the subsystem
- Conclusions about capability, operability, and suitability of the ESF actuation systems requested in Section 7.2.5.

7.6 Control Console and Display Instruments

Control console and display instrument systems and equipment include displays for the reactor operator to view such operating information as current values of operating parameters and the status of systems and equipment. The system also enables the operator to control the reactor.

The applicant should describe how the control console and display instruments have been designed to collect and display the operating information in such a manner that it can be readily observed and interpreted by the operator. It should describe how the manual control inputs (pushbuttons, switches, and other equipment) have been grouped, oriented, and located with respect to the relevant display instruments to enable the operator to best observe and interpret the operating information and thereby take prompt and accurate steps to supply control inputs on which the reactor control systems can act. In addition, the combined and integrated functioning of the control console and display system should be described to demonstrate how major equipment is designed to function as an integrated information-handling system to readily aid the operator in

controlling operation of the reactor. The control console design should prevent unauthorized operation of the reactor.

The advancement of digital technology has simplified the ability to gather, analyze, manipulate, and display large amounts of data. A number of licensees have considered adding internally developed operator information display systems and operating aids to their I&C systems. If these systems digitally process control console information and present this information to the reactor operator to inform the operator of the status of the reactor, or if the operator uses such information to make decisions about the operation of the reactor, the systems need to go through the same review, including verification and validation of software as a digital RCS or RPS. It is acceptable to locate these systems in areas where they cannot be viewed by the reactor operator. The applicant should ensure that any interface between the information display system and the control console is isolated.

The SAR system design criteria and basis information should include a system description and a system performance analysis for each instrument system or major equipment connected to or displayed at the control console. The description and analysis should be similar to those specified in Section 7.2 and should address the following:

- the outputs, controls, and operator interfaces
- how the output instruments are placed and how they are related to the reactor and other system controls in the main console and auxiliary control room racks
- drawings or photographs showing the arrangement of the display instruments and console control equipment
- sufficient reactor-specific information for operators to understand functions of both analog and digital systems, including connections and interaction between them, and both redundancy and diversity of such systems
- the conclusion about operability and suitability for human factors as requested in the general system recommendations of Section 7.2

7.7 Radiation Monitoring Systems

Radiation monitoring instrument systems should be designed to perform several important diverse functions in the operation of a non-power reactor. These monitors should indicate radiation intensity and may be used for reactor operations such as to indicate the following: low coolant level, the need to actuate containment or confinement systems, and the need for personnel radiation

protective actions, and to monitor release of radioactive material to the environment. These systems include area radiation monitors, with displays near the instrument location and in the control room. These systems may monitor radioactive effluents in the form of gases, liquids, and airborne particulates and provide continuous air monitoring (CAM) for airborne radioactivity in occupied spaces such as the reactor room. Portable radiation monitors and personal dosimetry systems should also be included to help assess exposure and prevent overexposure of workers and other personnel. The radiation protective instruments and measures should be discussed in detail in Chapter 11, "Radiation Protection Program and Waste Management." The present chapter should concentrate on the I&C aspects of the radiation monitoring systems and should be coordinated with the information in Chapter 11.

The applicant should briefly summarize the radiation-monitoring I&C system for the facility and list the various systems and types of equipment. Since some of the systems may provide input to the RPS or ESF actuation system, radiation monitoring systems should meet the applicable criteria and requirements in Section 7.2 for those systems.

For each radiation monitoring system planned for the facility, the applicant should give the I&C system design-basis information, a system description, a system performance analysis, and a conclusion about the suitability of the system to perform its function as specified in Section 7.2.

7.8 Bibliography

American National Standards Institute/American Nuclear Society, ANSI/ANS 10.4, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," ANS, LaGrange Park, Illinois, 1987.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.15, "Criteria for the Reactor Safety Systems of Research Reactors," ANS, LaGrange Park, Illinois, 1978.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.20, "Criteria for the Control and Safety Systems for Research Reactors" (draft), ANS, LaGrange Park, Illinois.

American Nuclear Society, "Transactions of the American Nuclear Society, Session on Digital Control of Nuclear Reactors," pp. 248–259, Vol. 64, San Francisco, California, November 10–14, 1991.

Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2, "IEEE Standard Criteria for Digital Computers Systems in Safety Systems of Nuclear Power Generating Stations," Piscataway, New Jersey, 1993.

Appendix 7.1

Regulatory Guide 1.152, Revision 1

Criteria for Digital Computers in Safety Systems of Nuclear Power Plants
January 1996

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
January 1996

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.152

(Draft was issued as DG-1039)

CRITERIA FOR DIGITAL COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS

A. INTRODUCTION

Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, among other things, that protection systems be designed for high functional reliability commensurate with the safety functions to be performed. Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of 10 CFR Part 50 requires, among other things, that quality standards be specified and that design control measures be provided for verifying or checking the adequacy of design.

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations for promoting high functional reliability and design quality for the use of digital computers in safety systems of nuclear power plants. The term "computer" is a system that includes computer hardware, software, firmware, and interfaces.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been approved by the Office of Management and Budget, Approval No. 3150-0011.

B. DISCUSSION

Instrumentation and Control (I&C) systems that use digital computers in safety systems make extensive use of advanced technology, i.e., equipment and design practices that are expected to be significantly and functionally different from current designs. These designs include, but are not limited to, the use of microprocessors, digital systems and displays, fiber optics, multiplexing, and different isolation techniques to achieve the needed independence and redundancy.

IEEE Std 7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,"[1] was jointly prepared by the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers (IEEE) and the Nuclear Power Plant Standards Committee of the American Nuclear Society (ANS). The NRC staff has worked with IEEE and ANS in developing IEEE Std 7-4.3.2-1993 to ensure that the guidance provided by the consensus standard is consistent with the Commission's regulations. IEEE Std 7-4.3.2-1993

[1]IEEE publications may be purchased from the IEEE Service Center, 445 Hoes Lane, Piscataway, NJ 08854.

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience.

Written comments may be submitted to the Rules Review and Directives Branch, DFIPS, ADM, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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1. Power Reactors
2. Research and Test Reactors
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4. Environmental and Siting
5. Materials and Plant Protection
6. Products
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8. Occupational Health
9. Antitrust and Financial Review
10. General

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Issued guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161.

has evolved from ANSI/IEEE-ANS-7-4 3.2-1982, "Applications Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." IEEE Std 7-4.3.2-1993 is a significant improvement over its 1982 version. The 1993 version was approved by the IEEE Standards Board on September 15, 1993. This standard identifies guidelines for digital computers (including hardware, software, firmware, and interfaces) to supplement IEEE Std 603-1991, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations."[1] The NRC staff recognizes that development processes for computer systems continue to evolve.

Digital I&C systems share data transmissions, functions, and process equipment to a greater degree than analog systems Although this sharing forms the bases for many of the advantages of digital systems, it also raises a key concern with respect to its vulnerability to a different type of failure. The concern is that a design using shared data bases and process equipment has the potential to propagate a common cause failure of redundant equipment. Another concern is that software programming errors can defeat the redundancy achieved by the hardware architectural structure. Because of these concerns, the NRC staff has placed significant emphasis on defense-in-depth against propagation of common cause failures within and between functions.

The principle of defense-in-depth is to provide several levels or echelons of defense to challenges to plant safety, such that failures in equipment and human errors will not result in an undue threat to public safety. A detailed defense-in-depth study and failure mode and effect analysis or an analysis of abnormal conditions or events should be made to address common cause failures. The Commission's position for providing defense against common cause failures in digital I&C systems for future light-water reactors is given in the Staff Requirements Memorandum of July 21, 1993, on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"[2] (specifically in point 18: II Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems").

Section 5.15, "Reliability," of IEEE Std 7-4.3.2-1993 states, "When qualitative or quantitative reliability goals are required, the proof of meeting the goals shall include software used with the hardware." The staff does not endorse the concept of quantitative reliability goals as a sole means of meeting the Commission's regulations for reliability of the digital computers used in safety systems. The NRC staff's acceptance of the reliability of the computer system is based on deterministic criteria for both the hardware and software rather than on quantitative reliability goals.

Software failures that are not the consequence of hardware failures are caused by design errors and, therefore, do not follow the random failure behavior used for hardware reliability. The NRC staff believes that quantitative reliability determination, using a combination of analysis, testing, and operating experience, provides information regarding the safety importance of the computer system and also provides an added level of confidence in its reliable performance. If quantitative software reliability goals are used, the staff believes that the amount of testing of the safety system instrumentation and control equipment will increase. The staff recognizes that the commercial dedication of "commercially" available digital systems in nuclear applications relies a great deal on quantitative methods because of the operating experience data (such as number of hours of successful operation) accumulated over the years. The staff does not intend to preclude operating experience data from the justification of a successful commercial dedication.

Section 6, "Sense and Command Features—Functional and Design Requirements," of IEEE Std 7-4.3.2-1993 indicates that no requirements beyond IEEE Std 603-1991 are necessary. IEEE Std 603-1991 specifies the need to ensure acceptable response time for the instrumentation and control system in order to accomplish necessary safety functions. Consideration of the sampling rate of plant variables is an important aspect of the design of a digital system when satisfying this criterion.

IEEE Std 7-4.3.2-1993 includes 8 annexes. This standard states that these informative annexes are not part of IEEE Std 7-4.3.2-1993. The NRC staff believes these annexes contain information that may be useful. However, the information in these annexes should not be viewed as the only possible solution or method. Since a consensus has not been reached in the nuclear industry, these annexes are not endorsed by the NRC staff.

C. REGULATORY POSITION

Conformance with the requirements of IEEE Std 7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," with the exception of relying solely on quantitative reliability goals (Section 5.15), is a method acceptable to the NRC staff for satisfying the Commission's regulations with respect to high functional reliability and design quality requirements for computers used as components of a safety system.

[2]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555, telephone (202) 634-3273; fax (202) 634-3343

Section 2 of IEEE Std 7-4.3.2-1993 references several industry codes and standards. If a referenced standard has been separately incorporated into the Commission's regulations, licensees and applicants must comply with the standard as set forth in the regulation. If the referenced standard has been endorsed by the NRC staff in a regulatory guide, the standard constitutes an acceptable method of meeting a regulatory requirement as described in the regulatory guide If a referenced standard has been neither incorporated into the Commission's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard if appropriately justified, consistent with current regulatory practice.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this guide.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used in the evaluation of submittals in connection with applications for construction permits and operating licenses. It will also be used to evaluate submittals from operating reactor licensees that propose system modifications voluntarily initiated by the licensee if there is a clear nexus between the proposed modifications and this guidance.

VALUE/IMPACT STATEMENT

A draft Value/Impact Statement was published with the draft of this guide, Task DG-1039, when it was published for public comment in May 1995. No substantive changes were necessary, but a few editorial changes were made for clarity and consistency. A copy of the revised Value/Impact Statement for Revision 1 of Regulatory Guide 1.152 is available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

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Appendix 7.2

Amendment No. 29
to Facility Operating License
No. R-38
General Atomics

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C 20555

October 4, 1990

Docket No. 50-89

Dr. Keith A. Asmussen, Manager
Licensing, Safety and Nuclear Compliance
General Atomics
10955 John Jay Hopkins Drive
San Diego, California 92121-1194

Dear Dr. Asmussen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. R-38 - General Atomics

The Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. R-38 for the General Atomics TRIGA Mark I research reactor. The amendment consists of changes to the Technical Specifications in response to your submittal dated July 19, 1990.

The amendment approves the installation of a microprocessor based instrumentation and control system. The Technical Specifications are amended to reflect the new system.

A copy of the related Safety Evaluation supporting Amendment No. 29 is enclosed.

Sincerely,

Marvin M. Mendonca

Marvin M. Mendonca, Senior Project Manage
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 29
2. Safety Evaluation

cc w/enclosures:
See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C 20555

GENERAL ATOMICS

DOCKET NO. 50-89

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. R-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Facility Operating License No. R-38 filed by General Atomics (the licensee), dated July 19, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4) and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-38 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 29, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Seymour H. Weiss

Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: October 4, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. R-38

DOCKET NO. 50-89

Replace the following page of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
16	16
	17

8. A summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure, and a brief summary of the results of radiation and contamination surveys performed within the facility.

TABLE I
MINIMUM REACTOR SAFETY SYSTEM SCRAMS

Originating Channel	Set Point	Mode in Which Effective	
		SS	Pulse
1. Power Level (2 independent channels)(1)	275 kW or lower	X	
2. Fuel Element Temperature (2 independent channels)	As specified in Section 7.4 or lower		X
3. Console Scram	Manual	X	X
4. Facility Power Supply	Supply Failure	X	X
5. Magnet Current Key Switch	Manual	X	X
6. Watchdog Timer(2)	Software Failure	X	X

(1) Of the minimum required two independent channels, no more than one channel shall utilize digital processing of power detector signals.

(2) These scrams are only applicable when computers are utilized to perform reactor control functions.

TABLE II
MINIMUM INTERLOCKS

Action Prevented	Mode in Which Effective	
	SS	Pulse
1. Withdrawal of more than one standard rod	X	
2. Withdrawal of any standard rod		X
3. Application of air to transient rod unless its movable cylinder is fully down	X	

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO

FACILITY OPERATING LICENSE NO. R-38

GENERAL ATOMICS

DOCKET NO. 50-89

1.0 INTRODUCTION

General Atomics (GA) has determined that due to the obsolescence and progressive deterioration of their control console, a new reactor instrumentation and control system is needed to maintain reliable operations. In December 1988, GA published their safety analysis of the new reactor instrumentation and control system. In this report GA concluded that the new system was an allowable change under 10 CFR 50.59. 10 CFR 50.59 permits licensees to make changes in the facility as described in the safety analysis report without prior Commission approval unless the proposed change, test, or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. "A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced."

The staff concluded from its review of the GA safety analysis report that NRC review and approval of the replacement computerized control system was required, since (1) the installation of the new reactor instrumentation and control system did present an unreviewed safety question because of the possibility of an accident or malfunction of a different type than any evaluated previously and (2) changes to the Technical Specifications were required.

Pursuant to 10 CFR 50.90, the licensee submitted by letter dated July 19, 1990, a request to amend Appendix A of Facility Operating License No. R-38, "Technical Specifications for the Torrey Pines TRIGA Reactor." The licensee's submittal of July 19, 1990 included the December 1988 safety analysis. The requested amendment would (1) allow installation of the micro-processor based instrument and control system, (2) add the watchdog (software failure) scram to Table 1 of the Technical Specifications, "Minimum Reactor Safety System Scrams", and (3) add a requirement that no more than one of the required two independent power level scram channels in Table 1 be a digital scram channel.

The licensee has installed, in parallel to their existing control console, the new digital microprocessor based instrumentation and control system. The transfer of control from the old to the new system (including scram) was via a

series of gradual steps accompanied by tests which demonstrated the reliability of the new equipment while maintaining the proven performance of the existing control system. Upon completion of all testing (described later in this SER), the new console was used to control (except for the hardwired trip functions) both the safety and nonsafety aspects of operation of the TRIGA reactor and the old analog console was disconnected. The new console replaced the old analog console in the control room. The primary functions of the new system remained the same as the old system: to monitor critical parameters and provide a scram signal when needed, to provide information to the operator and to provide control for the pulse and steady-state modes of operation.

2.0 HARDWARE AND SYSTEMS ASSESSMENT

This portion of the review focused on the areas of potential vulnerability or susceptibility of the new control console which might compromise its ability to present accurate information to the operator and to provide scram signals when required. No assessment was made of the reliability of the nonsafety-related controls. Issues investigated included single failure, environmental qualification, seismic qualification, surge withstand capability (SWC), electromagnetic interference (EMI), failure modes and effects, reliability, error detection, and independence.

The primary review criteria for instrument and control systems for research reactors are presented in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria which apply to current vintage nuclear power plants. However, due to the inherent reactivity insertion safety feature of the TRIGA reactor design and minimal decay heat generation that reduce the probability of fuel damage to a minimum; the staff has concluded that these power plant criteria may serve as guidelines and that strict adherence to the power plant criteria is generally not warranted. The exceptions are noted in the appropriate sections below.

During the review, the licensee described the new system including licensing, engineering, testing and training aspects. The staff also had benefit of material from the U.S. Air Force, the University of Texas at Austin and the console owners group, as well as an independent safety review performed by ORI, Inc. which concluded that the system was acceptable. The system for GA's Mark I reactor is a similar system to that reviewed and approved in the "Issuance of Amendment No. 19 to Facility Operating License No. R-84 - Armed Forces Radiobiology Research Institute" (AFRRI).

Similar to AFRRI, the GA Safety System Scram Circuit consists of two analog nuclear power monitor channels (NP-1000, and NPP-1000) and two fuel temperature channels which are hardwired. Different from AFRRI, the NM-1000 microprocessor based nuclear power channel that monitors reactor power is wired to the scram circuit and provides input to rod block. Also, wired into the scram circuit at GA are contacts for manual scram, facility power supply failure scram, key switch scram, and watchdog (software failure) scram. Further, although not required by Technical Specifications, there are scram features on (1) detector

high voltage failure on any one power channel, (2) loss of ac power to the Instrumentation and Control System due to earthquake switch trip, (3) externally generated conditions, and (4) reactor power reaching 1100 MW during a pulse.

2.1 Environmental and Seismic Qualification

The new control system is installed in the control room and the reactor hall. The staff considers the reactor hall to be a mild environment when compared to power plant requirements and therefore the entire system can be considered to be in a mild environment. The system has been constructed in standard commercial enclosures suitable for a mild environment. The testing and operations, to date, have not revealed any problems related to temperature or humidity. The new system should not be unduly susceptible to temperature or humidity problems and is therefore acceptable to the staff.

Though there have been no requirements promulgated for seismic qualification testing of research reactor control equipment, the staff considered the equipment to determine general ruggedness. The licensee indicated that the equipment is mounted in a commercial quality fashion which should prevent any significant movement of components within the console and racks. In this TRIGA reactor, an inadvertent scram does not present a significant challenge to reactor safety systems because a scram consists of the removal of current to the control rod magnets allowing the control rods to drop into the core by gravity; and no other equipment is required to maintain the reactor in a safe shutdown condition. The primary concern remaining would be relay contact chatter which could prevent a scram when required. The safety system scram circuits for this system are designed to scram on failure (which includes contact chatter) and therefore the staff concludes that any further testing is not warranted and the system is acceptable.

2.2 Electromagnetic Interference (EMI)

The staff reviewed the susceptibility of the new equipment to EMI due to the potential for common mode interference which could disable more than one system at a time. As discussed earlier, due to the design characteristics of the TRIGA reactor, an inadvertent scram does not present a significant challenge to safety systems, though it might cause operational difficulties such as disrupting an experiment.

Industrial-type isolators are generally used which prevent conducted EMI from being transmitted between the control and safety mechanisms. The neutron flux signal cabling is shielded to reduce the impact of radiated EMI. Previous experience with similar equipment provided by several different vendors at other facilities has indicated that if EMI causes any perturbance in the system it will most likely cause a scram, which as previously discussed is not a safety concern. Based on the above, the staff concludes that EMI should not prevent a scram when required and the design is therefore acceptable.

2.3 Power Supplies

The power supplies for the system are buffered to reduce the possible impact of minor power line fluctuations. The scram circuits for the new system are

designed to scram when power is lost to them. The NP-1000 and NPP-1000 are analog devices and will respond to power fluctuations similar to the existing analog equipment. The digital NM-1000 nuclear power channel uses a battery backed-up random access memory (RAM) to store constant data during loss of power. In addition to self-diagnostics, the NM-1000 has a watchdog timer circuit which puts the NM-1000 in a tripped condition and scrams the reactor if power fluctuations prevent proper software operation. As described in the NM-1000 Software Functional Specification and Software Verification Program (March 1989), the NM-1000 is also tested to verify that the system returns to proper operation following restoration of power. The staff finds this acceptable.

2.4 Failure Modes and Effects

The December 1988 safety analysis included Scram Circuit Safety Analysis performed by the University of Texas at Austin. This study identified the various ways in which the reactor safety system could fail. These include:

1) Physical System Failure (wire breaks, shorts, ground fault circuits)
2) Limiting Safety System Setting Failure (failure to detect)
3) System Operable Failure (loss of monitoring)
4) Computer/Manual Control Failure (automatic and manual scram).

This study was based on a fault tree approach which predicted failure to scram for various failure modes. The study concluded that a failure of all safety systems and therefore failure to scram was extremely unlikely. Failures attributable to the unique failure modes of the software of the NM-1000 were considered. The staff concludes that the failure modes and effects of the new system were acceptably addressed.

2.5 Independence, Redundancy and Diversity

The staff reviewed the data link between the safety channels and the nonsafety systems. The safety channels provide direct hard-wired scram inputs and are also hardwired directly to independent indicators on the control console. The operators are provided with information from both the analog NP-1000 and NPP-1000 power monitors and the digital NM-1000 monitor. The information is displayed on both direct wired bar graphs and on a graphic CRT. In addition, the safety channels provide inputs to the Non-Class 1E Data Acquisition Computer (DAC) through isolators. The isolators used have not been tested for maximum credible faults which the staff requires for power plant use, but have been tested by the manufacturer to standard commercial criteria. The DAC is then connected via redundant high speed serial data trunks to the Non-Class 1E Control System Computer (CSC) which interfaces with the operator by controls, a keyboard and CRT displays. Since the CSC does communicate with the safety channels, this aspect of the system would not meet the independence requirements of a power plant, but the staff concluded it was not necessary for the current application at GA.

Further, the scram circuit is essentially unchanged in that it maintains the fail safe design using the same automatic and manual contacts which open to remove power to the control rod magnets. For the GA facility, redundant fuel temperature inputs are provided to the scram circuit. Redundant power level inputs (NP-1000, NPP-1000) to the scram circuit are also provided.

This system has also added the computer watchdog scram and the digital NM-1000 scram. At GA, in addition to the NM-1000 being wired to the scram circuit, it provides inputs to the rod withdrawal prevent interlock system. The use of both analog and digital neutron monitoring, and the watchdog scram function provides additional diversity and redundancy to the scram system. The system as installed meets most of the requirements of IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE 379-1977 "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems."

The staff has concluded that the level of independence, redundancy and diversity which has been maintained is acceptable for the GA TRIGA reactor.

2.6 Testing and Operating History

Extensive testing of the new system has been done by both GA and AFFRI. A significant number of design changes took place during the testing and phase-in of the new system. The staff has reviewed the problems discovered during testing of the system and has concluded that the resolutions appear acceptable. The staff also agrees with the licensee that long-term operability and safety is enhanced due to installation of equipment which has spare parts readily available. An additional improvement is the self diagnostics feature which allows continuous on-line testing and reduces the possibility of undetected failures.

3.0 SOFTWARE ASSESSMENT

3.1 Criteria

The staff requires an approved verification and validation (V&V) plan for software which performs a safety function or provides information to the operator. At GA, the NM-1000 provides inputs to the scram circuit and the rod withdrawal prevent interlock system block function. The NM-1000 software development was reviewed by the staff to determine the acceptability of the V&V plan. The staff compared the General Atomics V&V plan to Regulatory Guide 1.152 "Criteria for Programmable Digital Computer Software in Safety-Related Systems at Nuclear Power Plants" which endorses ANSI.IEEE 7-4.3.2 - 1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." The staff has concluded that this standard is appropriate for use in reviewing research reactor software.

3.2 Verification and Validation Plan

The staff audited the verification and validation documentation provided by General Atomics. For the installation at the GA TRIGA the NM-1000 is wired directly into the scram circuit, and therefore requires highly reliable software to perform its safety function when required. The assessment of the

NM-1000 software built by General Atomics is an assessment of the methodology and procedures used to develop the software. The process is evaluated by reviewing the verification and validation trail through the development process.

Verification and validation (V&V) are two separate but related activities that follow the development of software. Verification determines whether the requirements of one phase of the development cycle have been consistently, correctly, and completely transformed (fulfill the requirements) to the subsequent phase of the cycle. Validation is the testing of the final product to ensure that performance conforms to the requirements of the initial specification. The need for V&V arose because software is very complex, and prone to human errors of omission, commission and interpretation. V&V provides for an independent verifier to work in parallel with, but independent of, the development team to ensure that human errors do not hinder the production of safety software that is reliable and testable.

In executing V&V, certain principles have proven over time to be very effective in software programs:

- Well defined systems requirements expressed in a well written document.
- Development methodology to guide the production of software.
- Comprehensive testing procedures.
- Independence of the V&V team from the development organization.

These principles can serve as a comprehensive reference base for applying the applicable criteria for software evaluations of Class 1E safety systems, and were used as guidance in the following review areas.

3.3 Independence

A key ingredient in an effective verification process is the independence of the verifier. For the NM-1000 the original software was developed by Sorrento Electronics. After General Atomics obtained the rights to market the NM-1000 for research reactors, a software consultant was used to modify the software. After many changes had been made another contractor was brought in. Each contractor in turn assured an additional level of independent review from the original design. Though the requirements imply a concurrent review the staff finds that the verification has been sufficiently independent and is therefore acceptable for research reactors.

3.4 Validation Testing

The validation testing must be done by a team that did not participate in the design or implementation of the software product. General Atomics used the Neutron Monitoring System Acceptance Test Procedure as part of the validation testing. In addition the staff reviewed substantial additional validation testing which has been performed at the AFRRI facility. The staff did note a functional description of unknown date which included samples of the computer code. Though the people involved in development knew the specific functions

which the NM-1000 was to perform these had never been written down, which allows substantial possibilities for omission when preparing test procedures. Upon request from the staff, General Atomics provided the functional specification E117-1001 (March 1989) which lists in detail the functions performed by the NM-1000. Included in this specification was a cross reference where the vendor verified that each specific functional requirement had been tested. The staff finds that this testing and verification is acceptable.

3.5 Discrepancy Resolution

A key element in any verification and validation effort is the process by which discrepancies uncovered during development are recorded, identified, resolved and corrected. The resolution of a discrepancy must be reflected in all applicable documents (e.g., source code, the software design specification, the software requirements, and the original systems specification). The staff reviewed discrepancies and other comments provided to General Atomics by the Console Owners Group and found that the process and resolution were documented and appeared adequate. When discrepancies resulted in code modification, a description of the changes and it's reason was added to the code annotation. The staff finds the discrepancy resolution methods by General Atomics to be acceptable.

3.6 Design Approach

The primary, software specification provides the foundation for not only sound development, but also for effective verification and validation activities. The individual requirements in the specification for any software system describe how the software is to behave in any circumstance. The specification must be reliable and testable. A reliable specification exhibits the following characteristics:

- Correct - Each requirement of the safety function has been stated correctly.
- Complete - All of the requirements for the safety function are included.
- Consistent - The requirements are complementary and do not contradict each other.
- Feasible - The requirements can be satisfied with available technology.
- Maintainability - The requirements will be satisfied for the lifetime of the equipment.
- Accuracy - The requirements include the acceptable bounds of operation.

The staff reviewed the design approach with General Atomics. The documentation was found to be lacking in several areas with the most significant being the lack of a functional requirements specification which GA has since prepared. Documentation of the early development was sketchy which was attributed partially to the transfer of the product without including all of the backup information. The documentation of recent changes has improved significantly. Though the staff finds that the design approach for the NM-1000 since inception has been erratic, the recent, development work appears to be improved in structure and control.

3.7 Software Evaluation

The software development plan for the NM-1000 appears to the staff to be a very specific design goal oriented development, where the application and basic hardware and software requirements were known by the designers; however, there was no step by step plan developed. The failure to have a step by step plan such as described in ANSI/IEEE 7-4.3.2 - 1982 resulted in the need for General Atomics to retrofit the functional requirements document and verify that each requirement had been tested. To meet this requirement GA developed the NM-1000 software verification program (E117-1002 March 1989). The staff also reviewed working copies of the NM-1000 design input which demonstrates that the functional requirements are currently well understood by the design team and concludes that the software should perform its intended safety function as required.

The staffs review indicated that GA could benefit through the development of a corporate software development plan that can be applied to any future Class 1E software prior to starting design. The plan could include a description of the development phases in sufficient detail so that the verification and validation efforts can be initiated at the beginning of any design effort. The plan could also contain a taxonomy of documentation, and reviews which demark the injection points for verification and validation activities. A corporate software development plan for Class 1E systems could prove to be effective in the development of reliable software consistent with the intent of ANSI/IEEE-ANS-7-4.3.2 - 1982.

3.8 Operator Task Analysis

In reviewing the documents it became apparent that there was not a formal task analysis to support the design of the operator interface. The initial specifications and descriptions were vague. After the equipment and software were substantially designed, the functional requirements and working level descriptions did include the operator task requirements. A task analysis prior to development would probably have minimized the software iterative process and therefore provided less opportunities for error. The staff concluded that through the V&V process the requirements have been specified, and incorporated in the design. Therefore, the V&V plan is acceptable.

4.0 TECHNICAL SPECIFICATIONS

As previously discussed, the presentation of correct, timely information to the reactor operator contributes to the safe operation of the reactor. The scram circuit at GA will include watchdog timer contacts which will provide a scram upon software failure. Therefore to assure the presentation of timely, correct information to operators or the proper safety system scram, the watchdog scram inputs are added to Table I, Minimum Reactor Safety System Scrams of the Technical Specifications. Additionally to assure acceptable diversity of the new system, Table I has been amended to specify that of the minimum required two independent power level channels, no more than one channel shall use digital processing of power detector signals.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff concludes that the hardware design of the new General Atomics console is acceptable for use in the GA TRIGA reactor. The Software design in the CSC, DAC and NM-1000 will not prevent the safety functions of the hardwired scram circuit from performing and is therefore acceptable. The technical specifications are amended to include the watchdog scram inputs and maximum use of digital power measurement channels.

The staff has also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: James C. Stewart

Dated: October 4, 1990

Appendix 7.3

Amendment No. 30
to Facility Operating License
No. R-2
Penn State Breazeale Reactor

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C. 20555

Docket No. 50-5

August 6, 1991

Dr. Charles L. Hosler
Vice President for Research and
Dean of the Graduate School
Pennsylvania State University
207 Old Main Building
University Park, Pennsylvania 16802

Dear Dr. Hosler:

SUBJECT: ISSUANCE OF AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. R-2 - PENN STATE BREAZEALE REACTOR

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. R-2 for the Penn State Breazeale Reactor. The amendment consists of changes to the Technical Specifications in response to your submittal of April 19, 1991 as supplemented on July 8, 1991.

The amendment approves the installation of a new reactor instrumentation and control console system. The Technical Specifications are amended to reflect this change.

A copy of the related Safety Evaluation supporting Amendment No. 30 is enclosed.

Sincerely,

Marvin M. Mendonca, Senior Project Manager
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 30
2. Safety Evaluation

cc w/enclosures:
See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C 20555

PENNSYLVANIA STATE UNIVERSITY

DOCKET NO. 50-5

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. R-2

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment filed by the Pennsylvania State University (the licensee), dated April 19, 1991 as supplemented on July 8, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4) and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-2 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 30, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of August 11, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard F. Dudley, Jr. for

Seymour H. Weiss, Director
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors and
Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance:

ENCLOSURE TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. R-2

DOCKET NO. 50-5

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change.

Remove	Insert
7	7
14	14
15	15
16	16
17	17
31	31
32	32

reactive rod is in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.1.42 SQUARE WAVE OPERATION

Square wave (SW) operation shall mean operation of the reactor with the mode selector switch in the square wave position which allows the operator to insert preselected reactivity by the ejection of the transient rod, and which results in a maximum power of 1 MW or less.

1.1.43 TRIGA FUEL ELEMENT

A TRIGA fuel element is a single TRIGA fuel rod of standard type, either 8.5 wt% U-ZrH in stainless steel cladding or 12 wt% U-ZrH in stainless steel cladding enriched to less than 20% uranium-235.

1.1.44 WATCHDOG CIRCUIT

A watchdog circuit is a circuit consisting of a timer and a relay. The timer energizes the relay as long as it is reset prior to the expiration of the timing interval. If it is not reset within the timing interval, the relay will de-energize thereby causing a SCRAM.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT-FUEL ELEMENT TEMPERATURE

Applicability

The safety limit specification applies to the maximum temperature in the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element and/or cladding will result.

Specifications

The temperature in a water-cooled TRIGA fuel element shall not exceed 1150°C under any operating condition.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured at a point within the fuel element. The measured fuel temperature is directly related to the maximum fuel temperature of the region. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the maximum fuel temperature exceeds 1150°C. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature, the ratio of hydrogen to zirconium in the alloy, and the rate change in the pressure.

3.2 CONTROL AND SAFETY SYSTEM

3.2.1 REACTOR CONTROL RODS

Applicability

This specification applies to the reactor control rods.

Objective

The objective is to assure that sufficient control rods are operable to maintain the reactor subcritical.

Specification

There shall be a minimum of three operable control rods in the reactor core.

Basis

The shutdown margin and excess reactivity specifications require that the reactor can be made subcritical with the most reactive control rod withdrawn. This specification helps assure it.

3.2.2 MANUAL AND AUTOMATIC CONTROL

Applicability

This specification applies to the maximum reactivity insertion rate associated with movement of a standard control rod out of the core.

Objective

The objective is to assure that adequate control of the reactor can be maintained during manual and 1, 2, or 3 rod automatic control.

Specification

The rate of reactivity insertion associated with movement of either the regulating, shim, or safety control rod shall be not greater than 0.63% $\Delta k/k$ (~90¢) per second when averaged over full rod travel. If the automatic control uses a combination of more than one rod, the sum of the reactivity of those rods shall be not greater than 0.63% $\Delta k/k$ (~90¢) per second when averaged over full travel.

Basis

The ramp accident analysis (refer to Safety Evaluation, Chapter IX) indicates that the safety limit will not be exceeded if the reactivity addition rate is less than $2.50/second, when averaged over full travel. This specification of $0.90/second, when averaged over full travel, is well within that analysis.

3.2.3 REACTOR CONTROL SYSTEM

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels listed in Table 1 are operable. (Note that MN, AU, and SW are abbreviations for manual, automatic and square wave, respectively).

Table 1

Measuring Channels

Measuring Channel	Min. No Operable	Effective Mode MN AU	Pulse	SW
Fuel Element Temperature	1	X	X	X
Linear Power	1	X		X
Percent Power	1	X		X
Pulse Peak Power	1		X	
Count Rate	1	X		
Log Power	1	X		X
Reactor Period	1	X		

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for the manual, automatic, square wave and pulsing modes of operation. The specifications on reactor power level and reactor period indications are included in this section to provide assurance that the reactor is operated at all times within the limits allowed by these Technical Specifications.

3.2.4 REACTOR SAFETY SYSTEM AND INTERLOCKS

Applicability

This specification applies to the reactor safety system channels, the interlocks, and the watchdog circuit.

Objective

The objective is to specify the minimum number of reactor safety system channels and interlocks that must be operable for safe operation.

Specification

The reactor shall not be operated unless all of the channels and interlocks described in Table 2a and Table 2b are operable.

Table 2a

Minimum PSBR Channels

Channel	Number Operable	Function	Effective Mode MN AU	Pulse	SW
Fuel Temperature	1	SCRAM ≤ 700°C	X	X	X
High Power	2	SCRAM ≤ 110% of 1 MW	X		X
Detector Power Supply	1	SCRAM on failure of supply voltage	X		X
Scram Bar on Console	1	Manual Scram	X	X	X
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X	
Watchdog Circuit	1	SCRAM on software or self-check failure	X	X	X

Table 2b

Minimum PSBR Interlocks

Interlocks	Number Operable	Function	Effective Mode MN	Pulse	SW
Source Level	1	Prevent rod withdrawal with less than two neutron induced counts per second on the startup channel	X		
Log Power	1	Prevent pulsing from levels above 1 kW		X	
Transient Rod	1	Prevent applications of air unless cylinder is fully inserted	X		
Shim, Safety, and Regulating Rod	1	Movement of any rod except transient rod		X	
Simultaneous Rod Withdrawal	1	Prevents simultaneous manual withdrawal of two rods	X		X

Basis

A temperature scram and two power level scrams provide automatic protection to assure that the reactor is shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system in any mode of operation if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety chambers, operation of the reactor without adequate instrumentation is prevented. The preset timer insures that the transient rod will be inserted and the reactor will remain at low power after pulsing. The watchdog circuit will scram the reactor if the software or the self-checks fail (see Safety Analysis Report,Chapter VII, sections H.2.d and I.4)

In the pulse mode, movement of any rod except the transient rod is prevented by an interlock. This interlock action prevents the addition of reactivity over that in the transient rod. The interlock to prevent startup of the reactor with less than 2 cps assures that sufficient neutrons are available for proper startup in all relevant modes of operation. The interlock to prevent the initiation of a pulse above 1 kW is to assure that the magnitude of the pulse will not cause the safety limit to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the manual mode. Simultaneous manual withdrawal of two rods is prevented to assure the reactivity rate of insertion is not exceeded.

insertion rates, and the reactivity worth of experiments inserted in the core.

4.2.2 REACTIVITY INSERTION RATE

Applicability

This specification applies to control rod movement speed.

Objective

The objective is to assure that the reactivity addition rate specification is not violated and that the control rod drives are functioning.

Specification

The rod drive speed both up and down and the time from scram initiation to the full insertion of any control rod from the full up position shall be measured annually, not to exceed 15 months, or when any significant work is done on the rod drive or the rod.

Basis

This specification assures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor. It also assures that the maximum reactivity addition rate specification will not be exceeded.

4.2.3 REACTOR SAFETY AND CONTROL SYSTEMS

Applicability

The specifications apply to the surveillance requirements for measurements, channel tests, and channel checks of the reactor safety systems and watchdog circuit.

Objective

The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specifications

a. A channel test of the scram function of the high power, fuel temperature, manual, and present timer safety channels shall be made on each day that the reactor is to be operated, or prior to each operation that extends more than one day.

b. A channel test of the detector power supply SCRAM function and the watchdog circuit shall be performed annually, not to exceed 15 months.

c. Channel checks for operability shall be performed daily on fuel element temperature, linear power, count rate, log power and reactor period when the reactor is to be operated, or prior to each operation that extends more than one day.

d. The percent power channel shall be compared with other independent channels for proper channel indication, when appropriate, each time the reactor is operated.

e. The pulse peak power channel shall be compared to the fuel temperature each time the reactor is pulsed, to assure proper peak power channel operation.

Basis

TRIGA system components have proven operational reliability. Daily channel tests insure accurate scram functions and insure the detection of possible channel drift or other possible deterioration of operating characteristics. The channel checks will make information available to the operator to assure safe operation on a daily basis or prior to an extended run. An annual channel test of the detector power supply scram will assure that this system works, based on past experience as recorded in the operation log book. An annual channel test of the watchdog circuit is sufficient to assure operability. Comparison of the percent power channel with other independent power channels will assure the detection of channel drift or other possible deterioration of its operational characteristics. Comparison of the peak pulse power to the fuel temperature for each pulse will assure the detection of possible channel drift or deterioration of its operational characteristics.

4.2.4 REACTOR INTERLOCKS

Applicability

This specification applies to the surveillance requirements for the reactor control system interlocks.

Objective

The objective is to insure performance and operability of the reactor control system interlocks.

Specifications

a. A channel check of the source interlock shall be performed each day that the reactor is operated or prior to each operation that extends more than one day.

b. A channel test shall be performed semi-annually, not to exceed 7 months, on the log power interlock which prevents pulsing from power levels higher than one kilowatt.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 30 TO

FACILITY OPERATING LICENSE NO. R-2

PENNSYLVANIA STATE UNIVERSITY

DOCKET NO. 50-5

1.0 INTRODUCTION

The Pennsylvania State University (the licensee) has determined that a new reactor instrumentation and control console system should provide improved reliability, easier repair, and increased information to the operators. Therefore, the licensee's submittal of April 19, 1991 as supplemented on July 8, 1991, requested changes to allow installation of the new reactor instrumentation and control console system.

2.0 DISCUSSION

The transfer of control to the new system (including scram functions) is to be via a series of gradual steps accompanied by tests which demonstrate the reliability of the new equipment while maintaining the proven performance of the existing control system. Upon completion of all testing (discussed later in this safety evaluation), the new console will be used to control (except for the hardwired trip functions) both the safety and non-safety aspects of operation of the TRIGA reactor. The new console will replace the old console in the control room. The primary functions of the new system remain the same as the old system: to monitor critical parameters and provide a scram signal when needed; to provide information to the operator; and to provide control for operations.

2.1 Hardware and Systems Assessment

This portion of the review focused on the areas of potential vulnerability or susceptibility of the new control console which might compromise its ability to provide accurate information to the operator and to provide scram signals when required. No assessment was made of the reliability of the non-safety-related controls. Issues investigated included single failure, environmental qualification, seismic qualification, surge withstand capability (SWC), electromagnetic interference (EMI), failure modes and effects, and independence.

The primary review criteria for instrumentation and control systems for research reactors are included in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria which apply to current vintage nuclear power plants. However, due to the inherent reactivity insertion safety feature of the TRIGA reactor design and minimal decay heat generation that reduce the

probability of fuel damage to a minimum, the staff has concluded that these power plant criteria may serve as guidelines and that strict adherence to the power plant criteria is generally not warranted. The exceptions are noted in the appropriate sections below.

2.1.1 Environmental and Seismic Qualification

The new control system will be installed in the control room and the reactor room. The staff considers the reactor room to be a mild environment when compared to power plant requirements and therefore the entire system can be considered to be in a mild environment. The system has been constructed in standard commercial enclosures suitable for a mild environment. The testing and operations, to date, have not revealed any problems related to temperature or humidity. The new system should not be unduly susceptible to temperature or humidity problems and is therefore acceptable to the staff.

Though there have been no requirements promulgated for seismic qualification testing of research reactor control equipment, the staff considered the equipment to determine general ruggedness. The licensee indicated that the equipment is mounted in a commercial quality fashion which should prevent any unacceptable movement of components within the console and racks. The most likely outcome in any seismic event for the new control system would be a reactor scram. In this TRIGA reactor, an inadvertent scram does not present a significant challenge to reactor safety systems because a scram consists of the removal of current to the control rod magnets allowing the control rods to drop into the core by gravity. No other equipment is required to maintain the reactor in a safe shutdown condition. The primary concern remaining would be relay contact chatter which could prevent a scram when required. The safety system scram circuits for this system are designed to scram on failure (which includes contact chatter) and therefore the staff concludes that any further testing is not warranted and the system is acceptable.

2.1.2 Electromagnetic Interference (EMI)

The staff reviewed the susceptibility of the new equipment to EMI for potential common mode interference which could disable more than one system at a time. Industrial-type isolators and input/output cards are generally used which prevent conducted EMI from being transmitted between the control and safety mechanisms. The console's signal cabling is shielded and where possible separated from the power cabling to reduce the impact of EMI. Further, the power supply is filtered for high frequency EMI. Based on the above, the staff concludes that EMI should not prevent a scram when required and the design is therefore acceptable. Previous experience with similar equipment provided by several different vendors at other facilities has indicated that if EMI causes any disturbance in the system it will most likely cause an inadvertant scram, which as previously discussed is not a safety concern.

2.1.3 Power Supplies

The power supplies for the system are filtered and protected with surge withstand capabilities to reduce the possible impact of minor power line fluctuations. The scram circuits for the new system continue to scram when power is lost to them. The control rod motor controllers use a battery back-up random access memory (RAM) to store constant data during loss of power. In addition to self-diagnostics, the computerized control system has watchdog timer circuits which scram the reactor if anything, including power fluctuations, prevent proper software operation. The control system is also tested to verify that the system returns to proper operation following restoration of power. The staff finds this acceptable.

2.1.4 Failure Modes and Effects

The licensee's safety analysis referenced "Failure Modes and Effects Analysis," AECL Document FMA-17-60501-001, Rev. 0. This study identified the various ways in which the reactor safety system could fail. The study concluded that: "no single failure will prevent a reactor scram or desirable interlock. Although certain failures may impair a particular trip, there is always an alternative trip parameter.... available to initiate a SCRAM." Further, the licensee indicated in their safety analysis that in a postulated event of the complete failure of the reactor safety system coincident with the occurrence of any analyzed accident, the radiological consequences would be negligible. The staff concludes that the failure modes and effects of the new system were acceptably addressed and even in the event of a unforeseen failure, the consequence would not be significant from a public health and safety viewpoint.

2.1.5 Independence, Redundancy and Diversity

The staff reviewed the data link between the safety channels and the non-safety systems. The safety channels provide direct hard-wired scram inputs and are also hardwired directly to independent indicators on the control console. The operators are provided with information from both the wide range and power range neutron monitors. The information is displayed on direct analog indicators and can be displayed on a graphic CRTs. In addition, the safety channels provide inputs to the computerized Protection, Control and Monitoring System (PCMS) through isolators. The isolators are of a standard commercial design. The PCMS provides connections with the operator by controls, a keyboard and CRT displays. Since the PCMS does communicate with the safety channels, this aspect of the system would not meet the independence requirements of a power plant, but the staff concluded it was not necessary for the current application.

Further, the scram circuit is essentially unchanged in that it maintains the fail safe design using the same automatic and manual contacts which open to remove power to the control rod magnets. Redundant power level inputs to the scram circuit continue to be provided. Also, additional external scram buttons are provided for experimenters.

This system has also added the PCMS SCRAM request and PCMS watchdog scrams. In addition to providing numerous, non-required scrams, the PCMS provides inputs to the rod withdrawal interlocks and the reactor protection mode interlocks. The use of two independent power monitoring instruments and an independent fuel temperature instrument, in addition to the watchdog scram function provides additional diversity and redundancy to the scram system. The system as installed meets most of the requirements of IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE-379-1977 "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems."

The staff has concluded that the level of independence, redundancy and diversity which has been maintained is acceptable.

2.1.6 Testing and Operating History

Extensive testing of the new system has been done at the licensee's facility. Additionally, similar control systems have been used in various capacitites, e.g., feedwater control, at nuclear power plants for some time. The staff also agrees with the licensee that long-term operability and safety is enhanced due to installation of the new console system, which should provide improved reliability, easier repair, and increased information to the operators. An additional improvement is the self diagnostics feature which allows continuous on-line testing and reduces the possibility of undetected failures.

2.2 Verification and Validation Plan

The staff requires an approved verification and validation (V&V) plan for software which performs a safety function or provides information to the operators. The software development was reviewed by the staff to determine the acceptability of the V&V plan. The staff compared the V&V plan to Regulatory Guide 1.152 "Criteria for Programmable Digital Computer System in Safety Systems of Nuclear Power Generating Stations."

2.2.1 Verification of Software Design

Extensive independent verification by individuals qualified to develop such software for the instrumentation and control console system provided acceptable assurance that the software would function as designed. Extensive verification testing of the software identified and resolved discrepancies and operating problems. This verification process included consideration for breaking down the software functions into individual discreet packages. The individual functions of these packages were verified, in addition to the validation of the integrated function of these packages. The staff has concluded that based on this review, and previous experience with similar software and with the specific vendor for this console system, the verification of the new system software is acceptable.

2.2.2 Validation Testing

The vendor's and licensee's initial testing program was designed to show that the new system is capable of performing assigned tasks. This integrated testing of the new console identified and resolved design problems, and provided added assurance of proper function of the software. Additionally, the system has monitored reactor parameters in parallel with the existing system, and additional problems were resolved to assure acceptable system function. Finally, a comprehensive process has been devised to provide testing and assurance that the new system will operate properly upon installation.

The staff's review finds this V&V testing program acceptable.

2.3 Maintenance and Surveillance

An extensive program is in place to evaluate the operation of the components of the control system. In addition to the initial testing program, a self-diagnostics feature, that provides continuous on-line testing, and routine surveillance tasks provide additional assurance of continued operation and rapid discovery of problem areas. The routine surveillance tasks include regularly-scheduled surveillance and test procedures which include those items required by the Technical Specifications as well as a number of other items. Major daily checks of all scram systems, and channel tests of a number of information systems are used to document the operability of the systems. Other tasks performed at monthly, semi-annual, and annual intervals include calibrations of important systems and functions (including the control rods, the power levels, the fuel temperature and the radiation-sensing instruments) as well as a number of non-safety-related parameters. These procedures, which have been in place for a number of years, assured proper maintenance of the present control system, and can be expected to provide continued assurance of proper operation and detection of problem areas with the proposed system. Further, the licensee has established a maintenance plan using the manufacturer's equipment manual and off-line testing package. Based on this experience, as well as the self-diagnostic features of the new console system, the staff concluded that the maintenance and testing program is acceptable for the new console.

2.4 Training of Operators

Reactor operations personnel have participated in training sessions and will participate in formal operational testing to complete training. Operations personnel also manipulated the controls and performed several operations on the new console system before installation. The staff concluded that this training was acceptable to assure operator understanding of the new instrumentation and control system.

2.5 Technical Specifications

Although the licensee provided arguments against the need for software watchdog scram Technical Specification requirements, the licensee and the staff recognize the desirablility of the watchdog scram. Therefore, the licensee

provided changes to implement the watchdog scram (Technical Specification pages 7, 16, 17, 31 and 32). The staff has concluded that these changes are needed to acceptably assure that potential undetected software problems would not result in a failure to scram or in inaccurate information to the operator. This requirement is consistent with the intent of previous amendments that allowed installation of computerized control console systems on TRIGA reactors, including General Atomic and the Armed Forces Radiobiological Research Institute. Therefore, the proposed changes are acceptable.

The licensee has also proposed changes to page 14 to incorporated reactivity addition limits applicable to the new consoles operational modes (Page 15 is also changed to account for the retyping changes made to page 14). Based on the staff's review of the licensee's analysis as described in the bases for this Technical Specification change, the staff finds these changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: Marvin M. Mendonca

Dated: August 6, 1991

Appendix 7.4

NRC Generic Letter 95-02

Use of NUMARC/EPRI Report TR-102348, "Guideline on Licensing Digital Upgrades," in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

April 26, 1995

NRC GENERIC LETTER 95-02: USE OF NUMARC/EPRI REPORT TR-102348, "GUIDELINE ON LICENSING DIGITAL UPGRADES," IN DETERMINING THE ACCEPTABILITY OF PERFORMING ANALOG-TO-DIGITAL REPLACEMENTS UNDER 10 CFR 50.59

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) staff is issuing this generic letter to inform addressees of a new staff position on the use of Nuclear Management and Resources Council/Electrical Power Research Institute (NUMARC/EPRI) Report TR-102348, "Guideline on Licensing Digital Upgrades," dated December 1993, as acceptable guidance for determining when an analog-to-digital replacement can be performed without prior NRC staff approval under the requirements of Section 50.59 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.59). The report applies to all digital equipment that uses software and, in particular, to microprocessor-based systems. The report, together with the clarifications discussed in this generic letter, represents a method acceptable to the staff for use in making a determination of whether or not an unreviewed safety question exists with respect to 10 CFR 50.59 requirements. It is expected that recipients will consider the information in this generic letter when performing analog-to-digital instrumentation and control systems replacements. However, suggestions contained in this generic letter are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

The age-related degradation of some earlier analog electronic systems and the difficulties in obtaining qualified replacement components for those systems, as well as a desire for enhanced features such as automatic self-test and diagnostics, greater flexibility, and increased data availability have prompted some operating reactor licensees to replace existing analog systems with digital systems. After reviewing a number of these digital system replacements and digital equipment failures in both nuclear and non-nuclear applications, the staff has identified potentially safety-significant concerns pertaining to digital systems in nuclear power plants. The concerns of the staff stem from the design characteristics specific to the new digital electronics that could result in failure modes and system malfunctions that either were not considered during the initial plant design or may not have been evaluated in sufficient detail in the safety analysis report. These

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concerns include potential common mode failures due to (1) the use of common software in redundant channels, (2) increased sensitivity to the effects of electromagnetic interference, (3) the improper use and control of equipment used to control and modify software and hardware configurations, (4) the effect that some digital designs have on diverse trip functions, (5) improper system integration, and (6) inappropriate commercial dedication of digital electronics.

As a result of the above concerns, the NRC staff issued a draft generic letter for public comment in the *Federal Register* (57FR36680) on August 14, 1992, wherein a position was established that essentially all safety-related digital replacements result in an unreviewed safety question because of the possibility of the creation of a different type of malfunction than those evaluated previously in the safety analysis report. The staff concluded, therefore, that prior approval by the NRC staff of all safety-related digital modifications was necessary. However, subsequent discussions and comments on the draft generic letter have resulted in the staff position as described in this letter.

Discussion

To assist licensees in effectively implementing digital replacements by addressing the concerns indicated above and in determining which upgrades can be performed under 10 CFR 50.59 without prior NRC staff approval, Report TR-102348 has been published. The NRC staff reviewed and provided comments on this report while it was in draft form, and the final report reflects a coordinated effort between industry and the NRC staff. The NRC staff believes that, when properly implemented, modern digital systems offer the potential for greater system reliability and enhanced features such as automatic self-test and diagnostics, as well as greater flexibility, increased data availability, and ease of modification.

Report TR-102348 contains guidance that will assist licensees in implementing and licensing digital upgrades in such a manner as to minimize the potential concerns indicated above. It describes actions to be taken in the design and implementation process to ensure that the digital upgrade licensing and safety issues are addressed, and ways to consider these issues when performing the 10 CFR 50.59 evaluation. It is not the intent of the report or of the NRC staff to predispose the outcome of the 10 CFR 50.59 process, but rather to provide a process that will assist licensees in reaching a proper conclusion regarding the existence of an unreviewed safety question when undertaking a digital system replacement. However, as shown in Example 5-6 of the report, when using this document as guidance for the analysis of modifications of some safety-significant systems such as the reactor protection system or an engineered safety feature system, it is likely these digital modifications will require staff review when 10 CFR 50.59 criteria are applied. Report TR-102348 states in the introduction that the guidance is supplemental to and consistent with that provided in NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations." Licensees should bear in mind that NSAC-125 has not been

endorsed by the NRC, and therefore any use of those guidelines is advisory only, and that nothing in NSAC-125 can be construed as a modification of 10 CFR 50.59. While the guidelines of NSAC-125 can be useful in the evaluation of systems, and are representative of logic used in making a 10 CFR 50.59 determination, the actual determination of whether or not an unreviewed safety question exists must be done in accordance with 10 CFR 50.59.

10 CFR 50.59(a)(2)(i) and (ii) states that a proposed change, test or experiment involves an unreviewed safety question if the probability or consequences of an accident or malfunction previously evaluated in the safety analysis report may increase, or if the possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report may be created. If during the 10 CFR 50.59 determination there is uncertainty about whether the probability or consequences may increase, or whether the possibility of a different type of accident or malfunction may be created, the uncertainty should lead the licensee to conclude that the probability or consequences may increase or a new type of malfunction may be created. If the uncertainty is only on the degree of improvement the digital system will provide, the modification would not involve an unreviewed safety question. If, however, the uncertainty involves whether or not this modification is more or less safe than the previous analog system, or if no degree of safety has been determined, an unreviewed safety question is involved.

The staff believes that two clarifications to Report TR-102348 are appropriate as follows:

1. 10 CFR 50.59 requires determination of whether "a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created." As a part of this determination, Report TR-102348 suggests looking for "any new types of system-level failures that would result in effects not previously considered in the FSAR." (For example, see TR-102348, Section 4.5, Question 6.) It is the NRC staff's position that the system-level considered in this regard should be the digital system being installed. The staff believes that this clarification is necessary because 10 CFR 50.59 does not refer to an accident or malfunction that results in a "system-level" failure different from any previously analyzed but rather to the malfunction of the equipment important to safety being modified. It is the change in the facility as described in the safety analysis report that is to be analyzed under 10 CFR 50.59 to determine if it involves an unreviewed safety question, that is, the digital equipment that replaced the analog equipment, rather than the otherwise unchanged system of which that equipment is a part is to be analyzed. This does not mean that all digital equipment usage will automatically result in an unreviewed safety question simply as a result of the use of software. Software failure, including common-mode failure, must be

considered during the 10 CFR 50.59 evaluation as a possible different type of malfunction. However, if software failure cannot cause an equipment malfunction of a different type than any previously evaluated in the safety analysis report, then no unreviewed safety question exists with respect to this criterion, and in the absence of other disqualifying criteria, the replacement can be performed under 10 CFR 50.59 without prior NRC approval. For many digital system modifications involving relatively simple systems such as discussed in example 5-5 of NUMARC/EPRI Report TR-102348, the NRC staff believes that a conclusion may be reached that there is no possibility that a different type of malfunction may be created.

As an example, when installing an upgraded digital high pressure function of the reactor trip system, it is the digital instrumentation and control circuitry associated with the high pressure reactor trip function that would be subject to the questions on failure modes and effects (equipment malfunctions) identified in the report that would be analyzed to determine involvement of an unreviewed safety question, not the entire reactor trip system. If the entire trip system is being replaced with a digital upgrade, then the entire replacement digital instrumentation and control system would be subject to the failure modes and effects analysis, not the full range of instrumentation and control systems being actuated to respond to a transient or accident.

2. 10 CFR 50.59 requires maintaining records that "include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question." Section 3.1.2 of the report points out that the use of qualitative engineering judgment is typically involved in areas that are not readily quantifiable, such as likelihood of the failure, its importance to the system and to the plant, and the practicality and incremental improvements of various options available for resolving the failure. Such judgments may be difficult to duplicate and understand at a later time. It is the NRC staff's position that the basis for the engineering judgment and the logic used in the determination should be documented to the extent practicable. This type of documentation is of particular importance in areas where no established consensus methods are available, such as for software reliability, or the use of commercial-grade hardware and software where full documentation of the design process is not available.

EPRI Report TR-102348, together with the clarifications discussed in this generic letter, can be used as guidance by licensees in both designing analog-to-digital replacements and, with respect to unreviewed safety question determinations, determining if an analog-to-digital replacement can be performed under 10 CFR 50.59 without prior staff approval.

This generic letter requires no specific action or written response. If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

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Associate Director for Projects
Office of Nuclear Reactor Regulation

Technical contact: Paul J. Loeser, NRR
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Attachment:
List of Recently Issued NRC Generic Letters

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter	Subject	Date of Issuance	Issued To
89-04, SUPP. 1	GUIDANCE ON DEVELOPING ACCEPTABLE INSERVICE TESTING PROGRAMS	04/04/95	ALL HOLDERS OF OLs OR CPs FOR NUCLEAR POWER REACTORS.
95-01	NRC STAFF TECHNICAL POSITION ON FIRE PROTECTION FOR FUEL CYCLE FACILITIES	01/26/95	ALL CURRENT LICENSEES & APPLICANTS FOR URANIUM CONVERSION & FUEL FABRICATION FACILITIES.
94-04	VOLUNTARY REPORTING OF ADDITIONAL OCCUPATIONAL RADIATION EXPOSURE DATA	09/02/94	ALL HOLDERS OF OLs OR CPs FOR NPRs, RADIOGRAPHY LICENSEES, FUEL PROCESSING LICENSEES, FABRICATING & REPROCESSING LICENSEES, MANUFACTURERS & DISTRIBUTORS OF BY-PRODUCT MAT'L, INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS, FACILITIES FOR LAND DISPOSAL OF LOW-LEVEL WASTE, & GEOLOGIC REPOSITORIES FOR HIGH-LEVEL WASTE.
94-03	INTERGRANULAR STRESS CORROSION CRACKING OF CORE SHROUDS IN BOILING WATER	07/22/94	ALL HOLDERS OF OLs OR CPs FOR BOILING WATER REACTORS EXCEPT FOR BIG ROCK POINT, WHICH DOES NOT HAVE A CORE SHROUD.
94-02	LONG-TERM SOLUTIONS AND UPGRADE OF INTERIM OPERATING RECOMMENDATIONS FOR THERMAL-HYDRAULIC INSTABILITIES IN BOILING WATER REACTORS	07/11/94	ALL HOLDERS OF OLs FOR BOILING WATER REACTORS EXCEPT BIG ROCK POINT
94-01	REMOVAL OF ACCELERATED TESTING AND SPECIAL REPORTING REQUIREMENTS FOR EMERGENCY DIESEL GENERATORS	05/31/95	ALL HOLDERS OF OLs FOR NPRs

OL = OPERATING LICENSE
CP = CONSTRUCTION PERMIT
NPR = NUCLEAR POWER REACTORS

8 ELECTRICAL POWER SYSTEMS

In this chapter of the SAR, the applicant should discuss and describe the electrical power systems at a non-power reactor facility. The electrical power systems to be described here are designed to support reactor operation. All non-power reactors require normal electrical service. Some non-power reactors may also require emergency electrical service to perform functions related to reactor safety to ensure that, given a loss of normal electric service, sufficient power will be available for mitigating the events discussed in SAR Chapter 13, "Accident Analysis." The functions to be performed and the type of emergency electrical power systems required are developed on a case-by-case basis in other chapters of the SAR. The information in Chapter 8 should be provided under two categories: normal and emergency electrical power systems.

8.1 Normal Electrical Power Systems

In this section, the applicant should discuss the design bases and functional description of the normal electrical power systems for the reactor facility. The information should include the following:

- The design bases of the normal electric power system, including how safe reactor shutdown will be ensured if offsite power is lost. (This discussion should address both short- (transient) and long-term electrical outages.)

- The ranges of electrical power capability required, for both reactor operation and utilization, in terms of various principal voltages, currents, wattage, and frequencies.

- Use of a substation, either one devoted exclusively to the reactor facility or a shared service with other activities.

- Special processing of the electrical service by such components as isolation transformers, noise limiters, lightning arresters, or constant voltage transformers.

- Schematic diagrams showing the basic distribution systems and circuits.

- Design and performance specifications of principal components, including any that are unique or not standard.

- Special routing or isolation of wiring or circuits for both reactor operations and experimental facilities. (The description should include provisions for

isolating electrical power service from instrumentation and control circuits and safety-related circuits to avoid electromagnetic interference.)

- Any deviations or exceptions from national or local electrical codes.

- Technical specifications, if required, with bases that ensure the operability of the normal electrical service, including surveillance requirements.

The applicant should discuss the spectrum of reactor operations that require normal electrical power, including the response of the reactor to both short (transient) and long interruptions of normal electrical service. The applicant should also discuss how safe reactor shutdown is ensured under all operating and accident conditions, both with and without normal electrical service available. The applicant should discuss how routine releases are controlled and monitored and how the uncontrolled release of radioactive material is prevented in the event that normal electrical power service is interrupted.

8.2 Emergency Electrical Power Systems

Emergency electrical power at a non-power reactor is defined as any temporary substitute for normal electrical service. The various functions of emergency electrical power can include operational convenience, assurance of experiment integrity, and performance of functions essential to reactor integrity. In this section, the applicant should describe all uses of emergency power systems, with emphasis on the design and functions of the emergency power systems required for reactor safety and for protecting the health and safety of the public. All non-power reactors should be designed for reactor shutdown in the event normal electrical power is lost. This includes the fail-safe actuation of the control rods. Some non-power reactors may also require emergency power to maintain the shutdown reactor in a safe condition. Some examples of uses of emergency electrical power follow:

- Power for reactor power level monitors, recorders, and necessary safety-related instruments.

- Power for effluent, process, and area radiation monitors, including recorders.

- Power for physical security control systems, information systems, or communications. (In this section, the applicant should only mention the existence of such emergency electrical power and should confine details to the facility physical security plan.)

- Placing or maintaining experimental equipment in a safe condition.

- Power for active confinement or containment engineered safety feature (ESF) equipment and control systems, such as blowers, fans, or dampers, and heating, ventilation, and air conditioning equipment. (This is the equipment necessary to maintain equipment and personnel habitability or to control concentrations or release of airborne radioactive material and to mitigate accident consequences.)

- Power for coolant pumps or systems that remove residual heat from the fuel.

- Power for the emergency core cooling system, including instrumentation and control systems.

- Power for other ESF equipment, if applicable.

- Power for emergency area lighting and communication equipment.

- Power for those instrument and control systems necessary to monitor reactor shutdown. (These could include fuel temperature, control rod positions, or fission product monitors.)

The types of emergency electrical power systems discussed in this section should be commensurate with the required design bases developed in other chapters of the SAR, such as Chapter 4, "Reactor Description"; Chapter 5, "Reactor Coolant Systems"; Chapter 7, "Instrumentation and Control Systems"; Chapter 9, "Auxiliary Systems"; Chapter 10, "Experimental Facilities and Utilization"; Chapter 11, "Radiation Protection Program and Waste Management"; and Chapter 13. These systems may range from automatic start generators to wet-cell or dry-cell batteries. In this section, the applicant should present a detailed functional description and circuit diagrams. In the design bases, the applicant should discuss if non-interruptible electrical power is required in the transfer from normal to emergency electrical service and if the transfer is manual or automated. The design bases should also provide voltage and power requirements for the emergency electrical power systems, the time duration over which these could be needed, and assurance that fuel will be available for the time required. The designs of the emergency electrical systems should provide that any use for non-safety-related functions could not cause loss of necessary safety-related functions. The design discussion should show how the emergency power supply system is isolated or protected, if necessary, from transient effects, such as power drains, short circuits, and electromagnetic interference. If the emergency electrical power systems are required during analyzed accidents, the designs should include this capability. The minimum emergency electrical power functions that would be required to protect the health and safety of the public should be included in technical specifications based on the discussions in Chapter 8. The technical

specifications should also identify the minimum equipment to be supplied by the emergency power system, important design parameters, and surveillance and inspection functions that ensure operability of the emergency electrical power systems and the supplied equipment.

9 AUXILIARY SYSTEMS

In this chapter of the SAR, the applicant should discuss the auxiliary systems at the reactor facility. Auxiliary systems are those systems not fully described in other chapters of the SAR that are important to the safe operation and shutdown of the reactor and to the protection of the health and safety of the public, the facility staff, and the environment. The applicant should provide sufficient information for all auxiliary systems to support an understanding of the design and functions of the systems, with emphasis on those aspects that could affect the reactor and its safety features, radiation exposures, and the control or release of radioactive material.

For each auxiliary system, the applicant should discuss the capability to function as designed without compromising reactor operation or the capability to shut down the reactor. This capability should be shown for normal operation and reactor accident conditions. The applicant should include the following information for each auxiliary system:

(1) design bases

(2) system description, including drawings and specifications of principal components and any special materials

(3) operational analysis and safety function

(4) instrumentation and control requirements not described in Chapter 7, "Instrumentation and Controls Systems," of the SAR

(5) required technical specifications and their bases, including testing and surveillance

The design, operation, and use of non-power reactors vary widely. Typical auxiliary systems that may be discussed in this chapter of the SAR include the following:

- Heating, ventilation, and air conditioning (HVAC) systems for normal reactor operation. (The applicant should discuss any engineered safety feature functions of the HVAC systems for accident conditions in Chapter 6, "Engineered Safety Features," of the SAR.)

- Handling and storage of reactor fuel, both new and irradiated, including tools, vaults, racks, pools, shields, casks, and preparations for shipping.

- Fire protection systems that could affect reactor safety or protection of licensed materials.

- Communication systems, both internal and external to the facility.

- Control, storage, or use of byproduct, source, and special nuclear material produced, used, or possessed under the reactor operating license. (The applicant should also discuss applicable laboratory facilities designed to handle or use byproduct materials other than radioactive waste.)

- Cover gas control and processing at certain reactors, such as a tank reactor using heavy water neutron moderators or reflectors. (The applicant should include features of closed primary systems designed to control radiolytic gases, if applicable.)

- Auxiliary coolant systems for experimental facilities and other equipment and uses that are not part of the primary coolant system described in Chapter 5, "Reactor Coolant Systems," of the SAR.

- Demineralizer resin regeneration system.

- Control and storage of radioactive waste and reusable radioactive components (e.g., experiments). (The applicant should describe the systems and show how they are designed to perform the design-basis functions derived in Chapter 10, "Experimental Facilities and Utilization," or Chapter 11, "Radiation Protection Program and Waste Management," of the SAR.)

- Control of contaminated air, gas, or liquid from experimental facilities. (The applicant should describe the systems and show how they are designed to perform the design-basis functions derived in Chapters 10 or 11.)

- Compressed air or gas systems for reactor operating systems and experiment equipment.

- Auxiliary physical protection and access control that are not part of the facility physical security plan.

These examples are not intended as a complete list of auxiliary systems that may be discussed in this chapter of the SAR. The descriptions of some auxiliary systems may be better suited to other chapters, which should be referenced in this section.

9.1 Heating, Ventilation, and Air Conditioning Systems

All used spaces in a facility may require HVAC systems to provide acceptable environments for personnel and equipment. In this section the applicant should describe how temperature and humidity are controlled and discuss the bases, including how the control function is integrated into the HVAC systems. The applicant should address the prevention of uncontrolled releases of airborne radioactive effluents to the environment for normal operation. The discussions should contain explanations of how airborne radioactive material from operations and experiments is limited in occupied areas to maintain radiation exposures below the requirements of 10 CFR Part 20 and the facility ALARA (as low as is reasonably achievable) program guidelines. Controls limiting diffusion or leakage of radioactive material to adjacent spaces should be presented. The applicant also should discuss how air exhaust systems or stacks are designed to reduce the radiological impact on the unrestricted environment during normal reactor operations.

Analyses of radiation exposures in Chapter 11 should include the applicable normal operating characteristics of the HVAC systems described in this section of the SAR. The interactions among airflow patterns in the reactor room, the air exhaust stacks, and the effluent and continuous air monitors should be discussed. If the HVAC systems also are designed to mitigate the consequences of accidents, the engineered safety features should be noted in this section of the SAR but described in detail in Chapter 6.

The applicant should describe instrumentation and control systems that control the release of radioactive material (automatic and manual) in Chapters 7 and 11 of the SAR. The information in this section of the SAR should be sufficient to support an understanding of the safety functions of radiation sensors that initiate alarms and automatic closures, fail-safe dampers, interlocks, and function displays during normal reactor operations. The applicant should discuss the bases and purpose of technical specifications that apply to the HVAC systems, including calibrations, testing, and surveillance.

The applicant should discuss the possible effects of malfunctions of the HVAC systems on safe reactor operation or on the release of airborne radioactive material during normal reactor operation. The radiological effects of malfunctions should be discussed in Chapter 11.

9.2 Handling and Storage of Reactor Fuel

In this section the applicant should discuss the life cycle of reactor fuel from the time it enters its jurisdiction until it is released from such jurisdiction. For most non-power reactors this means from arrival on site until shipment off site. The

discussions should include descriptions of the tools, cranes, racks, design features, and administrative controls that protect new fuel from damage.

The applicant should provide analyses and discuss how subcriticality is ensured (k_{eff} not to exceed 0.90) under all conditions, except during transportation off site. During transportation, the shipping container license is applicable. (Existing usage with k_{eff} greater than 0.90 will be acceptable if the usage was previously reviewed and approved by NRC.) The applicant should address the applicability and implementation of 10 CFR 70.24, which addresses criticality monitors.

The applicant should discuss briefly the methods that ensure the prudent control of fuel. The discussion should include a description that does not contain proprietary or safeguards information of the physical protection of fuel against theft or diversion in the facility physical security plan. Reference can be made to the physical security plan, which should be treated as proprietary or safeguards information.

The applicant should address the handling, storage, and shipment of new and irradiated fuel. Included should be a discussion of the tools used to insert or remove fuel from the core as well as the physical and administrative methods specified to control their use. The details should include the design of handling tools, transfer casks, and other radiation shields; the design of storage facilities; methods of preparing fuel for shipping; and shipping methods. Descriptions of procedures and systems for the storage and handling of irradiated fuel should include radiation shielding, protection from physical damage, physical control, and sufficient cooling to prevent overheating and surface corrosion. Irradiated fuel cooling systems and methods may be described in detail in Chapter 5 of the SAR if they are integral to the reactor coolant system. Otherwise, the discussion should be in this section.

During storage or handling, if a loss of fuel or cladding integrity could result in the release of fission products, the applicant should discuss the mechanisms and analyze the consequences in Chapter 13, "Accident Analyses," of the SAR. Detailed discussions of radiological considerations for storing and handling fuel should be provided in Chapter 11 and should include, if applicable, the implementation of 10 CFR 73.6(b) concerning self-protection for irradiated highly enriched uranium fuel.

The applicant should include in its discussions of fuel handling and storage the bases of related technical specifications, including inspections, testing, and surveillance and applicable administrative controls and procedures.

Additional information concerning this topic can be found in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.19-1991.

9.3 Fire Protection Systems and Programs

In this section the applicant should describe the systems and programs designed to protect the reactor facility from damage by fire and discuss how the facility meets all local building and fire codes. For a new facility this could be a general discussion of how the facility meets local fire and building codes. Documentation from the local authority that authorizes the construction or verifies compliance with local codes could be submitted as part of the discussion. NRC construction inspectors would review design features for fire protection during facility construction. For existing facilities requesting license renewal for which the original construction documentation might be difficult to reproduce, the applicant could submit the results of a recent fire inspection to show compliance with local codes. The applicant should discuss additional active and passive design features required by the reactor design characteristics. Further, the discussion should address the potential for release of radioactive material as a result of a fire. Active systems might include sprinkler, suppression, hand extinguisher, and detection systems. Passive systems might include fire walls and doors, isolation, and control of combustible materials.

The applicant should discuss how the potential release of radioactive materials as a result of fires in the reactor room and other applicable spaces was considered in the design of the facility. The discussion should include the reactor and all facilities where special nuclear material and other radioactive materials are stored or used under the reactor license. It should include any possible effects of a fire on safe shutdown of the reactor. The objectives of the fire protection program should include the following:

- preventing fires, including limiting combustible materials
- detecting, controlling, and extinguishing fires to limit consequences
- protecting reactor systems so that a continuing fire would not prevent safe reactor shutdown or cause an uncontrolled release of radioactive material

The applicant should discuss the bases of any technical specifications, including testing and surveillance, as they relate to the fire protection systems and programs. The discussion should also include the relationship between fire protection plans, operating procedures, and the facility emergency plan.

ANSI/ANS 15.17-1987 contains general information on fire protection. The applicant may also consult National Fire Protection Association, NFPA 802, 1993 Edition.

9.4 Communication Systems

The applicant should describe the communication systems that will be used at the facility for which public disclosure is not limited by the physical security plan. Communication systems used between the control room, the reactor room, reactor access point or top, reactor utilities rooms, experiment areas, and all other required areas should be described. Such systems as telephone, paging, radio, or video that will be used to announce changes of reactor status to experimenters, summon supervisory operators, request radiation protection assistance, and announce emergencies should be discussed. For a complete description of communications, the applicant should also summarize briefly in this section the communication systems used for emergency or physical security purposes (this discussion should not contain proprietary or safeguards information).

The discussions of communication systems should include the bases of any related technical specifications, including testing and surveillance.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

The 10 CFR Part 50 operating license applies to possession and operation of the reactor, possession and use of byproduct material produced by the operation of the reactor, and, to the extent authorized, the receipt, possession, and use of other byproduct, source, or special nuclear material (material) needed for operation of the reactor and its experimental programs. Examples include sources for radiation monitor calibration, depleted uranium for shielding of experiments, reactor fuel, fission plates for thermal columns, and fission chambers for reactor monitoring and control.

The NRC regulatory approach is to include in the reactor license only material that is produced by the reactor or is required to directly operate the reactor and associated experimental facilities. Other material at a non-power reactor facility is authorized by an NRC byproduct, source, or special nuclear materials license. If the facility is located in an Agreement State, an Agreement State license may also exist. This other material is normally not required to operate the reactor or associated experimental facilities. A special case exists for material that is received for irradiation from another licensee. If this material is to be placed into the reactor for irradiation within 31 days of receipt, it may be possessed under the reactor license. However, this authorization for receipt and possession must be specifically stated in the reactor license. If more than 31 days should pass before the material is placed into the reactor, it should be included in an NRC or Agreement State materials license until irradiation occurs. Further information on this subject may be found in memoranda dated March 8 and August 18, 1988,

from Dennis M. Crutchfield, Director, Division of Reactor Projects - III, IV, V, and Special Projects, to NRC regional administrators (provided as Appendices 9.1 and 9.2).

The receipt, possession, or use of materials authorized by the reactor license may occur in the reactor room and contiguous operational spaces and also in laboratory spaces for research and development purposes. Some licensees take a narrow view, transferring material produced in the reactor to another NRC or Agreement State license when the material is removed from the reactor pool. Others take a broad view and allow all materials produced by the reactor or authorized by the license to be in various locations and laboratories in the facility. Spaces could be used to process and package byproduct materials for shipment or could be used for performing experiments involving the byproduct materials. A broad view of materials and areas authorized by the 10 CFR Part 50 reactor license avoids maintaining multiple licenses and allows, in some cases, indemnity protection for materials in laboratories and other auxiliary spaces. The applicant should clearly state the materials and areas of the facility requested to be authorized by the reactor license. The reactor license and technical specifications also will include regulatory conditions that apply to the possession, management, and use of such materials, including requirements stated in 10 CFR Parts 20, 30, 40, or 70.

The applicant should discuss in this section laboratories under the reactor license in which reactor-licensed material will be used. This discussion should address all five factors noted at the beginning of this chapter for any such auxiliary laboratories. The applicant should specify the types and quantities of radionuclides authorized, as well as the general types of experiments or uses. Radiological design bases for handling radioactive materials and radioactive waste should be derived from Chapter 11 of the SAR. These design bases may apply to chemical, fume, and air exhaust hoods; to drains for radioactive liquids; and to radiation shields. The discussions should show how the physical security and emergency plans apply to the licensed spaces and possession of byproduct materials. The applicant should discuss the bases for special operating procedures. The administrative aspects of the use of materials in these areas should be addressed in Chapter 12, "Conduct of Operations," of the SAR.

9.6 Cover Gas Control in Closed Primary Coolant Systems

Some non-power reactor designs have a reactor core tank system in the primary coolant loop that is sealed against the atmosphere. At some of these reactors, heavy water (D_2O) is used as a moderator, reflector, or coolant, and the admixture of atmospheric water vapor and loss of the heavy water must be prevented. At others a primary system operates with ordinary light water at or above atmospheric pressure. In both types of systems, radiolytic decomposition of the water leads to

a hydrogen-oxygen mixture that could reach explosive concentration without processing.

For such reactors, the applicant should discuss gas handling in closed primary coolant systems, addressing the five factors listed at the beginning of this chapter. The discussions should describe cover gas systems that circulate, decontaminate, recover, store, monitor, and dispose of the gas, as well as process or recombine radiolytic components. The design bases should define which inert gases are acceptable to use, their impact on safe reactor operations and shutdown, the bases for limiting concentrations of hydrogen-oxygen mixtures, and the methods for controlling the concentrations. The discussions should include the bases of any required technical specifications applicable to cover gas systems, including testing and surveillance.

9.7 Other Auxiliary Systems

As noted previously, a unique set of auxiliary systems could exist at a non-power reactor. The above examples are found at many reactor facilities; other facilities may have additional auxiliary systems. The applicant should describe and analyze all auxiliary systems, address the five factors listed at the beginning of this chapter, and include the following:

- Demonstrate that the auxiliary system will function under analyzed reactor accident conditions, if required.
- Demonstrate that the auxiliary system and any malfunction could not create conditions or events that could cause an unanalyzed reactor accident or the uncontrolled release of radioactive material beyond those analyzed in Chapter 13 of the SAR.
- Demonstrate that the auxiliary system could not prevent safe reactor shutdown.
- Provide a discussion of and the bases for any technical specifications related to the auxiliary system, including testing and surveillance.

9.8 References

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.17, "Fire Protection Program Criteria for Research Reactors," 1987.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.19, "Shipment and Receipt of Special Nuclear Material (SNM) by Research Reactor Facilities," 1991.

National Fire Protection Association, NFPA 802, "Recommended Practice for Nuclear Research and Production Reactors," 1993.

Appendix 9.1

Regulatory Responsibilities for Byproduct Materials in Non-Power Reactors

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 8, 1988

MEMORANDUM FOR: Frank J. Congel, Director
Division of Radiation Safety and Safeguards, RI

Douglas M. Collins, Director
Division of Radiation Safety and Safeguards, RII

John A. Hind, Director
Division of Radiation Safety and Safeguards, RIII

Richard L. Bangart, Director
Division of Radiation Safety and Safeguards, RIV

Ross A. Scarano, Director
Division of Radiation Safety and Safeguards, RV

FROM: Dennis M. Crutchfield, Director
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

SUBJECT: REGULATORY RESPONSIBILITIES FOR BYPRODUCT MATERIALS IN NON-POWER REACTORS

In a memorandum dated June 8, 1987, Region IV requested guidance for determining cases where licensed material in a non-power reactor facility may be covered by a NRC material license or an Agreement State license, rather than the reactor license. This issue becomes important in determining compliance and issuing notices of violation involving licensed material in a reactor facility. All regions were asked to comment on this issue. After consideration of your comments, we are providing the following guidance. The guidance has been coordinated with NMSS, GPA, and OGC.

1. Generic guidance related to this issue is contained in Inspection Manual Chapter 2882, Appendices 1 and 2. Normally, material within a non-power reactor facility will generally be assumed to be possessed by the reactor licensee, unless there is prior documentation approved by NRC, or some other clear demonstration that the licensed material is covered under another license.

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NRR/PDSNP
Ext. 21102

2. Consistent with #1 above, NMSS does not normally issue separate licenses which authorize possession of licensed material within an operating reactor facility. If a reactor facility license is silent with regard to possession of byproduct material, it should be amended. NRC normally exercises exclusive federal jurisdiction within operating reactor facilities.

3. All byproduct material which is to be inserted into a reactor, or which is removed from the reactor, must be covered by the reactor license while the material is within the facility.

4. The facility boundaries for a non-power reactor are normally defined by the Safety Evaluation Report or Technical Specifications. In the absence of identifiable facility boundaries, the Regions should establish a facility boundary with the licensee for compliance purposes, and the boundary should be specified in TS or FSAR.

5. As indicated in Manual Chapter 2882, Appendix 2, there are exceptions to the above guidelines, and specific cases can be complex. Questionable cases should be referred to Headquarters for resolution along with a proposed course of action.

Questions concerning this guidance or specific cases should be referred to this Division for resolution. We will coordinate with NMSS, GPA, and OGC as appropriate.

Dennis M. Crutchfield

Dennis M. Crutchfield, Director
Division of Reactor Projects - III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Appendix 9.2

License Condition for Byproduct Material To Be Irradiated in a Non-Power Reactor

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 18, 1988

MEMORANDUM FOR: Stewart D. Ebneter, Director
Division of Radiation Safety and Safeguards, RI

J. Phillip Stohr, Director
Division of Radiation Safety and Safeguards, RII

John A. Hind, Director
Division of Radiation Safety and Safeguards, RIII

Richard L. Bangart, Director
Division of Radiation Safety and Safeguards, RIV

Ross A. Scarano, Director
Division of Radiation Safety and Safeguards, RV

SUBJECT: LICENSE CONDITION FOR BYPRODUCT MATERIAL TO BE IRRADIATED IN A NON-POWER REACTOR

At the Reactor Health Physics Counterpart Meeting of May 18-19, 1988, a question arose as to what the appropriate license condition should be for possession of byproduct material at non-power reactor facilities (see Enclosure 1, item 8). The question was prompted by a statement in guidance provided to the Regions in a memorandum dated March 8, 1988 (Enclosure 2). The statement in enclosure 2 appears in item 2 and reads as follows:

> ...If a reactor license is silent with regard to possession of byproduct material it shall be amended...

All non-power reactor licenses have a license condition which permits the licensee to - "possess, but not to separate such byproduct material as may have been produced by operation of the facility." This license condition, however, does not adequately cover byproduct material received at the facility which is going to be irradiated in the reactor. Enclosure 2 (Memorandum, D.M. Crutchfield to Regions, March 8, 1988), item 3 states that -

> All byproduct material which is to be inserted into a reactor, or which is removed from the reactor, must be covered by the reactor license while the material is within the facility.

In order to satisfy this condition, the license condition dealing with possession of byproduct material should be amended if a licensee receives

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byproduct material which is to be irradiated in the reactor. The license condition should read as follows:

> Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material," (and Part 70)*, to receive byproduct material which is to be irradiated in the reactor within 31 days of receipt, and to possess, but not separate, such byproduct (and special nuclear)* materials as may be produced by operation of the facility.

* Delete if Part 70 not applicable

Licensees must request an amendment to their license to include this condition if they receive byproduct material to be irradiated by their reactor, unless the material is covered by another license before it is inserted into the reactor.

Violations involving byproduct material that is to be irradiated in a non-power reactor should generally be charged against the reactor license unless some other specific documentation has been developed by the licensee. In this regard the statement in enclosure 2, item 3 is modified to read as follows:

> All byproduct material which is to be inserted into a reactor, should be covered by the reactor license; byproduct material which is removed from the reactor must be covered by the reactor license.

Dennis M. Crutchfield
Dennis M. Crutchfield, Director
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
As stated

10 EXPERIMENTAL FACILITIES AND UTILIZATION

In this chapter of the SAR, the applicant should describe and discuss the experimental facilities at the non-power reactor facility, their intended use, and the experimental program. This chapter of the SAR should contain a description of the proposed experimental program and the safety analyses for each type of experimental facility. The design, construction, and placement of each experimental facility should be analyzed for inherent safety questions that exist apart from the experiments accommodated therein. The experiments should be analyzed by using a separate experiment safety analysis methodology to show compliance with the technical specifications, primarily the associated limiting conditions for operation (LCOs) as indicated in Chapter 14, "Technical Specifications," of this format and content guide. The applicant should provide sufficient information to demonstrate that no proposed operations involving experimental irradiation or beam utilization will expose reactor operations personnel, experimenters, or the general public to unacceptable radiological consequences. In addition to the guidance in this format and content guide, Regulatory Guide 2.2 and 2.4 (see Appendices 10.1 and 10.2) contain guidance on technical specifications and experimental programs that may be useful to the applicant in preparing the SAR.

Non-power reactors may be used for many purposes including radiation physics, chemistry and biology studies, materials irradiation, radionuclide production, and educational purposes. The experimental facilities may penetrate the reactor core or reflector or be located near the core. Neutron or other radiation beams can be extracted from the core region through the biological shield. At many non-power reactors, the experimental facilities are integral components of the entire reactor.

Utility, integrity, longevity, versatility, diversity, and safety should be considered for the experimental facilities in the same manner they are considered for the reactor core and its operational components and systems. Therefore, the safety analyses of the reactor facility should include the experimental facilities and their interactions with the reactor components and systems. If changes in reactor operating characteristics are considered, potential interactions between the core and the experimental facilities should be analyzed.

Experimental programs and the range of experiments vary widely among non-power reactor facilities. Furthermore, as the licensee and the facility users gain experience and as technology develops, the experimental program and many of the specific experiments may change over the life of the reactor. This makes it very difficult and impractical for the applicant to describe specific experiments in the SAR. The applicant should describe and analyze in this chapter of the SAR and incorporate into the facility technical specifications enveloping conditions of

experiment attributes such as reactivity limits or material properties to allow the greatest flexibility in the experimental program. Potential experimental needs should be considered when establishing these limiting safety aspects in the SAR, so that determinations in accordance with 10 CFR 50.59 can be made expeditiously. Experience has shown that most licensees have successfully implemented changes in experimental programs without prior NRC approval under the provisions of 10 CFR 50.59. This regulation allows licensees to (1) make changes in the facility as described in the SAR, (2) make changes in the procedures as described in the SAR, and (3) conduct tests or experiments not described in the SAR without prior NRC approval, unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. A proposed change, test, or experiment is deemed an unreviewed safety question if (1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, (2) a possibility for an accident or malfunction of a type different from any evaluated previously in the SAR may be created, or (3) the margin of safety as defined in the basis for any technical specification is reduced.

Some non-power reactors are operated as critical facilities to demonstrate fuel loading and perform reactor physics studies. In such cases, the reactor itself can also be considered an experimental facility. In this case, the safety analysis and experiment technical specifications will include the limitations on core configurations and operational limitations when the core is the experiment.

The applicant should provide an analysis to demonstrate that the reactor and experimental facilities can be operated safely. This analysis should include the range of normal operations, accidents, and malfunctions of experimental facilities. It should address any impact the experimental facility imposes on the reactor and any impact the reactor imposes on the experimental facility.

Consideration should be given to the possibility of the experimental facility causing an accident that requires analysis in Chapter 13, "Accident Analyses," of the SAR. In some cases, the failure of an experiment can be the maximum hypothetical accident (MHA) for the reactor. This possibility is most prevalent with fueled experiments. Experiments can result in the maximum uncontrolled reactivity addition accident at a facility. Limiting experiment failure should be considered in Chapter 13.

The SAR should be written to accommodate the nature of varying experiments and meet the requirements of future experimentation. The applicant should show that there is no undue risk to the health and safety of the public.

Discussions in this chapter of the SAR should include design bases, facility descriptions, functional and safety analyses, and the applicant's safety conclusions

for all experimental facilities. The structural design and its potential impact on reactor operation should be analyzed for those experimental facilities that are permanently attached to the reactor support structure, reactor vessel, or pool hardware. For those experimental facilities that penetrate the reactor vessel below any primary coolant water level, an analysis of the experimental design should demonstrate that the design is resistant to failure and that if failure occurs, it is bounded by the analysis in Chapter 13 of the SAR for a loss-of-coolant accident (LOCA). The placement or use of experimental facilities shall not compromise the functionality of any reactor safety system or engineered safety feature. The discussion should include the capabilities, limitations, and controls on reactor operation, including engineering or procedural controls for experiments, that ensure radiation doses do not exceed the requirements in 10 CFR Part 20 and are consistent with the facility program to keep exposure to radiation as low as is reasonably achievable (ALARA).

Because of the potentially unlimited variety of experiments that can be accommodated in a non-power reactor, the applicant should show that administrative controls are adequate to ensure that the health and safety of the public are protected. The actual experiments to be performed need not be discussed in detail in this chapter of the SAR, but the limiting and enveloping features of the experiments and the administrative procedures used by the applicant to review, approve, and safely control experiments should be described. The applicant should provide the bases for experiment-related LCOs and for a detailed description and justification of the experiment review and acceptance program that are then specified in the technical specifications.

10.1 Summary Description

In this section of the SAR, the applicant should briefly describe the principal features of the experimental and irradiation facilities associated with the reactor. The applicant should discuss the scope of the experimental program and define what is considered to be an experiment. Discussions should include experimental compatibility with normal reactor operations and accidents and measures taken to avoid interference with the reactor shutdown and other systems.

The applicant should include the following information:

- general focus of the experimental program (radiation science, medical, materials testing, teaching, etc.)
- a list of experimental facilities
- basic type of experiments that will be conducted (incore, thermal column, external beam, etc.)

- a brief description of experiment monitoring and control and the interaction between the experiment and the reactor control and safety systems

- a brief overview of design requirements for the experiment and of the review and approval process

Simple block diagrams and drawings may be used to show the location, basic function, and relationship of each experimental facility to the reactor. The summary description should contain enough information to support an overall understanding of the functions of the experimental facilities and the experiment review and approval process.

A brief description of typical experimental facilities found at non-power reactors follows. This list, however, is not exhaustive.

- *Incore facilities* are those facilities that are surrounded on at least two sides by fuel. Such facilities are commonly called void tubes, flux traps, central irradiation facilities, incore irradiation facilities, radioisotope facilities, dummy and demountable fuel elements, fast and thermal neutron irradiation facilities, or central and offset thimbles. If the cross-sectional area of an incore facility is greater than 16 square inches, the reactor is considered a test reactor if the thermal power level exceeds 1 megawatt. The facility is also considered a test reactor if there is a circulating loop through the core for conducting fuel experiments and the reactor power exceeds 1 megawatt.

- *In-reflector facilities* are those facilities that are physically located in the reflector and are surrounded either on all sides or on at least three sides by reflector material. In-reflector facilities might include lazy susans, void tubes, flux traps, thimbles, standpipes, or thermal neutron irradiation facilities.

- *Automatic transfer facilities*, sometimes called rabbits, are a special class of incore and in-reflector experimental facility. They often protrude into or are adjacent to the core or reflector and contain the experimental material. However, rabbit facilities allow the experimental material to be moved quickly into and out of the desired flux region of the core by pneumatic, hydraulic, or mechanical means. The material can be moved while the reactor is operating if limits on reactivity changes in the reactor are observed.

- *Beam ports* are hollow tubes that can abut the core or protrude into the core or reflector. However, unlike the previously described incore and in-reflector facilities, they may or may not contain the experimental material.

Instead, they may be used to channel radiation from the core to a position, usually outside the reactor vessel and the biological shield, where the experiment is located. Neutrons and gamma-ray beams are tailored to suit the experimental needs.

- *Thermal columns* function in a way similar to that of beam ports in that they allow transport of radiation away from the core to areas where the experiment is located. Rather than a tube to guide radiation beams, they consist of a neutron moderator, typically a large volume of graphite blocks, enclosed in a container. The column is located at one face of the reactor in place of the reflector. Fast neutrons are thermalized within the moderator and may be used outside or inside the reactor shield for experiments.

- *Irradiation rooms* or other dry cavities in the biological shield may be located adjacent to the reactor core (or the core moved into position) for irradiation of large volumes of material or objects.

- *Cold neutron source* is a special type of beam port. The neutrons are passed through a very cold moderator, such as frozen heavy water or hydrogen cooled by active cryostatic systems, to reduce their energy to below the normal thermal range and increase the relative flux densities of very slow neutrons. This allows a wider range of materials to be probed and the probability of interactions in some materials to increase. These cold neutron beams are sometimes used with neutron guides, which can carry the neutrons substantial distances from the reactor without significant losses.

  If the source includes hydrogenous neutron moderators and cryogenic equipment, unique safety questions could arise. Such a source could be included in the initial reactor design, obtained by later modification of an existing reactor, or installed in an approved experimental facility, such as a beam port or thermal column, of an existing reactor.

10.2 Experimental Facilities

In this section of the SAR, the applicant should describe and discuss in detail all experimental facilities. The design should ensure that risks to the public, staff, and experimenters are acceptable.

The applicant should discuss specifications and important design and operating parameters for the experimental facilities and give design details and the physical size, including all dimensions. Simplified engineering drawings or schematics may be used, especially for more complex facilities. The applicant should discuss the location of the experimental facility in relation to the core, safety systems, core

support, neutron detectors, coolant system components, and any other reactor systems, components, or structures.

Features of the experimental facility that could interfere with safe reactor shutdown or with adequate core cooling shall be included. The source of experiment cooling and any dependence on or interaction with the reactor coolant system should be discussed. For any experimental facilities that require a special cooling system independent of the reactor primary coolant system, the technical evaluation considerations are similar to those for the reactor coolant system. The applicant should follow the applicable guidance in Chapter 5, "Reactor Coolant Systems," in this format and content guide for independent experiment cooling systems.

Since integrity of the experimental facility is important, the capability to contain or withstand any postulated pressure pulse and preclude any inadvertent primary coolant leakage or facility collapse should be discussed. Analysis should be presented for vessel or pool penetrations that could affect the risk of a LOCA. For experimental facilities that penetrate the reactor vessel below the water level of the pool surface, the applicant should show that if a LOCA does occur, the consequences are bounded by the LOCA analysis in Chapter 13 of the SAR. The LOCA analysis should also apply to the experimental facility if the facility must be cooled.

The applicant should discuss the materials used in the construction of the experimental facilities, addressing radiation and chemistry impacts. Materials and design, including physical dimensions, should limit any rapid reactivity insertion if the facility is suddenly voided or flooded. The supporting analysis should be included in Chapter 13 of the SAR, where change in the limiting experiment failure reactivity should be analyzed. The bases of applicable LCOs for the technical specifications should be developed and justified.

The radiological considerations associated with the design and use of the experimental facilities, generation of radioactive gases (including argon-41), release of fission products or other radioactive contaminants, and exposure of personnel to neutron and gamma beams should be summarized in this section of the SAR and discussed in greater detail in Chapter 11, "Radiation Protection Program and Waste Management."

Direct radiation streaming from the experimental facilities and the effect of scattered (skyshine) radiation should be discussed briefly in this section of the SAR and analyzed in Chapter 11. The analysis should clearly show all pertinent radiation sources, distances, dimensions, materials, radiation scattering, and material attenuation factors.

Facilities that could fail and release argon-41 or other airborne radioactivity into the facility air or to the environment should be analyzed. The analysis in Chapter 13 of the SAR and summarized in this section should show the concentrations of radioactive material in the experimental facility, the release pathway, and the concentrations of radioactive material in the reactor facility and the outside environment. In some cases, this type of failure could be the MHA for the reactor, which should be analyzed in Chapter 13.

Any radiation monitors specifically designed and placed to detect experiment radiation and to monitor personnel should be discussed briefly in this section of the SAR and discussed in greater detail in Chapters 7, "Instrumentation and Control Systems," and 11. Additionally, reactor operating characteristics, including scrams and runbacks associated with experimental measurements, should be analyzed.

Any physical restraints, shields, or beam catchers, both temporary and permanently installed, that are used to restrict access to radiation areas associated with experimental facilities should be described and analyzed. Descriptions and analyses should show that the placement, dimensions, and materials (1) are sufficient to limit the expected radiation doses to experimenters, reactor operators, and other personnel to levels below those specified in 10 CFR Part 20 and (2) are consistent with the facility ALARA program. For reactor beams, the applicant should describe the approach to compliance with the regulations concerning access to high radiation areas and very high radiation areas, as appropriate. These issues should be analyzed in Chapter 11 and summarized in this section of the SAR.

Permanently installed safety instrumentation for the experiment facility, including the location and function of sensors, readout devices, and scram or interlock capabilities, should be discussed briefly in this section of the SAR and in greater detail in Chapter 7.

In addition to the applicable information discussed above, this section should contain the following for cold sources because of their unique safety considerations:

- description of the cold source facility, including the operating principles and the design of the systems and components
- description of the relevant ambient environmental conditions, such as radiation intensities and actual thermal sources, and their potential impact on the cold source components and materials
- discussion of the physical and chemical characteristics of the neutron moderator and coolant fluids, handling systems, volumes and states of

matter at the operating temperatures and at ambient temperature, and all hardware, shielding, control, and safety features of the cold source

- description of all applicable operations for preparing and using the facility, such as inserting and removing moderator and coolant fluids; storing, sensing, and measuring inventories; determining contamination and leakage; determining chemical and physical changes of fluids and interactions with hardware; and producing and monitoring the operating temperatures and pressures

- discussion of the effect of the radiation environment, such as radiolysis and other radiolytic changes, on fluids, formation or release of ozone and other gas, heating of fluids and components caused by radiation and by conduction or convection from nearby shield and structural components, and radioactivity of fluids and components,

- discussion of the effect of leakage of fluids, such as toxicity, flammability, and potential to detonate, addressing changes in composition, mixtures, or other characteristics of the neutron moderator and coolant fluids with use and cycling

- description of provisions for safe (passive) shutdown of the cold source and reactor as a system

Technical specifications for experimental facilities, as discussed in Chapter 14 of this format and content guide, should be presented and justified in this section of the SAR.

10.3 Experiment Review

Because of the variety of experiments that can be conducted in a non-power reactor, the administrative controls of the applicant should be adequate to ensure the protection of the public. The administrative procedures used by the applicant to review and approve experiments should be described in detail in this section of the SAR and summarized in Chapter 12, "Conduct of Operations," and the operating limits should be included in the technical specifications. The applicant should state the safety analysis requirements for the experiment safety analysis report and the experiment review and approval methodology and should briefly discuss the authority and role of the experiment review committee.

The applicant should discuss experiment classification and approval authority. The applicant should state the methodology used to categorize proposed experiments according to risk potential, the categories expected at the reactor facility, and the safety requirements for each category. The methodology should describe how

10 CFR 50.59 will be used in the review of all experiments not described in the SAR, as well as how Regulatory Guides 2.2 and 2.4 (see Appendices 10.1 and 10.2) will be used. The appropriate level of review authority required to approve experiments in each category should be discussed. The applicant should be specific in delineating the bounds of the risk categories, such as gram amounts, temperature degree limits, radioactivity limits, or reactivity limits, and should develop the bases of applicable technical specifications. The experiment safety analysis process should demonstrate compliance with these limits and establish any special controls on the experiment.

The applicant should discuss administrative controls for the experiment and list the administrative controls used to protect facility personnel and the public from radiation or other possible hazards, such as chemical releases, in the implementation of the experimental program. Where appropriate, the discussion should delineate areas where reactor operations and experiment operations are performed under separate authority and by different personnel. The discussion should include access to experimental facilities and areas, lockout procedures, communications with reactor operating personnel, alarms, and reactor scrams. The administrative procedures should address basic protection and recovery procedures after a malfunction of experiments or experimental facilities.

The applicant should discuss the generic safety assessment of experimental materials and limitations, consistent with the guidance in Regulatory Guide 2.2, from which experiment and reactor LCOs are incorporated in the technical specifications. Malfunctions or failures of experiments with significant potential for radiological consequences should be analyzed in Chapter 13 of the SAR and summarized in this section. For some reactors, the most serious accident or the MHA could be initiated by an experiment malfunction. Areas of assessment should include the following:

- fissile materials and radiological risks from radiation fields or release of radioactive material
- trace elements and impurities
- effects on reactivity, both positive and negative
- explosive, corrosive, and highly reactive chemicals
- radiation-sensitive materials
- flammable or toxic materials
- cryogenic liquids

- unknown materials
- radiation heating or damage that could cause experiment malfunction
- heating that could cause departure from nucleate boiling on surfaces

Appendix 10.1

Regulatory Guide 2.2

Development of Technical Specifications for Experiments in Research Reactors

U.S. ATOMIC ENERGY COMMISSION

November 1973

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 2.2

DEVELOPMENT OF TECHNICAL SPECIFICATIONS FOR EXPERIMENTS IN RESEARCH REACTORS

A. INTRODUCTION

Paragraph 50.34(b)(4) of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each application for an operating license provide a final analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Section 50.36 of 10 CFR Part 50 requires that each such application also include proposed technical specifications derived from the analyses and evaluation performed for the safety analysis report (SAR).

This guide describes information that should be included in proposed technical specifications for experiments in research reactors It identifies considerations that should be addressed in the evaluation of experimental programs as well as considerations that should be addressed to define limits and other requirements to be included in the technical specifications. It is expected that the guidelines delineated here will be adapted, as required, to specific features and characteristics of individual research reactors.

B. DISCUSSION

Each safety analysis report (SAR) contains a description of the proposed experimental program and safety analyses for each type of experimental facility proposed. It includes descriptions of and safety analyses for permanently installed facilities such as beam tubes, thermal columns, hydraulic or pneumatic tube systems, and other types of capsule irradiation facilities, and movable experimental facilities (in some types of reactors) which accommodate placement of shells, tubes, trays, baskets, or other guiding or positioning devices in or adjacent to the reactor core. Safety analyses for special modes of reactor system or component use to accommodate individual, repetitive, or multiple experiments should also be provided. These can include such categories as reactor pulsing, use of reactor coolant or fuel as gamma radiation sources, or use of fuel in subcritical arrays separated from the core.

The design, construction, and placement of each experimental facility should be analyzed for inherent safety questions that exist apart from experiments accommodated therein. In addition, for each experimental facility and mode of reactor system or component use, the descriptions and safety analyses should address the types and scopes of experiments intended to be performed.

The purposes of presenting such safety analyses are (1) to demonstrate that the experimental program as envisioned at the time of presentation of the SAR can be carried out without undue risk to the public health and safety, (2) to demonstrate the technical ability to carry out the kind of safety analyses which is expected to be done on a continuing basis throughout the evolution of the experimental program, (3) to establish bases against which unreviewed safety questions can be measured pursuant to paragraph (c) of §50.59, and (4) to develop subject matter appropriate for inclusion in technical specifications.

Safety in research reactor experimentation requires that consideration be given to any feature of the design or conduct of an experiment, including intended functions and possible malfunctions, which can create, directly or indirectly, a radiological exposure hazard Safety analyses for experiments should consider (1) any

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Published guides will be revised periodically, as appropriate, to accommodate comments and to reflect new information or experience.

Copies of published guides may be obtained by request indicating the divisions desired to the U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Director of Regulatory Standards. Comments and suggestions for improvements in these guides are encouraged and should be sent to the Secretary of the Commission, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Staff.

The guides are issued in the following ten broad divisions:

1. Power Reactors
2. Research and Test Reactors
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9. Antitrust Review
10. General

interaction of an experiment with the reactor system that has the potential for breaching any primary barrier to fission product release from fuel, (2) any interaction of an experiment with the reactor system that could adversely affect any engineered safety features or control system features designed to protect the public from a fission product release, (3) any inherent feature of an experiment that could create beams, radiation fields, or unconfined radioactive materials, and (4) any potentially adverse interaction with concurrent experimental and operational activities.

A variety of specific technical factors, considered against the foregoing criteria, can give rise to safety problems as follows

1 Factors in experiments which could cause a breach in any of the fission product barriers.

a. Reactivity effects as a result of placement or removal of an experiment or of motion of material within the experiments due, for example, to forced or natural convection of fluids, phase changes, chemical or radiolytic dissociation, or mechanical instability.

b Thermal effects on fuel which alter local heat generation or heat transfer rates as a result of neutron flux perturbations, gamma heating, electrical heating, or alteration of coolant temperature or flow by experiment components or failure thereof due to heating, radiation degradation, or radiolytic dissociation.

c Mechanical forces on fuel cladding arising from the manipulation of experimental components, from tools used for such manipulation, from thermal stress, vibration, or shock waves, or from missiles arising from functioning or malfunctioning experiments.

d. Chemical attack, including corrosion, resulting from the use in or escape of materials into the fuel environment or accelerated corrosion due to elevated temperatures.

2. Factors in experiments which could adversely affect engineered safety features or control system features.

a Neutron flux perturbations affecting calibrations of safety channels and/or rod worths

b. Mechanical forces adversely affecting shielding or confinement arising from causes as in 1.c. above.

c. Radiation fields or radioactive releases from experiments which can mask the performance of an operational monitoring system intended for the detection of fission product releases at early stages.

d Physical interference by experiment components with reactor system components such as control or safety rods or physical displacement of reactor system shielding.

3 Factors in experiments which could create radiological risks due to radiation fields or unconfined radioactive material

a Use of materials which are or become chemically unstable or highly reactive or are subject to buildup of temperature or pressure, e.g., pressure buildup in special beam port plugs.

b. Irradiation of finely divided solids, liquids, or gases which are readily airborne if inadequately confined.

c. Degradation or failure of materials intended to confine experiments, e.g., by radiation decomposition of nonmetallic capsules, weld failures, gasket failures, excessive internal heat generation, or inadequate cooling.

d. Degradation or failure of vent systems or filter installations or inadequate shielding thereof.

e. Degradation or failure of safety-related instruments or control devices on experiments

f. Mechanical instability resulting in unintended movement of an experiment relative to its shielding, e.g., by faulty stacking of lead brick, by exceeding floor loading capabilities, or by capsules becoming buoyant in water.

g. Use of inadequate devices for shielding and handling experiment components or capsules following irradiations.

4. Factors relating to interactions with other experiments or with operational activities.

a. Reactivity effects of concurrent motion occurring in two or more experiments.

b. Potentially adverse interactions resulting from the use of common electric circuits and supplies and common portions of fluid systems such as manifolds for cooling water, vent, or drain systems.

c. Physical interference by experiments with patterns of operational activity which could impede or prevent a safety or emergency function, e.g., blocking of access routes.

d. Creation of industrial hazards such as the generation or release of toxic or noxious materials which could impair the ability of operators to perform necessary reactor safety functions.

e Special modes of reactor operation such as pulsing abnormal occurrences in reactor operation, or reactor accidents which could trigger failures in experiments.

The proposed technical specifications that are relevant to experiments in research reactors should (1) have bases relating to safety considerations as required by §50.36(a), (2) address subject areas that are clearly under the direct control of the licensee, and (3) fall under the categories of limiting conditions for operation, surveillance requirements, design features, or administrative controls, as specified in §50 36 Situations may arise in which the safety analyses of some unique experiments establish the need to consider the effects of such experiments on the safety limits and limiting safety system settings for reactor operation

Technical specifications should provide reasonable flexibility to perform experiments, install new experimental facilities, or change or remove from use

facilities previously described. Proposed technical specifications should address safety-oriented considerations, as distinct from functional or end-use descriptions of experimental programs. On the other hand, all safety considerations implicit in each individual experiment proposed must be enumerated and evaluated to determine whether or not they fall within the safety analysis for reactor operation presented in the SAR. In addition the proposed experiment should be evaluated in detail and its execution controlled so as to reduce any radiation dose to plant personnel and the public to the lowest practicable level.

C. REGULATORY POSITION

The safety-oriented considerations from which technical specifications for experiments should be developed include (1) the physical conditions of the design and conduct of experiments, (2) the materials content of experiments, and (3) the administrative controls employed to evaluate, authorize, and carry out experiments. The material that follows is organized according to the above three considerations, but it is not intended that this be the only format acceptable for use for proposed technical specifications. The definitions of certain terms used in this section are given in Appendix A.

1. Physical Conditions

a. Reactivity Effects

From a safety standpoint, the principal concern is that associated with a net positive reactivity effect, whether it is caused by the insertion of an experiment having a positive reactivity effect or by the removal of an experiment having a negative reactivity effect. Credit may be taken for the operation of the reactor safety system and engineered safeguards systems provided (1) they have been designed to standards and criteria establishing very high reliability, such as ANSI N42.7 (IEEE-279), (2) adequate quality assurance was provided in their construction and is provided during operation, and (3) it can be shown that they can function independently of the assumed experiment failure mode. All proposed transients should be analyzed to assure that a safety limit would not be exceeded.

(1) Every experiment should be evaluated for its static reactivity worth and its potential reactivity worth.

(2) The potential reactivity worth of each secured removable experiment should be less than that value of reactivity which, if introduced as a positive step change, could result in a transient that would be likely to lead to doses in any restricted or unrestricted area in excess of the limits set forth in 10 CFR Part 20.

(3) The magnitude of the potential reactivity worth of each unsecured experiment should be less than that value which, if introduced as a positive step change in reactivity, would cause a violation of a safety limit or of the minimum shutdown margin.

(4) The rate of change of reactivity of any unsecured experiment, any movable experiment, or any combination of such experiments introduced by intentionally setting the experiment(s) in motion relative to the reactor should not exceed the capacity of the control system to provide compensation.

(5) The sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist should not exceed the maximum value of potential reactivity worth authorized for a single secured removable experiment or the minimum shutdown margin, whichever is less.

b. Thermal-Hydraulic Effects

(1) Every experiment should be evaluated for its actual and potential thermal effects on reactor components and coolant. Normally, this evaluation should be made for the reactor at the extremes of its operating margin, as defined by limiting safety system settings.

(2) Experiments should be designed to prevent the negation of any flux peaking or reactor coolant flow considerations that have been used to define or are implicit in the safety limits for the reactor. Coolant flow considerations should include potential blockage or redistribution and potential phase changes in liquid coolant.

(3) The surface temperature of the material which bounds or supports any experiment should not exceed the lowest of the following, where applicable:

(a) the saturation temperature of liquid reactor coolant at any point of mutual contact.

(b) a temperature conservatively below that at which the corrosion rate of the boundary material at any surface would lead to its failure, or,

(c) a temperature conservatively below that at which the strength of the boundary material would be reduced to a point predictably leading to failure.

c. Mechanical Stress Effects

(1) Every experiment should be evaluated with respect to the storage and possible uncontrolled release of any mechanical energy.

(2) Experiments involving a potential for creating objects with substantial momentum (missiles) should be oriented in such a way as to minimize the probability of damage to the reactor system.

(3) Materials of construction and fabrication and assembly techniques utilized in experiments should be so specified and used that assurance is provided that no stress failure can occur at stresses twice those anticipated in the manipulation and conduct of the experiment or twice those which could occur as a result of unintended but credible changes of, or within, the experiment.

(4) Prototype testing under experiment conditions should be employed to demonstrate the ability to withstand failure

2. Material Content of Experiments

Certain kinds of materials which may be used in experiments possess properties with significant safety implications Limitations on the amounts of such materials can limit the consequences of experiment failures The material content of every experiment should be analyzed and limited according to the classifications given below.

a. Radioactive materials

(1) The radioactive material content, including fission products, of any singly encapsulated experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR Part 20 This dose limit applies to persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

(2) The radioactive material content, including fission products, of any doubly encapsulated or vented experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 rem to the whole body or 1.5 rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 rem to the whole body or 30 rem to the thyroid.

(3) For purposes of applying the above considerations, a single-mode nonviolent failure of the encapsulation boundary that releases all radioactive material into the immediate environment of the experiment or to the reactor building, as appropriate, should be assumed. The analysis should establish the most probable trajectory of the material, if any, into restricted and unrestricted areas. Credit for natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms. In addition, with respect to installed features, credit taken for their effectiveness should depend on the adequacy of the related quality assurance procedures undertaken, including the extent to which surveillance tests simulate the conditions to be met in practice If assumptions regarding atmospheric dilution are involved, they should not be less conservative than those used in the analysis of Design Basis Accidents

Irradiation of fissionable materials excluding the fissionable material content of fuel element assemblies described in the technical specifications, should be deemed an unreviewed safety question unless a specification meeting the above criteria and its related safety analysis have been approved by the Commission. With respect to other radioactive materials, specifications and safety analyses should be submitted that are representative of experiments with either the highest inventory of radioactive materials or the highest probability for failure that could result in the escape of such material into restricted and unrestricted areas. In addition, records should be generated and maintained to allow for review to demonstrate that the radioactive material content of each individual experiment does not exceed that allowed by the stated criteria.

These considerations should not be interpreted (1) to permit or encourage any unnecessary intentional releases of radioactive materials to unrestricted areas, or (2) to relieve the obligation to minimize and control radiation doses in restricted areas

b. Trace Elements and Impurities

A reasonable effort should be made to identify in advance of an experiment trace elements or impurities whose activation products may represent the dominant radiological hazard.

c. High-Cross-Section Materials

Nuclides possessing high thermal neutron absorption cross sections should be identified and limited with respect to their quantity or method of inclusion in individual experiments in order to control reactivity or thermal effects within the limitations specified.

d. Highly Reactive Chemicals

The inclusion of explosive materials in experiments constitutes an unreviewed safety question unless such usage has been reviewed and approved by the Commission, except that amounts up to 25 milligrams of TNT equivalent may be irradiated or stored inside the reactor confinement system in accordance with regulatory position C.1.c.

e. Corrosive Chemicals

A list should be prepared identifying materials which are chemically incompatible with the reactor system from the viewpoint of corrosion and which should be excluded from any experiments or the use of which is subject to special scrutiny and control. This list should be provided to all who use the reactor.

f. Radiation-Sensitive Materials

The evaluation of each experiment should include an assessment of the consequences of physical or

chemical changes in the material content as a result of its presence in a radiation environment, particularly for nonmetallic materials

Effects to be considered include the alteration or degradation of mechanical properties due to radiation-induced decomposition, e.g., of plastics or polymers, and radiolytic generation of excessive gas pressure or explosive gas mixtures.

g. Flammable or Toxic Materials

Procedures control should incorporate mechanisms for handling and limiting the quantities of highly flammable or toxic materials used in experimental programs or used in the reactor room.

h. Cryogenic Liquids

The inclusion of cryogenic liquids within the biological shield of a research reactor would constitute an unreviewed safety question unless such usage has been reviewed and approved by the Commission.

i. Unknown Materials

No experiments should be performed unless the material content, with the exception of trace constituents, is known.

3. Administrative Controls of Experiments

a. Internal Authorization

(1) Evaluation by Safety Review Group

(a) No experiment should be performed without review and approval by a technically competent Safety Review Group or Committee. Repetitive experiments with safety considerations in common may be reviewed and approved as a class.

(b) Criteria for review of an experiment or class of experiments should include (1) applicable regulatory criteria, including those in 10 CFR Part 20 and the technical specifications and (2) in-house safety criteria and rules which have been established for facility operations, including those which govern requirements for encapsulation, venting, filtration, shielding, and similar experiment design considerations, as well as those which govern the quality assurance program required under § 50.34.

(c) Records should be kept of the Safety Review Group's review and authorization for each experiment or class of experiments.

(2) Operations Approval

(a) Every experiment should have the prior explicit written approval of the Licensed Senior Operator in charge of reactor operations.

(b) Every person who is to carry out an experiment should be certified by the Licensed Senior Operator in charge of reactor operations as to the sufficiency of his knowledge and training in procedures required for the safe conduct of the experiment

b. Procedures for Active Conduct of Experiments

(1) Detailed written procedures should be provided for the use or operation of each experimental facility.

(2) The Licensed Operator at the console should be notified just prior to moving any experiment within the reactor area and should authorize such movement.

(3) Each experiment removed from the reactor or reactor system should be subject to a radiation monitoring procedure which anticipates exposure rates greater than those predicted. The results of such monitoring should be documented.

c. Procedures Relating to Personnel Access to Experiments

(1) There should be a documented procedure for the control of visitor access to the reactor area to minimize the likelihood of unnecessary exposure to radiation as a result of experimental activities and to minimize the possibility of intentional or unintentional obstruction of safety.

(2) There should be a written training procedure for the purpose of qualifying experimenters in the reactor and safety-related aspects of their activities, including their expected responses to alarms.

d. Quality Assurance Program

There should be a Quality Assurance Program covering the design, fabrication, and testing of experiments, including procedures for verification of kinds and amounts of their material contents such as those described in regulatory position C 2.

APPENDIX A

DEFINITIONS

1. **Experiment**--An experiment, as used herein, is any of the following
   a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein,
   b. An evaluation or test of a reactor system operational, surveillance, or maintenance technique;
   c. An experimental or testing activity which is conducted within the confinement or containment system of the reactor, or
   d. The material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

2. **Experimental Facility**--An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

3. **Explosive Material**--Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Third Ed. (1968), or is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M 1966, "Identification System for Fire Hazards of Materials," also enumerated in the "Handbook for Laboratory Safety" 2nd Ed. (1971) published by The Chemical Rubber Co.

4. **Movable Experiment**--A movable experiment is one which may be inserted, removed, or manipulated while the reactor is critical.

5. **Potential Reactivity Worth**--The potential reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

   The evaluation must consider possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes such as creating or filling of void spaces or motion of mechanical components. For removable experiments, the potential reactivity worth is equal to or greater than the static reactivity worth.

6. **Removable Experiment**--A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

7. **Secured Experiment**--Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

8. **Static Reactivity Worth**--As used herein, the static reactivity worth of an experiment is the absolute value of the reactivity change which is measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment. For removable experiments, the terminal positions are fully removed from the reactor and fully inserted or installed in the normal functioning or intended position.

9. **Unsecured Experiment**--Any experiment, experimental facility, or component of an experiment is deemed to be unsecured if it is not and when it is not secured as defined in 7. above. Moving parts of experiments are deemed to be unsecured when they are in motion.

Appendix 10.2

Regulatory Guide 2.4

Review of Experiments for Research Reactors

U.S. NUCLEAR REGULATORY COMMISSION July 1976

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 2.4

REVIEW OF EXPERIMENTS FOR RESEARCH REACTORS

A. INTRODUCTION

Section 50.36, "Technical Specifications," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each applicant for a license authorizing operation of a production or utilization facility include in his application proposed technical specifications. If acceptable, these technical specifications, along with any other such specifications that the Commission finds appropriate, are incorporated into the facility license that is issued by the Commission and are conditions of the license.

Paragraph (c)(5), "Administrative controls," of § 50.36 of 10 CFR Part 50 requires that technical specifications for nuclear reactors include provisions relating to the organization and management procedures, recordkeeping, review and audit, and reporting necessary to ensure operation of the facility in a safe manner. Section 50.59, "Changes, tests and experiments," of 10 CFR Part 50 permits each holder of a license authorizing operation of a production or utilization facility to make changes in the facility and procedures as described in the safety analysis report (SAR) and to conduct tests or experiments not described in the SAR, without prior Commission approval, unless the proposed change, test, or experiment involves a change in the technical specification incorporated in the license or an unreviewed safety question.

This guide describes procedures acceptable to the NRC staff for the licensee's review and approval of experiments performed at research reactor facilities.

B. DISCUSSION

Standard ANSI N401-1974 (ANS-15.6), "Review of Experiments for Research Reactors,"* was prepared by Work Group ANS-15.6 and sponsored by

*Copies may be obtained from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521.

USNRC REGULATORY GUIDES

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Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. However, comments on this guide, if received within about two months after its issuance, will be particularly useful in evaluating the need for an early revision.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Section.

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Subcommittee ANS-15 (Research Reactors) of the American Nuclear Society (ANS). This standard was approved by the American National Standards Committee N17, Research Reactors, Reactor Physics and Radiation Shielding, and its Secretariat in March 1974. It was subsequently approved and designated ANSI N401-1974 by the American National Standards Institute (ANSI) on November 19, 1974. The standard provides guidance for the licensee's review and approval of experiments performed at research reactor facilities by identifying substantive areas for each experiment that should be reviewed to provide assurance that the experiment (1) falls within the limits delineated in the technical specifications, (2) does not present an unreviewed safety question as defined in §50.59, "Changes, tests and experiments," of 10 CFR Part 50, (3) does not constitute a threat to the health and safety of any individuals, and (4) does not constitute a hazard to the reactor facility or other equipment. In addition, this standard recommends a system for classifying experiments to establish levels of licensee review and approval commensurate with the level of risk inherent in the experiment. Both the requirements and the recommendations of the standard have been evaluated by the staff in evaluating the acceptability of this standard.

C. REGULATORY POSITION

The requirements and recommendations provided in ANSI N401-1974, "Review of Experiments for Research Reactors," are generally acceptable to the NRC staff. The guidance provides an adequate basis for the review and approval of research reactor experiments performed in accordance with §§50.36 and 50.59 of 10 CFR Part 50, subject to the following:

1. The last sentence of the paragraph defining, "shall, should and may," as given in Section 3, "Definitions," of ANSI N401-1974 should be modified to read as follows: "To conform to this standard, experiment review shall be performed in accordance with the standard's requirements and recommendations."

2. The definition of non-secured experiment, as given in Section 3, "Definitions," should be modified to read as follows: "Any experiment, experimental facility, or component of an experiment is considered to be unsecured when it is not secured as defined under secured experiment in Section 3."

3. Subsection 4.1, "Classification System," should be modified by adding the following sentence: "The experiment classification system to determine level of approval for the experiments should be reviewed and approved by the Reactor Safety Committee designated in the Technical Specifications."

4. In addition to the experiment plan (Section 5, "The Experimental Plan"), there should also exist detailed procedures for carrying out an

experiment, and these procedures should be reviewed as required by the facility's technical specifications. A single experimental procedure may be used for more than one exposure or more than one identical experiment, but such a procedure should expire after a specified interval.

5. Subsection 6.1, "Review Procedure," should be modified by adding the following sentence: "The experiment review procedure should be reviewed and approved by the Reactor Safety Committee designated in the Technical Specifications."

6. Paragraph (3) of Subsection 6.2, "Considerations," should be replaced with the following: "Does the experiment meet all criteria regarding reactivity effects? These criteria include assurance that (1) the potential reactivity worth of each secured experiment would be less than that value of reactivity which, if introduced as a positive step change, could result in a transient that would be likely to lead to doses in any restricted or unrestricted area in excess of the limits set forth in 10 CFR Part 20; (2) the magnitude of the potential reactivity worth of each non-secured experiment would be less than that value which, if introduced as a positive step change in reactivity, would cause a violation of a safety limit or of the minimum shutdown margin; and (3) the rate of change and magnitude of reactivity of any moveable experiment, moveable parts of experiments, or any combination of such experiments introduced by intentionally setting the experiments in motion relative to the reactor would not exceed the capacity of the control system to provide compensation."

7. In Subsection 6.3, "Review Personnel," the last paragraph should be replaced with the following: "Members of the Committee should disqualify themselves from the review of experiments in which they are directly involved. They may act as consultants to the review group but should not be involved with the final decision for approval or disapproval of the experiment."

8. The specific applicability or acceptability of items 1, 3, 4, and 5 of Section 9, "References," of ANSI N401-1974 will be covered separately in other regulatory guides, where appropriate.

D. IMPLEMENTATION

The purpose of this section is to provide guidance to applicants regarding the NRC staff's plans for using this regulatory guide in the review of research reactor facility applications. The staff will use this guide in evaluating applications submitted after the date shown below. However, all or part of this guide may be used by the staff to the extent reasonable and practicable for evaluating prior applications. Such use is usually reflected in the staff review questions and subsequent evaluations

for specific cases. Backfitting action, if required, will be considered separately pursuant to Section 50.109 of 10 CFR Part 50.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in evaluating applications in connection with research reactor facility construction permits, operating licenses, or proposed amendments thereto submitted for approval after March 1, 1977.

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11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

In this chapter of the SAR, the applicant should discuss and analyze all radiological consequences related to normal operation of the reactor. In general, the design and function of structures, systems, and components and all facility operations and materials authorized by the operating license should be described in detail in other chapters of the SAR. Chapter 11 should contain the principal discussions of the facility program to control radiation and expected exposures due to operation, maintenance, and use of the reactor. In this chapter, the applicant should develop the methods for quantitative assessment of radiation doses in the restricted, controlled (if present), and unrestricted areas; should apply those methods to all applicable radiation sources related to the full range of operation; should describe the program and provisions for protecting the health and safety of the public (including workers) and the environment; and should provide the bases for analyzing radiological consequences from potential accidents addressed in detail in Chapter 13, "Accident Analyses."

In accordance with 10 CFR 20.1101, it is the responsibility of the applicant to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the regulations in 10 CFR Part 20. To the extent practicable, the applicant should also use procedures and engineering controls based on sound principles of radiation protection to keep doses to occupational workers and members of the public as low as is reasonably achievable (ALARA).

Waste materials resulting from maintenance, normal operations, or accident conditions at non-power reactors may contain radioactive isotopes. Such wastes are governed by the operating license, and, like other licensed materials, they must be controlled. At a non-power reactor, management and control responsibility for radioactive waste may be assigned to the organization responsible for reactor operations, and the radiation protection organization may provide independent oversight for monitoring, assessing, and limiting risks related to radiation sources. Alternatively, facility management could assign primary responsibility for handling and disposing of radioactive wastes to the organization responsible for radiation protection. In either case, the applicant should require procedures to ensure that radiation exposures and releases of radioactive material are adequately assessed and controlled. The applicant should discuss these issues and submit the information necessary for NRC review. This format and content guide for Chapter 11 of the SAR integrates radioactive waste management and radiological protection in some sections, and provides separate sections for some information. The applicant should organize the functions and present the information as best suits the facility consistent with this guide.

11.1 Radiation Protection

The sections that follow provide guidance on the information the applicant should include in the description of the radiation protection program. The program is applied to the design of the reactor and its equipment, the experimental facilities, reactor operations, design and use of associated laboratories, planning and procedures, and the instrumentation, techniques, and practices employed to verify compliance with the radiation dose limits and other applicable requirements specified in the regulations. Plans and the bases used to develop procedures for assessing and controlling radioactive wastes and the ALARA program should be included. The responsibilities of the health physics organization at the reactor facility, as well as any other applicant radiation protection organizations (e.g., under a separate materials license), should be described. Facility organization charts should be included that show independence of the radiation protection function from the facility operations function.

In this chapter, the applicant should address all radiation sources and radioactive materials produced in the reactor and possessed or used within the reactor facility under the authorization of the reactor license. Other byproduct, special nuclear material (SNM), and source material possessed or used under the authorization of the reactor license but not produced by reactor operation should be described. Program details should be given in the sections that follow.

11.1.1 Radiation Sources

In this section, the applicant should describe the sources of radiation that are monitored and controlled by the radiation protection and radioactive waste programs. In general, the sources should be categorized as airborne, liquid, or solid as discussed in the sections that follow.

The applicant should include in this description a tabulation of all standard, check, and startup sources categorized by isotopic composition, principal radiations (e.g., beta and gamma ray energies, abundance > 10%), activity (curie content), neutron characteristics, geometry, physical and chemical form, and whether the source is sealed or unsealed.

The applicant should also tabulate all fissile and fissionable materials, including fuel elements and assemblies, showing the status (fresh, in-core, interim storage, or spent), original enrichment [including uranium-235 (U-235) and total uranium (U) content], and current enrichment [including current U-235, total U, and total plutonium (Pu) (if appropriate)].

Because of the varied nature of experimental programs, the source strengths of irradiated experimental materials are not necessarily tabulated in an SAR.

However, the full range of source strengths expected to be encountered in the experimental program should be listed and discussed. Experimental protocols should provide detailed source data and should be subject to the review of facility operations staff, the health physicist, and, in the case of new experiments and specified deviations from previous experiments, the reactor review or audit committee. In evaluating all experiments, the applicant shall also consider the requirements of 10 CFR 50.59.

Conservative estimates should be made of the quantities and types of radioactive wastes expected to result from reactor operations and use, based on previous or other similar reactor facility experience. Identification of such wastes should distinguish, if possible, which are associated with the operation of the reactor and which are associated with the utilization of the reactor, if utilization occurs under the reactor license. Non-power reactor applicants have a tendency to provide overly conservative estimates; although estimates should be conservative, they should also be realistic.

Where feasible, the applicant should include the physical and chemical form, amounts, use, storage conditions, and locations of all sources. In occupied or accessible areas, conservative estimates of external radiation fields should be given. An estimate of the maximum annual dose and collective doses to workers and the public should be given for major and repetitive activities involving radiation. The applicant should discuss how the requirements of Subpart C of 10 CFR Part 20 (20.1201–20.1208), which contains regulations for occupational dose limits, and Subpart D of 10 CFR Part 20 (20.1301–20.1302), which contains regulations for radiation dose limits for individual members of the public, will be met. Regulations concerning compliance with dose limits for individual members of the public are given in 10 CFR 20.1302. Applicants that have licensed non-power reactors usually have historical information on radiation doses. They should discuss this information.

License conditions and, if applicable, technical specifications, concerning material possession limits, enrichment, material forms, and source strengths, should be developed and analyzed in this and other chapters, such as Chapter 4, "Reactor Description," of the SAR. These will control the use of the sources discussed above.

11.1.1.1 Airborne Radiation Sources

Airborne radioactive sources should be described in a manner suitable for designing worker protective measures and assessing and controlling workers' doses. Airborne radionuclides are important because they typically are the principal source of radiation exposure to the public from a non-power reactor. In a table, the applicant should summarize the predicted concentrations and quantities

of airborne radionuclides during the full range of normal operation (which includes maintenance activities) according to the areas that could be occupied by personnel. The applicant should estimate the release of airborne radionuclides to the environment and should use these releases to determine consequences in the offsite environment. The applicant should discuss compliance with the applicable regulations (10 CFR Part 20). Note that while airborne radioactive sources from accidents are discussed in Chapter 13, the calculational methodologies developed here should be applicable to accident release analysis. Therefore, the models and assumptions used for the prediction and calculation of the dose rates and accumulative doses in both the restricted, controlled (if present), and unrestricted areas should be provided in detail. The guidance that follows gives an example of a description of appropriate methodology as illustrated for argon-41 (Ar-41) that is also applicable to any airborne radionuclide, provided both internal and external dose delivery are accounted for.

The potential for argon-41 production exists at most non-power reactor facilities over the full range of normal operations, and argon-41 could be the predominant radionuclide released to the unrestricted area. Argon-41 is produced when the Ar-40 in air and air in solution in water is activated by neutrons. Argon-41 may be considered a radioactive waste produced by reactor operations. The specific source locations (e.g., primary coolant water, beam tubes, exposure rooms, and air-driven rabbit systems), predicted production rates, release mechanisms and rates, concentrations in occupied areas, possible personnel doses and dose rates, release points from the restricted area, dilution air (quantities and sources), quantities and concentrations predicted to be released, annual average atmospheric conditions, diffusion and dispersion, predicted concentrations in unrestricted areas, and potential dose rates and annual doses, including gamma-ray shine from elevated plumes, should be addressed in detail.

For argon-41, as well as other noble gases at non-power reactors, it is acceptable to assume that all significant radiation risk is from external exposure to beta and gamma radiation. Other radionuclides (e.g., halogens or particulates) could cause internal radiation risk by being ingested or inhaled. All these doses should be addressed, as applicable. The assumptions and methods should be conservative but physically realistic, and the validity of dose calculations should be assessed. Some non-power reactor applicants have used conservative assumptions and methods that have resulted in answers that, although acceptable, are conservative by large factors. The applicant should consider discussing the amount of conservatism built into the calculations. All assumptions should be justified, and sources of information should be adequately referenced. The calculations should address possible doses in the restricted areas, in the controlled areas (if applicable), and in the unrestricted areas. In the unrestricted area, potential doses should be analyzed for the maximally exposed individual, at the location of the nearest permanent residence, and at any locations of special interest, such as a classroom

or a campus dormitory. Due care should be taken if finite or non-uniform airborne distributions are intermingled with infinite cloud approximations within buildings or in idealized gaussian plumes. Any such intermingling of models or assumptions should be justified. Similar discussions in this paragraph of the SAR should address the production of airborne particulates, aerosols, vapors, and nitrogen-16 (N-16) or other radionuclides.

The discussion and calculations should show how the facility design ensures that doses to the facility staff and the public will not exceed 10 CFR Part 20 limits and that its ALARA requirements for effluents are satisfied.

11.1.1.2 Liquid Radioactive Sources

The applicant should identify all expected liquid radioactive sources, such as reactor primary coolant, experimental solutions, reference sources, and fissile material. The applicant should identify their origin and should specify whether they result from reactor operations or the utilization program or whether they exist for special purposes. Information should include radionuclides, concentrations, total curie strength, solubility, container characteristics, and planned release or disposition. Liquid radioactive wastes should be included. However, since the types of such wastes, their origins, and the source strengths will vary with time and with the nature of the utilization program, only limited descriptions of liquid wastes should be provided. The applicant should estimate the quantity of liquid effluent released to the unrestricted environment. The applicant should discuss if credit is taken for dilution preceding release. The applicant should discuss compliance with the applicable sections of 10 CFR Part 20, such as 10 CFR 20.2003 and any disposal of licensed material approved under 10 CFR 20.2002. Any storage or disposal facilities should be noted, with reference to their management and use and the design basis of their radiation protection capabilities.

11.1.1.3 Solid Radioactive Sources

The applicant should identify all expected solid radioactive sources, such as reactor fuel (spent, in-core, and fresh), calibration and test sources, experiment samples, and facility components. The information should include, among other things, radionuclides, curie strengths, and physical characteristics and whether the source is sealed or unsealed. Solid radioactive waste should be noted, but because the types and quantities will vary with time and the utilization program, only limited descriptions of solid wastes need be provided. Provisions for classifying, monitoring, storing, packaging, volume reduction prior to shipment, and disposing of solid radioactive wastes should be discussed. The applicant should estimate the annual volume of solid waste expected to be removed from the site and its radioactive content (in curies). The applicant should discuss compliance with

applicable sections of 10 CFR Parts 20, 61, and 71, and Department of Transportation regulations (49 CFR) for transporting radioactive material.

The applicant should discuss any capabilities or approvals received under NRC or State material licenses for onsite or offsite storage of solid radioactive wastes, including how the necessary characteristics of a restricted area are maintained. The applicant should discuss any disposal of licensed material approved under 10 CFR 20.2002.

This section should contain the design bases for temporary, permanent, and installed shielding components at the facility, including utilization, laboratory facilities, and radiation beams.

The following areas of the facility should be examined when developing the program for inventory and control of radiation sources:

- the exterior of the reactor biological shielding and reactor auxiliary locations (e.g., primary coolant system components and demineralizers) accessible to personnel
- the reactor experimental facilities, including beam ports, thermal columns, pneumatic or hydraulic transfer facilities, and all other irradiation facilities
- the radioactive material handling, preparation, packaging, and utilization facilities, including laboratories, hot cells, caves, and storage and processing areas
- other extraneous sources, including, for example, neutron and gamma irradiation facilities, check and standard sources, neutron sources, fuel handling and storage facilities, experimental equipment storage facilities, and radioactive waste handling and storage facilities

11.1.2 Radiation Protection Program

In this section, the applicant should describe the structure of the organization that administers the radiation protection program required by 10 CFR 20.1101, including information about staffing levels, positions of authority and responsibility, and position qualifications. Working relationships with other safety organizations, including the reactor facility operations staff, should be described. The applicant should discuss the charters, standards, procedures, and other documents that specify the authority and responsibilities of the organization, including authority to interdict perceived unsafe practices. The administrative plans and procedures that implement the facility policy, the overall program, and the way the organization, policy, and program are designed for effective operation

should be discussed. In this discussion, the applicant should describe the management policy governing the program and the allocation of policymaking responsibilities. Reference can be made to Chapter 12, "Conduct of Operations," if such information appears there.

The information should include the document control measures employed to ensure that the plans and procedures relative to the radiation protection program, including changes, are reviewed for adequacy, approved by authorized personnel, and distributed to and used by the applicable staff at the locations where radiation exposures could be encountered.

The radiation safety training program should be described in detail. This discussion should give the scope, and a summary of the content, of the training provided or required for all personnel, including facility-employed personnel, health physics personnel, non-facility-employed research and service personnel, visitors, and security, fire, and other emergency personnel.

The applicant should describe the purpose, organization, and functions of any review and audit committees with responsibilities relating to radiation safety, including the charter, frequency of meetings, audits, scope of any reviews, and qualifications and requirements for committee members. The applicant should describe how each committee's work relates to the radiation safety organization and how a comprehensive program is ensured. If this information is discussed in Chapter 12, it can be referenced here.

The program for conducting facility radiation safety audits of all functional elements of the radiation protection program to meet the requirements of 10 CFR 20.1101(c) should be described, identifying the scope of the audits, the bases for scheduling the audits, the qualifications of the auditors, the management level to which reports are sent, and the process for following up on audit findings. The relationship of this program to any other self-assessment/internal appraisal program should be discussed. The bases for technical specifications related to facility radiation safety audits should be provided.

The system that examines the experiences of the radiation protection program and uses these experiences to improve the program and the facility design for radiation protection should be described. This system should also examine problem and incidents and develop "lessons learned," root causes, and effective corrective actions.

For activities not described in the SAR or not governed by procedures, a work control process such as the use of radiation work permits should be used. The applicant should discuss the control program used at the facility.

The applicant should describe the recordkeeping process for the radiation safety program, including record-retention periods, accessibility, review, and archiving. Review of radiation safety records for accuracy and validity should be discussed. The use of records for developing trend analyses, informing management, planning radiation-related actions, and reporting to regulatory and other duly authorized entities should be discussed.

11.1.3 ALARA Program

In this section, the applicant should describe the ALARA program for the facility required by 10 CFR 20.1101. The description should include the basis for the program and the management level and authority by which the facility ALARA policy is established. The applicant should discuss how this program is implemented to maintain radiological doses of all personnel at the facility and releases of effluents to the unrestricted area ALARA. The applicant should discuss the criteria used to determine how low the projected doses should be to permit task implementation (i.e., ALARA goals). The discussion should include methods to ensure that the radiation protection staff, with their considerations of the facility ALARA program, are specifically involved during review and approval of design, in construction of facilities, in the planning and implementing of reactor utilization (experiment design and planning) and operation, in maintenance activities, and in the management and disposition of radioactive wastes.

11.1.4 Radiation Monitoring and Surveying

The program employed to routinely monitor workplaces and other locations accessible to people for identification and control of sources of radiation exposure should be described in this section, including the measures designed to ensure that air, liquids, and solids are monitored in all applicable areas. The applicant should also discuss the bases of the methods and procedures used for detecting and assessing contaminated areas, materials, and components, and should describe the records that document the applicability, quality, and accuracy of monitoring methods, techniques, and procedures.

The applicant should provide summary descriptions of all radiation monitoring equipment employed throughout the facility, including locations and functions of each device and system. This summary should also describe sampling equipment for liquid and gaseous process and effluent streams. This discussion may be combined with (and appropriately cross-referenced to) the discussions in Chapter 7, "Instrumentation and Control Systems." The applicant should discuss the interface between the radiation monitoring system and engineered safety features discussed in Chapter 6, "Engineered Safety Features," if any exist. Types of equipment should include systems of the following types (as appropriate to the facility):

- continuous air monitors (CAMs), including fixed and moving filter, and gaseous monitors
- portable survey instruments (radiation fields and contamination)
- remote area monitors (RAMs)
- samplers
- effluent monitors
- environmental monitors (details should appear in Section 11.1.7)
- personal dosimeters
- portal monitors
- radwaste storage monitors
- criticality monitors

The calibration of radiation protection instrumentation, including the procedures and standards governing calibration, control of the calibration process, use of national standards, and verification should be described. In this section, the applicant should also describe the calibration equipment and discuss sensitivities to environmental and other conditions with respect to the calibration requirements. The program to ensure that routine periodic calibration is performed in a timely manner and the bases of calibration schedules should be described.

The applicant should describe how routine monitoring provided at the facility is planned to ensure that radiation exposures to the public and workers or material releases can be detected, and should discuss how the approach used for routine monitoring provides reasonable assurance that all radiation at, and released from, the site will be appropriately monitored.

Technical specifications and their bases related to the radiation monitoring equipment and procedures, as discussed in Chapter 14, "Technical Specifications," should be justified in this section.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure is controlled by controlling radioactive materials and effluent radioactive material releases. In this section of the SAR, the applicant should describe the design bases for the equipment and procedures utilized for controlling

exposures to personnel and releases of radioactive materials from the facility, and should discuss how the facility structures, systems, and components are designed to provide assurance that there will be no uncontrolled effluent radioactive releases to the environment or to work areas. Some systems, such as containment, confinement, and ventilation, may have been discussed in other chapters of the SAR; reference to those discussions in this chapter of the SAR is appropriate. The applicant should also discuss how the bases of radiation shielding, ventilation, and remote handling and decontamination equipment are designed to ensure that doses to the workers are maintained ALARA and within the applicable regulatory limits.

How the design of required entry control devices (i.e., alarms, signals, or locked entry ways) alerts workers to, or prevents entry into, high radiation and very high radiation areas should be described. The regulations in 10 CFR Part 20, Subpart G, "Control of Exposure From External Sources in Restricted Areas," contains requirements for control of access to high and very high radiation areas. It should be noted that 10 CFR 20.1601(c) allows a licensee to apply to the Commission for approval of alternative methods for controlling access to high radiation areas if the licensee finds that the stated methods of control in the regulations would interfere with utilization programs. The application should contain a description of the proposed method along with a discussion of how the entrance or access point to high radiation areas will be controlled.

Equipment and materials (e.g., anti-contamination clothing and respiratory protection equipment) to protect personnel employed in the facility should be discussed. The applicant should describe the facility conditions for which this personnel protective equipment should be employed, and should also discuss whether respirators should be used at the facility. The use of respiratory protection equipment requires implementing and maintaining a respiratory protection program in accordance with the requirements of 10 CFR 20, Subpart H. If a respiratory protection program will be maintained, that program should be described as it relates to the minimum program requirements of 10 CFR 20.1703.

The bases and values for the expected annual radiation exposure for all locations of the facility should be discussed, including the exposure estimates for applicant-employed personnel, non-applicant-employed research and service personnel, and visitors. This discussion should include the exposure limits and controls for such groups as embryos, fetuses, declared pregnant women, minors, and students. The plans and procedures for exposure control and dosimetry during the full range of normal facility operations, potential accident conditions, rescue and recovery, and planned special personnel exposures (non-emergency) should also be discussed. The applicant should describe the dosimetry used for assessing external radiation exposures (e.g., whole body, extremities), including the frequency of dosimeter readings, administrative dose action levels, and the suitability of the dosimetry chosen with respect to the radiation sources anticipated and observed. The same

factors for how internal exposures and doses are assessed, evaluated, and controlled should be described.

The applicant should describe the type of records retained to document the conditions under which individuals were exposed to radiation. The applicant should discuss the historical and current exposures to personnel and the associated trends.

11.1.6 Contamination Control

The applicant should discuss the plans and bases of procedures for identifying and controlling radioactive contamination, including methods established to assess the effectiveness of the contamination control program. The discussion should include information on the following topics, showing their relationship to regulatory requirements and ALARA concepts:

- a program for routine monitoring to detect and identify fixed and loose contamination
- programs to control access to contaminated areas, avoid further spread of contamination, and remedy contaminated areas
- personal monitoring and assessment of internal and external doses to personnel occupying or entering contaminated areas, and methods for appropriate surveying and "frisking" upon exit
- use of anti-contamination techniques to protect workers, and control and disposition of possibly contaminated clothing and materials
- procedures for monitoring and handling contaminated equipment and components outside of contaminated areas that have not been decontaminated
- criteria for classifying contaminated material, equipment, and working areas, and managing, controlling, storing, and disposing of identified contamination
- training programs for staff and visitors on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination
- recordkeeping for contamination events, both for personnel and for locations, including records to be available for facility maintenance and for eventual decommissioning

- the bases of technical specifications, if needed, applicable to contamination control: for example, limits on storage and handling of radioactive sources, especially unsealed ones; limitations on encapsulation of irradiated materials; and use of fume hoods and hot-waste drains

11.1.7 Environmental Monitoring

The applicant should describe the environmental monitoring program, including information relating to the following:

- verification of compliance with commitments made in environmental reports, or other documents, if applicable; discussion of any standards used in the environmental monitoring program

- for established programs, evaluation of the effectiveness of the program

- identification of potential facility impacts on the environment and the evaluation of the need for remedial action or mitigation measures

- establishment of baselines for environmental quality, including data comparing preconstruction or preoperational with operational environmental monitoring results

The applicant should describe the written plans and the bases of procedures for implementing the environmental monitoring program, and should discuss the document control measures employed to ensure that the plans and procedures, including changes, are reviewed for adequacy and approved by authorized personnel, and are distributed to and used at the appropriate locations throughout the facility.

The environmental surveillance program and its bases should be described. Air, water, and land environments should be specifically discussed. These discussions should include information on at least the following topics:

- probable facility-related contaminants and pathways to people

- selection of sampling materials and locations

- sample collection methods and frequency of collection

- sample analyses (analytical techniques) and sensitivities (detection limits)

- records of results and trends

11.2 Radioactive Waste Management

Each facility that is licensed to operate or utilize a non-power reactor should establish a program and procedures that are designed to ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner to protect the health and safety of the public and the environment. The magnitude and nature of the effort required should depend upon the size and complexity of both the reactor facility and its utilization programs. Therefore, the nature and details of the radioactive waste management program should also be commensurate with those factors. As noted previously, management of radioactive wastes could be an auxiliary function assigned to existing personnel, such as people engaged in radiation protection or operations. Earlier sections of this chapter have addressed the program and procedures for controlling and assessing radiation exposures and doses at the facility due to all radiation and radioactive sources. In this section, the applicant should address the program and procedures for further managing sources classified as radioactive waste.

11.2.1 Radioactive Waste Management Program

In this section, the applicant should discuss the philosophy and objectives of the program for managing radioactive waste. The applicant should describe the organizational structure within which it will administer the reactor-related radioactive waste management program, including the organization and staffing levels, authorities and responsibilities, and position qualifications. The working relationships between such facility organizations as radiation protection and operations staff, and the standards, charters, procedures, or other documents that specify the authority, duties, and responsibilities of the personnel in the radioactive waste management organization should be discussed. The policy governing the program, the allocation of policymaking responsibilities, and the administrative plans and procedures that implement the facility policy should be described. The overall program and how the organization, policy, and program lead to effective management of radioactive waste should be evaluated and described.

The applicant should describe the purpose, organization, and functions of any committees assigned responsibility for overseeing radioactive waste management. The description should include each committee's charter, responsibilities, frequency of meetings, audit and review responsibilities, scope of any audits or reviews, and qualifications and requirements for committee members. How each committee's work relates to the waste management organization and how they work together should be discussed. If this information has already been described, reference that discussion.

The applicant should describe the waste management training program. This description should include the scope of facility waste management training, as well as specific training requirements for personnel associated with the operation and use of the facility.

The applicant should describe the document control measures that ensure that the plans and procedures involving radioactive waste, including changes, are reviewed for applicability, approved by authorized personnel, and distributed to and used at the locations where waste management activities are conducted.

The applicant should describe the scope of waste management reviews and audits. This description should include the authority of waste management review and audit teams, the objectives and purposes of reviews and audits, and the bases for scheduling these reviews and audits.

The applicant should describe the radioactive waste management recordkeeping process, including retention periods, accessibility, review, and archiving, and should discuss any special review of waste management records for accuracy and validity. Records of radioactive wastes stored for the life of the facility or buried on site should be discussed, as well as records for trend analysis.

The bases for any technical specifications related to the radioactive waste management program should be described.

11.2.2 Radioactive Waste Controls

The applicant should discuss the definition of radioactive waste, the point in any process that a radioactive component or material becomes classified as waste, and the criteria for defining such waste. The applicant should describe the waste management program procedures which ensure that radioactive wastes are identified and characterized appropriately, as noted above, and the bases of the procedures which ensure that radioactive wastes are adequately segregated from nonradioactive wastes. The plans and procedures for managing all forms of radioactive wastes generated during operations, research, and utilization of the reactor should be described. Since radioactive wastes are radiation sources, they should be described, along with other such sources, in Section 11.1 of the SAR.

The applicant should describe the plans and bases for procedures for managing gaseous and other airborne radioactive wastes generated during operations, research, and utilization of the reactor, and radioactive waste off-gas collection systems designed to be utilized at the facility. The function and the location of each off-gas collection system should be described. At many non-power reactors, the system for removing gaseous radioactive waste is integral to the ventilation system for the facility and may have engineered safety functions. If these systems

have been described in other chapters of the SAR, reference may be made here to those discussions. For all off-gas and ventilation systems, the applicant should describe the wastes produced by operation of the systems. Such items as filters and scrubbers, which collect and concentrate wastes, should be discussed to indicate the disposition of the radioactive material upon regeneration or replacement. If the radioactive materials enter other waste treatment systems, the applicant should indicate how such transfers are made and note any possible chemical or radiological effects of the transfer. The operation of any gas-cleaning equipment and its designed performance should be discussed in this section. The bases of any applicable technical specifications that control these functions should be given. Also, the applicant should describe all secondary radioactive residues that are generated during process treatment, their chemical and physical composition, and the modes for handling, controlling, and storing them.

The applicant should describe how liquid radioactive wastes are generated and where they enter the waste control and treatment systems. Such items as laboratory wastes, liquid spills, and cleanup solutions, including detergent wastes, should be discussed. Information about the projected inventory levels, interim and long-term storage, and processing of those streams to achieve volume reduction or solidification should be included. This discussion should include information about coolant cleanup systems and resin regeneration solutions and wastes, if applicable.

The objectives of the processes designed to treat radioactive or mixed liquid wastes should be described. Any backup and special safety features designed to ensure that the radioactive waste is contained during treatment should be described. The designed equipment and systems should be described, and appropriate engineering drawings to show the location of the equipment, flow paths, piping, valves, instrumentation, and other physical features, should be included, along with information on all features, systems, or special handling techniques that prevent uncontrolled releases or personnel exposures.

The applicant should describe the plans and procedures for managing solid radioactive wastes generated during operations, research, and utilization of the reactor. This description should include how solid radioactive materials are generated and where they enter the waste control and treatment systems. For solid radioactive wastes retained or stored on site for the life of the facility, the applicant should discuss the control methods used. Integrity and corrosion characteristics and the monitoring of the containment should be discussed, as well as the plan for disposing of these radioactive wastes when the facility is permanently decommissioned.

The applicant should describe the systems and equipment selected for identifying, segregating, and safely managing the solid, liquid, and gaseous radioactive waste that is generated, and should include appropriate engineering drawings showing

the location of the equipment and associated features used for volume reduction, containment, and/or packaging, storage, and disposal. The applicant should also discuss the bases of procedures associated with operating treatment equipment, including performance tests, process limits, and the means for monitoring and controlling to meet these limits. The bases of applicable technical specifications that control these procedures and functions should be discussed. The methods and agents planned for all activities involving routine disposal or release to the environment of radioactive wastes generated in the facility should be described, as should methods used for packaging and shipping solid and liquid radioactive wastes to other facilities or other means for processing, storage, or other disposition.

The applicant should describe the program for minimizing radioactive waste for the facility with respect to the following topics: (1) the specific numerical goals for reducing the volume or radioactivity of each waste stream; (2) the periodic assessments of reactor operations and experimental or utilization activities to identify opportunities to reduce or eliminate the generation of wastes; (3) the continuing efforts to identify and, where cost effective, implement waste reduction technologies; and (4) any periodic independent reviews performed to evaluate the effectiveness of programs to minimize radioactive waste.

11.2.3 Release of Radioactive Waste

The applicant should identify all radioactive waste materials for which controlled release to the environment or transfer to other parties for disposal is planned. This discussion should include the projected concentrations, forms, chemical compositions, and annual quantities of radioactive waste released under normal operating conditions.

All points from which radioactive waste effluents are designed to be released from the facility to the environment should be identified, using a site map to locate the effluent release points and effluent monitoring equipment. Discussions and detailed analyses of potential radiological impact of radioactive waste effluents and the bases for continuous or intermittent monitoring should be provided in the earlier sections of Chapter 11. For liquid releases to the sanitary sewerage, the applicant shall ensure that the requirements of 10 CFR 20.2003 are met. The applicant should describe the systems and procedures designed to ensure that doses resulting from releases of radioactive effluents do not exceed applicable regulatory limits and ALARA goals.

11.3 Bibliography

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.7, "Research Reactor Site Evaluation," ANS, LaGrange Park, Illinois, 1977.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1993.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.19, "Shipment and Receipt of Special Nuclear Material by Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1991.

Code of Federal Regulations, Title 10, "Energy," and Title 49, "Transportation," U.S. Government Printing Office, Washington, D.C., revised periodically.

U.S. Nuclear Regulatory Commission, NUREG-0851, "Nomograms for Evaluation of Doses From Finite Noble Gas Clouds," January 1983.

U.S. Nuclear Regulatory Commission, NUREG/CR-2260 "Technical Basis for RG 1.145," 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents."

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, proposed Revision 1, "Interpretation of Bioassay Measurements," Task DG-8009.

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Operational Exposures As Low As Is Reasonably Achievable."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.13, Revision 2, "Instructions Concerning Prenatal Radiation Exposure."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.29, proposed Revision 1, "Instruction Concerning Risks From Occupational Radiation Exposure," Task DG-8012.

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.37, "ALARA Levels for Effluents From Materials Facilities."

12 CONDUCT OF OPERATIONS

In this chapter of the SAR, the applicant should describe and discuss the conduct of operations at the reactor facility. The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator requalification plan, the startup plan, and environmental reports. The administrative aspects of facility operations are the facility organization, review and audit activities, organizational aspects of radiation safety, facility procedures, required actions in case of license or technical specification violations, reporting requirements, and recordkeeping. This chapter of the SAR forms the basis of Section 6 of the technical specifications.

12.1 Organization

In this section of the SAR, the applicant should discuss the organizational structure, responsibilities, and staffing, including selection and training of personnel. This discussion should show that the management and staff of the facility are knowledgeable about the technical requirements to operate a safe facility, are responsible for complying with regulations and license conditions, and will implement a meaningful radiation protection program that will protect the health and safety of the public, the facility users, and the staff. Additional information on these topics is given in Chapter 14, "Technical Specifications," of this format and content guide.

Not all owners and operators of non-power reactors have the same management organization or office titles. Regardless of the details of the management organization or the complexity of the facility, the administrative functions that should be in place at a non-power reactor facility are consistent among the various non-power reactor designs.

12.1.1 Structure

The applicant should discuss the organizational structure for the facility and should submit an organization chart. Lines of responsibility and lines of communication between groups should be shown in a multilevel chart. The individual or group with legal responsibility for holding the reactor license (e.g., the university provost or the dean) should be shown at the top of the organization, and the individuals who operate the reactor should be shown at the bottom. Intermediate levels should show the individuals who are in charge of the reactor facility and reactor operations (e.g., the facility director and the operations manager). The description of the organizational structure should include the radiation safety function and indicate how the staff implementing that function interacts with the staff

responsible for reactor operations and the top administrative officials. The multilevel chart should show the relationship of the review and audit function to the organizational structure. The persons implementing the review and audit function should communicate with the management of the reactor facility but should report to an organizational level above this management to ensure independence of the review and audit function.

12.1.2 Responsibility

The applicant should discuss the individuals or groups that appear in the organizational structure and their respective responsibility for the safe operation of the reactor and the reactor facility, the protection of the health and safety of the public and the workers at the facility, and the protection of the environment.

12.1.3 Staffing

The applicant should discuss staffing issues and the minimum staffing of the facility when the reactor is not secured. The applicant should also discuss the availability of senior reactor operators during routine operations and should list the events that require the presence of a senior reactor operator at the facility. Staffing shall meet, at a minimum, the requirements of 10 CFR 50.54(I), (j), (k), (l), and (m)(1). The applicant should show that these requirements are met.

12.1.4 Selection and Training of Personnel

The applicant should discuss the selection and training of personnel. If minimum requirements exist for the facility staff, they should be discussed in this section. For example, the facility director may have to meet certain educational standards. The applicant should discuss the training programs at the facility, including the initial training and the requalification training of reactor operators. The applicant and licensed operators shall comply with 10 CFR Part 55. The applicant shall meet the requirements of 10 CFR Part 19. The applicant should discuss the training necessary to ensure that all workers meet the requirements of 10 CFR Part 19. Specialized training may be needed for researchers who work near neutron beams or whose experiments may cause changes in reactivity in the reactor. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4–1988 contains additional guidance on the selection and training of personnel for research reactors.

12.1.5 Radiation Safety

The applicant should describe the organizational aspects of the radiation safety program. The program itself should be discussed in detail in Chapter 11, "Radiation Protection Program and Waste Management," of the SAR. The

radiation protection staff can be part of the reactor facility staff or may be provided as a service by a university or a corporation-wide group. The radiation protection staff can report to managers of either the university or the reactor. If this staff reports to the reactor manager, the applicant should discuss the method of raising concerns to the level of the review and audit committee or to managers above the facility management. The applicant should specifically discuss the authority of the radiation safety staff to interdict or terminate safety-related activities.

Additional guidance for radiation safety programs at non-power reactors is given in ANSI/ANS 15.11–1993.

12.2 Review and Audit Activities

In this section of the SAR, the applicant should discuss the review and audit activities at the reactor facility. This independent oversight is very important to the safe operation of the facility and the protection of the health and safety of the public. NRC expects review and audit programs to be viable and to be strongly supported by the licensee. Having independent experts examine certain activities improves the quality of the program. Independent audit allows the licensee to find and correct problems before NRC discovers them. Under its enforcement program, NRC looks favorably on licensees that aggressively seek out and correct problems in their programs.

The committee established for the review function may be assigned approval authority by the facility manager, or the facility manager may retain the authority. The applicant should explicitly state who holds the approval authority and should specify the committee's authority and how it communicates and interacts with facility management and university or corporate management.

12.2.1 Composition and Qualifications

The applicant should discuss the composition and qualifications of the review and audit committee. Both functions can be performed by one committee or two separate committees. The applicant should discuss the minimum number of committee members required. Committee members can be members of the reactor facility staff as long as they constitute a minority of the quorum for voting. The applicant should also discuss the qualifications of committee members. Because it is sometimes difficult, especially for small facilities, to have all types of engineering and health physics expertise on staff, the review and audit committee provides access to expertise. The committee members should have a variety of backgrounds to provide both additional expertise that the reactor staff may not have and to allow a second opinion in areas in which the reactor staff has expertise. The applicant should discuss the use of committee members from outside the university or corporation. It is desirable to have some members on the

committee from outside the university or corporation to increase the independence of the committee.

12.2.2 Charter and Rules

The applicant should discuss the committee's charter and rules, including the number of times the committee meets, the way the committee conducts business, the requirements for a quorum when voting, and the way the committee distributes its reports and reviews to the applicant. It is important that the definition of a quorum ensures that the reactor staff does not form a majority of members present for any vote.

12.2.3 Review Function

The applicant should list and discuss the items that must be reviewed by the committee. Additional information on the review function can be found in Chapter 14 of this format and content guide.

12.2.4 Audit Function

The applicant should list and discuss the items that must be audited by the committee. In addition to audits by the facility committee, the licensee may consider entering into a auditing agreement with other non-power reactor facilities to bring in staff members from other non-power reactors to perform an audit. This approach has been very productive at the facilities that have used it. The applicant should consider all aspects of facility operations for audit, including the radiation protection program (see Section 11.1.2 of this format and content guide) and the laboratory program at the facility if the program is conducted under the reactor license (see Section 9.5 of this format and content guide). The emergency plan, the physical security plan, and the operator requalification plan should be specified for audit, although the requirement for auditing these plans may be contained in the plan itself.

12.3 Procedures

In this section of the SAR, the applicant should discuss the use of procedures at the facility. NRC does not usually review procedures as part of licensing reviews. The applicant should discuss the basic topics that the procedures do or will cover. If laboratory work is conducted under the reactor license, the applicant should reference procedures applicable to this work. The applicant should discuss the methodology used for developing procedures, including the approval process. The applicant should also discuss the process required to make changes to procedures including substantive and minor permanent changes, as defined in ANSI/ANS 15.1–1990, and temporary deviations to deal with special or unusual circumstances

during operation. The applicant should note that 10 CFR 50.59 may apply to changes to procedures. See Chapter 14 of this format and content guide for additional information on the minimum acceptable set of procedures.

12.4 Required Actions

In this section of the SAR, the applicant should discuss actions to be taken in the event of a violation of the facility safety limits or the occurrence of a reportable event. In the case of a safety limit violation, the reactor shall be shut down, the proper facility management notified, the event investigated, and NRC notified. Additional information on required actions for a safety limit violation is given in Chapter 14 of this format and content guide..

The applicant should discuss what occurrences are considered reportable events and the actions to be taken if a reportable event occurs. These actions usually consist of returning the reactor to its normal condition or shutting the reactor down and reporting the event to facility management and NRC. Additional information on reportable events is given in Chapter 14 of this format and content guide.

12.5 Reports

In this section of the SAR, the applicant should discuss what information should be reported to NRC, the format of reports, the timing of reports, and the distribution of reports to NRC. This discussion and related technical specifications on reports take the place of the reporting requirements in 10 CFR 50.72 and 73, which are not applicable to non-power reactors. The applicant should discuss the annual operating report and should list the required contents. The annual report may also be used to meet the reporting requirements of 10 CFR 50.59 by discussing changes made to the facility and procedures and the tests and experiments conducted under the authority of 10 CFR 50.59. Additional information on the contents of annual operating reports is given in Chapter 14 of this format and content guide.

The applicant should discuss special reports, which are used to inform NRC of the violation of safety limits or the occurrence of reportable events, as previously discussed. The special reports are also used to inform NRC of changes in facility management and of changes in the transient and accident analyses in the SAR. Additional information on special reports is given in Chapter 14 of this format and content guide.

12.6 Records

In this section of the SAR, the applicant should discuss facility records and describe the recordkeeping system for the facility, the types of records that need to be retained, and the period of retention. Additional information on records is given in Chapter 14 of this format and content guide.

12.7 Emergency Planning

In this section of the SAR, the applicant should give a brief overview of the plan. The reader should be referred to the emergency plan for additional details.

The applicant should follow the guidance of ANSI/ANS 15.16–1978, which is endorsed and amplified by Regulatory Guide 2.6, March 1983, "Emergency Planning for Research and Test Reactors" (appears as Appendix 12.1). The applicant should also review NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors" (appears as Appendix 12.2).

The applicant should ensure that the terminology for releases discussed in the emergency plan match that used in the revision of 10 CFR Part 20 that became mandatory on January 1, 1994. Specifically, effluent concentrations (ECs) have replaced maximum permissible concentrations (MPCs), although use of dose values in millirem (millisievert) would be a more appropriate protective action guideline because dose is the ultimate criterion specified. The applicant should ensure that the action levels discussed in the emergency plan for each emergency class follow the guidance. If it is impossible for an event at a particular facility to reach a given action level, that emergency class is not possible and that fact should be so stated in the plan.

12.8 Security Planning

In this section of the SAR, the applicant should briefly discuss security planning. The information in the SAR must be nonproprietary and must not contain safeguards information. The proprietary or safeguards version of the security plan is protected from disclosure by the regulations. The guidance in Regulatory Guide 5.59 (appears as Appendix 12.3) should be referred to when developing a security plan. If an applicant believes that the special nuclear material the facility possesses or will possess will not allow the applicant to fall under the regulations for material of moderate or low strategic significance, the applicant should contact the project manager.

12.9 Quality Assurance

Section 50.34(a)(7) of 10 CFR requires that applicants for construction permits to build non-power reactors describe a quality assurance program for the design and construction of the structures, systems, and components of the facility. Section 50.34(b)(6)(ii) requires a description in the SAR of managerial and administrative controls to be used to ensure safe operation. The applicant should consider the guidance in Regulatory Guide 2.5 (appears as Appendix 12.4) and ANSI/ANS 15.8–1976 in developing quality assurance programs for non-power reactors. These guidance documents provide an acceptable method of complying with the program requirements of 10 CFR 50.34.

12.10 Operator Training and Requalification

Each reactor operator or senior reactor operator is required to successfully complete a requalification program developed by the licensee that has been approved by the Commission by passing a comprehensive written examination and an annual operating test. The operator requalification plan should describe the operator requalification program to be administered to non-power reactor personnel possessing operator and senior operator licenses in order for them to maintain active status. The plan should also discuss the requirements for reactor operators and senior reactor operators to maintain active status and the steps that should be taken to return an inactive operator to active status. The operator requalification plan is normally a separate document. It should include the organizational structure for implementing the program.

Section 50.54(I-1) of 10 CFR requires that within 3 months after an operating license is issued, the licensee have in effect an operator requalification program, which at a minimum meets the requirements of 10 CFR 55.59(c). In accordance with 10 CFR 55.59(a)(1) and 55.59(c)(1), the requalification period should not exceed 24 months.

The operator requalification plan should include information on the following:

- requalification schedule
- lectures, reviews, and examinations
- on-the-job training
- emergency procedures
- inactive operators
- evaluation and retraining of operators
- requalification documentation and records
- requalification document review and audit

Preplanned lectures should be given on a regular and continuing basis as required by 10 CFR 55.59(c)(2). Some examples of lecture topics are the following:

- nuclear theory and principles of operation
- facility design and general and specific operating characteristics
- instrumentation and control systems
- reactor protection system
- engineered safety features
- normal, abnormal, and emergency operating procedures
- radiation control and safety
- technical specifications and applicable portions of 10 CFR
- other facility specific training

Self-study methods may be considered an adequate and appropriate training method for the lecture program topics when learning objectives are properly measured by examination or documentation of expertise. Self-study programs must be justified by the applicant.

On-the-job training should occur during the requalification period so that each operator (1) is involved in facility manipulations [see 10 CFR 55.59(c)(3)(I)], (2) understands the operation of apparatus and mechanisms associated with control manipulations and knows operating procedures [see 10 CFR 55.59(c)(3)(ii)], (3) is cognizant of changes in facility design, procedures, and license [see 10 CFR 55.59(c)(3)(iii)], and (4) reviews the contents of all abnormal and emergency procedures on a regular basis [see 10 CFR 55.59(c)(3)(iv)]. Reactor operators must perform and senior reactor operators must either perform or supervise the appropriate items from 10 CFR 55.59(c)(3)(I), such as reactor startup, shutdown, and significant power change, on an annual basis.

NRC has approved requalification plans for on-the-job training that are similar to the following:

- Over the 2-year requalification period, each licensed individual should perform at least ten reactivity control manipulations in any combination of reactor startups, shutdowns, or significant reactivity changes, and each licensed operator should perform at least one reactor startup and one daily checkout quarterly at intervals not to exceed 4 months.

- Any changes in procedures, technical specifications, regulations, as well as any change with safety significance to the facility should be reviewed by every licensed operator on an ongoing basis as part of a required reading list.

- Each operator should participate in at least half of the emergency drills per year specified in the technical specifications. Operators who did not meet

the minimum drill requirement should receive special training on proper response to emergencies and a documented review of the last drill missed as well as a walkthrough of the facility related to proper emergency responses. The drill and applicable emergency procedures should be reviewed with all licensed operators within 30 days after completion of the drill.

The requalification program should include provisions for evaluating operators [see 10 CFR 55.59(c)(4)] that include written examinations, observation and evaluation of operator performance, and simulation of emergency or abnormal conditions. The applicant should discuss provisions for accelerated requalification if performance evaluations indicate the need [see 10 CFR 55.59(c)(4)(v)].

Examinations should be administered as necessary to assess the progress of the lectures or self study. A comprehensive requalification written examination is required for all operators at least biennially [see 10 CFR 55.41, 55.43, 55.59(a)(2)(I), and 55.59(c)(4)]. The contents of the comprehensive examination should be listed and should cover all lecture topics and requirements of the regulations.

For example, NRC has accepted requalification plans similar to the following:

- The acceptance criterion for all graded examinations is 80 percent and all operators are required to complete each examination satisfactorily.

- A score on the written or other examination equal to or greater than 80 percent may require no additional training. Nevertheless, the results of all examinations including missed questions should be reviewed with the operator to ensure proper understanding.

- A score on the written or other examination in the range of 65 to 79 percent requires additional training on those areas or topics where weaknesses or deficiencies are indicated. This retraining and retesting are completed within 60 days from the date the examination was administered and before the candidate is considered requalified. In this case the candidate need not be removed from licensed duties subject to the evaluation of the reactor manager or a duly authorized representative.

- A score on the written or other examination of less than 65 percent requires that an evaluation be performed by the facility director or designated representative within 1 month. The evaluation is to determine if the deficiencies require that the individual be removed from licensed duties pending completion of any accelerated retraining. In any case the licensed operator is removed from licensed duties if within 4 months he or she does not achieve a passing grade after reexamination.

- Regardless of the score, if the individual's test indicates a deficiency in a critical area that affects safety, training is promptly administered to correct the deficiency or the operator is removed from licensed duties in the affected area until the deficiency is corrected.

Each reactor operator and senior reactor operator is required to take an annual operations test to demonstrate operational proficiency and understanding of system responses [see 10 CFR 55.45(a)(2-13), 55.59(a)(2)(ii), and 55.59(c)(4)]. NRC has approved requalification plans similar to the following:

- Each licensed reactor operator and senior reactor operator demonstrates satisfactory understanding of the operation of the facility systems, operating procedures, and license as well as changes in facility procedures and the license during an annual walkthrough examination administered by a designated senior reactor operator.

- The annual operations test and the annual walkthrough examination are key factors in evaluating the continued competence of the licensed operator for demonstrating both (1) operational proficiency and understanding of system responses and (2) overall satisfactory understanding of the operations of the facility, operating procedures, and facility license changes. The results of these two examinations are used as primary input for evaluating operator performance for requalification purposes.

- An indepth evaluation of the operating performance of each licensed operator is performed and documented biennially, at a minimum, by a summary and judgment statements. The biennial evaluation includes results from the written examinations, the annual operations test, the annual walkthrough examination, and other on-the-job evaluation of operational proficiency as well as any other available indications of the operator's capability to discharge duties in a safe and competent manner, including participation in practical and special training, instructional activities, and other work activities.

The requalification plan should discuss records associated with the program [see 10 CFR 55.59(c)(5)]. NRC has accepted requalification plans similar to the following:

- Operator requalification records are kept to ensure that all requirements of the facility plan are met. Each operator has an individual folder or notebook containing signature blocks for lectures attended, prepared or assigned self-study sessions, reactivity manipulations performed, weekly and daily checkouts performed, and quarterly emergency drills participated in. The notebook also contains copies of written examinations administered, the

answers given by the operator, results of any evaluations, and documentation of any additional training administered in areas in which an operator has exhibited deficiencies. The performance of, or participation in, special training such as for fuel handling or use of emergency equipment is also logged in the notebook.

- A master requalification training manual is used to organize training requirements; the manual contains a schedule of all required lectures, reviews, emergency drills, and other exercises. The date the item is performed is indicated in this manual. A section of this manual is designated to contain completed training items, attendance sheets, master copies of tests given, and lecture outlines if available.

- Required documents and records pertaining to the requalification program are maintained as part of the facility records for at least 6 years. The records including the master training file are retained for each reactor operator or senior reactor operator until the respective operator's license is renewed or surrendered.

To maintain active status, 10 CFR 55.53(e) requires that each licensed reactor operator or senior reactor operator actively perform the functions of a reactor operator or senior reactor operator for a minimum of 4 hours each calendar quarter. For senior reactor operators, direct supervision of these operations may be considered equivalent to actual performance. If this requirement is not met, the license becomes inactive; before reactivation of the license, the licensee should verify that the qualifications and status of the operator are current and the operator should perform 6 hours of licensed activities in the position that is being recertified under the direction of a licensed operator or senior reactor operator as appropriate. For example, NRC has approved requalification plans similar to the following:

- An operator who has not been actively performing licensed functions for a period in excess of 4 months is required to demonstrate to the reactor manager or duly authorized representative that his or her knowledge and understanding of the operation and administration of the facility are satisfactory before returning to licensed duties. An interview and evaluation or a written, oral, or operational examination, or a suitable combination, can be used.

- Any deficiencies uncovered are corrected before the individual resumes performance of licensed functions.

- The operator performs 6 hours of operation under the direction of a reactor operator or senior reactor operator as appropriate.

The applicant should discuss audits of the requalification plan in the plan or as part of Section 6 of the technical specifications. The plan should be audited by the facility review and audit committee at least every other calendar year, with the interval between audits not to exceed 30 months. The audit should consist of a sampling of the plan records for completeness and compliance with the requirements of the plan.

12.11 Startup Plan

In this section of the SAR, the applicant should describe the startup plan for the facility. This section is applicable to new facilities or license amendments authorizing modifications that require verification of operability before normal operations are resumed. Startup plans ensure that the operating characteristics are well understood and validate the predicted behavior of the reactor. Measurements of selected parameters of the reactor should be compared to calculated values to verify analytical methods and ensure that meaningful acceptance criteria for the reactor have been established from the calculational methods. The acceptance criteria should ensure that the reactor is functioning within the bounds for which it was designed and analyzed and that the license and the technical specifications are satisfied.

The plan should include the following:

- a well-planned systematic set of subcritical multiplication measurements or an inverse multiplication approach to critical measurement during fuel loading, and confirmation that subcritical multiplication or critical fuel loading is within preestablished acceptable limits

- an experimental measurement plan to determine the important operational reactor physics parameters (such as control rod worth, excess reactivity, reactor thermal power, coefficients of reactivity, and power peaking factors) and thermal-hydraulic parameters (such as fuel, cladding, and coolant temperatures, reactor coolant system flow rates, and pressure drops, if appropriate); comparisons with predictions and acceptance criteria; and investigation and discussions of any discrepancies that may have arisen

- measurements of magnitudes of area radiation fields and radioactive effluents, and comparisons with predictions in the SAR and preestablished acceptance criteria

- measurements of performance of other systems (e.g., ventilation, engineered safety features, or emergency power), and comparisons with predictions in the SAR and preestablished acceptance criteria

The applicant should submit a startup report to NRC in accordance with the timeframe required in the technical specifications. A sample outline for a startup report follows (the actual contents of the report will vary by reactor type and scope of the report):

- measurements and comparisons with the prediction of subcritical multiplication for initial core fuel loading
- critical mass and final criticality conditions for the initial core and operational core, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- control and regulating rod calibration, including measurements of differential and integral rod worth for the initial and operational core, and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- excess (operational) reactivity, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- measured shutdown margin, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- reactor power calibration, including methods and measurements that ensure operation within the license limit; discussion of nuclear instrumentation setpoints, detector positions, and detector output; and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- thermal neutron flux distributions, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- radiation measurements of reactor coolant fission product inventory or release during the startup test program to detect potential degradation of the fuel fission product barrier or contamination from other sources; results of radiation measurements to show effectiveness of facility shielding; measurements of airborne effluents released from the facility; and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- reactivity worths of experimental facilities, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- results of determination of temperature coefficients and void coefficients for the reactor core, and comparisons with acceptance criteria preestablished in the SAR calculations and analyses

- Thermal-hydraulic characteristics such as fuel, cladding, and coolant temperatures, and reactor coolant system flow rates and pressure drops

- measurements of performance of engineered safety features and other tested systems, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses

This list is not complete. The components and systems to be tested and the test results reported to NRC will vary with reactor design and the subject of the licensing action.

12.12 Environmental Reports

NRC must comply with the National Environmental Protection Act of 1969, as amended, and recognizes a continuing obligation to conduct its domestic licensing in a manner that is receptive to environmental concerns. This means that licensing actions for non-power reactors must address environmental concerns. These licensing actions usually fall into one of three categories: those identified as categorical exclusions, those requiring the preparation of an environmental assessment, and those requiring the preparation of an environmental impact statement. Generally, most license amendments for non-power reactors fall under one of the categorical exclusions identified in 10 CFR 51.22; however, this is not always the case. Issuance of construction permits or operating licenses and license renewals for test reactors falls under 10 CFR 51.20 as regulatory action requiring environmental impact statements. Construction permits, operating licenses, license renewals, decommissioning plan orders, and license termination orders for research reactors are usually actions requiring environmental assessments. The environmental assessment is used to determine if an environmental impact statement or if a finding of no significant impact should be prepared.

For those actions for which an environmental impact statement or an environmental assessment must be prepared by the staff, NRC may require the applicant to submit an environmental report to help the staff to comply with the environmental regulations. The requirements for the contents of an environmental report are given in 10 CFR 51.45. For example, for an initial application for a construction permit and operating license or for license renewal, the applicant should describe the facility, discuss the environmental effect of facility site preparation and construction, environmental effects of facility operation, alternatives to construction and operation of the facility, and costs and benefits of facility alternatives.

12.13 References

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, LaGrange Park, Illinois, 1988.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1976.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Protection of Research Reactor Facilities, ANS, LaGrange Park, Illinois, 1993.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, "Emergency Planning for Research Reactors," ANS, LaGrange Park, Illinois, 1978.

Appendix 12.1

Regulatory Guide 2.6
Emergency Planning for Research and Test Reactors

Revision 1
March 1983

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 2.6
(Task HF 201-4)

EMERGENCY PLANNING FOR RESEARCH AND TEST REACTORS

A. INTRODUCTION

Paragraph 50.34(b)(6)(v) of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that each application for a license to operate a facility include in a Final Safety Analysis Report (FSAR), along with other information, the applicant's plans for coping with emergencies, including the items specified in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. Paragraph 50 54(q) requires licensees to follow and maintain in effect emergency plans that meet the requirements of Appendix E to 10 CFR Part 50

This regulatory guide provides licensees and applicants with a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the content of emergency plans for research and test reactors

Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0011

B. DISCUSSION

Working Group ANS-15.16 of the American Nuclear Society Subcommittee ANS-15 has developed ANSI/ANS 15.16-1982, "Emergency Planning for Research Reactors,"* which is generally consistent with current regulatory requirements. This standard was developed to provide specific acceptance criteria for complying with the applicable requirements set forth in § 50 54 and in Appendix E to 10 CFR Part 50 These criteria provide a basis for research and test reactor licensees and applicants to develop acceptable radiological emergency response plans and improve emergency preparedness at their facilities

The Commission's interest in emergency planning is focused primarily on situations that may cause or may threaten to cause radiological hazards affecting the health and safety of the public. Emergency plans should be directed toward mitigating the consequences of emergencies and should provide reasonable assurance that appropriate measures can and will be taken to protect the health and safety of the public in the event of an emergency. Although it is not practicable to develop a completely detailed plan encompassing every conceivable type of emergency situation, advance planning and provisions for ensuring the availability of necessary equipment, supplies, and services can create a high order of preparedness and ensure an orderly and timely decisionmaking process at the time of an emergency The plans should be an expression of the overall concept of operation that describes how the elements of advance planning have been considered and the provisions that have been made to cope with emergency situations

In the judgment of the NRC staff, the potential radiological hazards to the public associated with the operation of research and test reactors are considerably less than those involved with nuclear power plants In addition, because there are many different kinds of research and test reactors, the potential for emergency situations arising and the consequences thereof vary from facility to facility. These differences and variations are expected to be reflected realistically in the emergency plans and procedures developed for each research and test reactor facility.

C. REGULATORY POSITION

The requirements in ANSI/ANS 15.16-1982, "Emergency Planning for Research Reactors," are generally acceptable to the NRC staff as a means for complying with the requirements in § 50 54 and in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50 as related to research and test reactors, subject to the following clarifications.

1. Responsibility for planning and implementing all emergency measures within the site boundaries rests with the licensee. In this context, the site boundaries should be

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Ill 60525.

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention Docketing and Service Branch.

The guides are issued in the following ten broad divisions:

1 Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust and Financial Review
10. General

Copies of issued guides may be purchased at the current Government Printing Office price. A subscription service for future guides in specific divisions is available through the Government Printing Office Information on the subscription service and current GPO prices may be obtained by writing the U S Nuclear Regulatory Commission, Washington, D C 20555, Attention Publications Sales Manager

clearly defined. Supporting organizations that would augment the licensee's emergency organization, e.g., fire department, hospitals, and security organizations, should be specified. Planning and implementation of measures to cope with reactor-related emergencies beyond the site boundary should be commensurate with and based on the potential consequences of credible accidents or incidents. The emergency plan should describe this planning basis and the corresponding arrangements and agreements among the licensee and the local, State, or Federal agencies expected to respond.

2. The radiation dose levels of the emergency action levels established for the various emergency classes are slightly different from those specified for power reactors. However, in the judgment of the NRC staff, the radiation dose levels specified in Table I of the standard are adequate for the credible accidents associated with the operation of research and test reactors, and the specified action levels provide reasonable assurance that protective measures associated with the action levels specified in Table I can and will be taken, provided appropriate emphasis is also given to developing emergency action levels that relate directly to facility parameters (e.g., pool water levels and area radiation monitors)

3. Emergency action levels related to facility parameters, effluent release levels, and equipment conditions should be developed to the extent feasible for each emergency class

4. Details that can reasonably be expected to change from time to time, e.g., names and telephone numbers, specific items of equipment and supplies, inventory lists, and step-by-step procedures or checklists that may be altered as a result of experience or test exercises, should not be incorporated into the plans but should be listed in the emergency implementing procedures.

5. Emergency procedures that implement the emergency plan need not be incorporated into the plan but should be listed by title in an annex to the emergency plan. The emergency implementing procedures should be maintained and available at the facility for inspection and review at any time by a representative of the NRC.

6. The procedural system used by the licensee for the review and approval of emergency implementing procedures should contain instructions governing the writing, revising, and updating of implementing procedures. The instructions should specify the methods to be used to ensure that procedures, revisions, and changes are reviewed for adequacy, approved for use, and distributed to user organizations and individuals having the responsibility for implementing the procedures.

D. IMPLEMENTATION

Except in those cases in which an applicant or licensee proposes acceptable alternative practices or methods for complying with specified portions of the Commission's regulations, the practices or methods described in this guide will be used as a basis for evaluating the adequacy of the emergency plans and preparedness of applicants for a license to operate a research or test reactor as well as the plans and preparedness of current licensees for such reactors.

VALUE/IMPACT STATEMENT

1. THE PROPOSED ACTION

The licensee of a research and test reactor is required by the Commission's regulations to develop plans for coping with emergencies Specific guidance is needed to provide acceptance criteria for complying with the applicable requirements set forth in § 50.54 and in Appendix E of 10 CFR Part 50 Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," provides basic guidance for complying with the regulations More definitive guidance, however, has been developed by the American Nuclear Society Subcommittee ANS-15 in ANSI/ANS 15 16-1982, "Emergency Planning for Research Reactors " The proposed action would endorse this standard with appropriate supplementary material in a revision to Regulatory Guide 2 6

1 1 Value/Impact Assessment

The proposed action would provide licensees and applicants definitive guidance for developing emergency plans that meet the appropriate regulation

Value - The value of the proposed action would be more effective emergency preparedness around research and test reactors Endorsing a national consensus standard reduces the expenditure of staff resources in developing the guidance

Impact - Most of the impact on industry has already occurred during development, review, and approval of the consensus standard and in attempting to comply with the upgraded emergency preparedness requirements promulgated in August of 1980 For those members of the research reactor community that have not previously upgraded their emergency plans, it is estimated that it will take approximately 2 man-months to do so

1.2 Decision on the Action

Regulatory Guide 2 6 should be revised to endorse ANSI/ANS 15 16-1982

2. TECHNICAL ALTERNATIVES

Because the regulatory guide would endorse a consensus standard, no technical alternatives have been considered.

3. PROCEDURAL ALTERNATIVES

Because ANSI/ANS 15.16-1982 is generally consistent with current regulatory requirements, revising Regulatory Guide 2.6 to endorse that standard was selected as the appropriate procedural alternative

4. STATUTORY CONSIDERATIONS

4 1 NRC Authority

Authority for this action is derived from the Atomic Energy Act of 1954, as amended, through the Commission's regulations in Title 10, Chapter I, of the Code of Federal Regulations

4.2 Need for NEPA Assessment

Since the guidance in the proposed regulatory guide revision does not represent a major action as defined by paragraph 51 5(a)(10) of 10 CFR Part 51, implementation of the regulatory guide does not require a NEPA assessment

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICY

This revision to Regulatory Guide 2.6 relates to the NRC emergency preparedness regulations, Regulatory Guide 1.101, and NUREG-0654/FEMA-REP-1.

6. SUMMARY AND CONCLUSION

A revision to Regulatory Guide 2.6 should be published.

Appendix 12.2

NUREG-0849
Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors

NUREG-0849

Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors

Manuscript Completed September 1983
Date Published: October 1983

Division of Emergency Preparedness and Engineering Response
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This document provides a Standard Review Plan to assure that complete and uniform reviews are made of research and test reactor radiological emergency plans.

The report is organized under ten planning standards which correspond to the guidance criteria in American National Standard ANSI/ANS 15.16 - 1982 as endorsed by Revision 1 to Regulatory Guide 2.6. The applicability of the items under each planning standard is indicated by subdivisions of the steady-state thermal power levels at which the reactors are licensed to operate.

Standard emergency classes and example action levels for research and test reactors which should initiate these classes are given in an Appendix.

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STANDARD REVIEW PLAN FOR THE REVIEW AND EVALUATION OF EMERGENCY PLANS FOR RESEARCH AND TEST REACTORS

INTRODUCTION

Safety analyses for research and test reactors are based on the concept of a postulated Design Basis Event (DBE), an event for which the risk to the public health and safety is greater than that from any event that can be mechanistically postulated. The rationale for using the DBE for research and test reactors is to assess the potential effects to the public health and safety and is based on the determination that the offsite doses from the DBE be within the requirements of 10 CFR Parts 20 and 100. Consequently, if the requirements are met for a DBE condition, then the capability of the facility to withstand normal and abnormal operational transients and a broad spectrum of postulated credible accidents without undue risk to the public would also be defined within the DBE.

The postulated radioactive releases from credible accidents associated with the operation of research reactors will not result in offsite radiological doses to the general public exceeding the Protective Action Guides (PAGs) of 1 rem whole body or 5 rem thyroid. Therefore, these facilities would not include the General Emergency class of accidents requiring Federal assistance as part of their emergency plan.

Pursuant to 10 CFR 50.54(q), each licensee who is authorized to possess and/or operate a research or test reactor under a license of the type specified in 10 CFR 50.21(c), shall follow and maintain in effect emergency plans which meet the requirements in Appendix E to 10 CFR Part 50. Appendix E to 10 CFR Part 50, "Emergency Plans for Production and Utilization Facilities," establishes minimum requirements for emergency plans to attain an acceptable state of emergency preparedness and to provide reasonable assurance that protective measures can and will be taken to protect the health and safety of workers and the public.

Regulatory Guide 2.6 (Rev. 1, March 1983) "Emergency Planning for Research and Test Reactors," which is specified by Appendix E as the guidance to be used to determine the acceptability of research and test reactor radiological emergency plans, describes a method acceptable to the NRC staff for complying with the Commission's emergency planning regulations. Revision 1 to Regulatory Guide 2.6 (dated March 1983), endorses American National Standard, ANSI/ANS-15.16-1982, "Emergency Planning for Research Reactors."[1] This Standard identifies the elements of an emergency plan which describes the approach to coping with emergencies and minimizing the consequences of accidents at research and test reactor facilities. The emergency plan shall be implemented by emergency procedures.

[1]American National Standard for Emergency Planning for Research Reactors, ANSI/ANS-15.16-1982, American Nuclear Society, La Grange Park, IL.

This Standard Review Plan (SRP) has been prepared for performing reviews and evaluations for the acceptability of research and test reactor radiological emergency plans. The purpose of the SRP is to assure that uniform evaluations and complete reviews are made of each research or test reactor radiological emergency plan.

The report is organized under ten planning standards which correspond to the guidance criteria in ANSI/ANS-15.16-1982.[2]

Within the research and test reactor community, the licensed thermal power levels range from 0.1 W to 50 MW.[3] The inventory of radionuclides generated in reactor operations and the potential for accidents that result in a degraded core are largely dependent upon power level and operating history. Hence, the applicability of the planning standards to research and test reactors is also based upon power levels. Four ranges of power levels (equal to or less than 100 W, greater than 100 W to less than 100 KW, equal to or greater than 100 kW to equal to or less than 2 MW, and greater than 2 MW) are used in the text. The applicability of the items under each planning standard to reactors in each range is identified by an "X" in the appropriate column of the review sections.

It should be noted that the radiation dose levels of the emergency action levels established for the various emergency classes in Appendix I are slightly different from those specified for power reactors. However, in the judgment of the NRC staff, the radiation dose levels specified are adequate for the credible accidents associated with the operation of research and test reactors, and the specified action levels provide reasonable assurance that appropriate measures associated with the action levels specified can and will be taken, provided appropriate emphasis is also given to developing emergency action levels that relate directly to facility parameters, e.g., pool water levels and area radiation monitors.

Four standard emergency classes are defined in 10 CFR 50 Appendix E. The classes are Notification of Unusual Events, Alert, Site Area Emergency, and General Emergency.

The General Emergency class of accidents is not credible for most research or test reactors as this class is reserved for accidents which could have a significant radiological impact at substantial distances from the reactor. Therefore, most research or test reactors would not include this class as part of their emergency plan.

Acceptable sizes for Emergency Planning Zones (EPZs) are given in Appendix II as a function of authorized steady-state thermal power level. These are consistent with those given in ANSI/ANS-15.16-1982. The EPZ size will be determined on a case-by-case basis for any research or test reactors with power levels greater than 50 MW.

[2]The planning standards are extracted from American National Standard ANSI/ANS-15.16-1982, with permission of the publisher, the American Nuclear Society.

[3]Power level in this document means authorized steady-state thermal power level of the reactor.

CONTENT OF EMERGENCY PLAN

An emergency plan shall be prepared that addresses the necessary provisions for coping with radiological emergencies. Activation of the emergency plan or portions thereof shall be in response to the emergency action levels. In addition to addressing those severe emergencies that will fall within one of the standard emergency classes, the plan also shall discuss the necessary provisions to deal with radiological emergencies of lesser severity that can occur within the operations boundary. The emergency plan should allow for emergency personnel to deviate from actions described in the plan for unusual or unanticipated conditions.

The plan shall consist of the following elements and address, as applicable, the provisions identified for each element.

AREAS OF REVIEW, PLANNING STANDARDS, AND EVALUATION ITEMS

1.0 INTRODUCTION

PLANNING STANDARD

The plan should briefly introduce the type of reactor, the reactor's purpose, where it is located, and the purposes of the emergency plan. The purpose of the introduction is to provide a general orientation and common understanding about the reactor and the objective of the plan for those members of the reactor organization, the public, and local and federal agencies that will read and study the plan.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. The emergency plan should include the following:				
a. A description of the reactor including authorized power level.	X	X	X	X
b. A description of the location of the reactor facility including access routes.	X	X	X	X
c. Identification of the owner/operator.	X	X	X	X
d. A definition of the objective of the emergency plan.	X	X	X	X

2.0 DEFINITIONS

PLANNING STANDARD

Terms unique to the reactor facility or that have a special meaning when used in the plan should be defined in the plan.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
1. The emergency plan should include definitions of words or phrases with meanings specific or unique to the plan or reactor.	X	X	X	X

3.0 ORGANIZATION AND RESPONSIBILITIES

PLANNING STANDARD

The plan should describe the emergency organization that would be activated to cope with radiological emergencies. This includes the onsite emergency organization and any augmentation from offsite groups. Persons or groups that will fill positions in the emergency organization should be identified by their normal everyday title. This organizational description should include as appropriate the following evaluation items.

Evaluation Items	Applicability by Reactor Operating Power Levels			
	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. The emergency plan should describe the following organizational considerations:[4]				
a. The functions as applicable to emergency planning of Federal, State, and local government agencies and the assistance that they would provide in the event of an emergency.			X	X
b. The reactor's emergency organization, including augmentation of the reactor staff to provide assistance for coping with the emergency situation, recovery from the emergency, and maintaining emergency preparedness.	X	X	X	X
c. The arrangements and agreements, confirmed in writing with local support organizations that would augment and extend the capability of the facility's emergency organization.	X	X	X	X

[4] One or more of these positions may be assigned to the same incumbent.

Evaluation Items	Applicability by Reactor Operating Power Levels: ≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
d. A block diagram that illustrates the interrelationship of the facility emergency organization to the total emergency response effort. Interfaces between reactor and other onsite emergency organization groups and offsite local support organizations and agencies should be specified.	X	X	X	X
e. The capability of the emergency organization to function around-the-clock for a protracted period of time following the initiation of emergencies that have or could have radiological consequences requiring around the clock emergency response.				X
f. The identification by title of the individual in charge of directing emergency operations, including a line of succession, and responsibilities and authorities and those responsibilities which may not be delegated (such as notification and protective action decisions).	X	X	X	X
g. The identification by title of the individual, including a line of succession, and authority and responsibilities for coordinating emergency preparedness planning, updating emergency plans and procedures, and coordinating plans with other applicable organizations.	X	X	X	X
h. The identification by title of the individual, with a line of succession, responsible for relating information about the emergency situation to the news media and the public.			X	X

Evaluation Items	Applicability by Reactor Operating Power Levels			
	≤100 W	>100 W to <100 kW	≧100 kW to ≤2 MW	>2 MW
i. The identification by title of the individual, with a line of succession, in charge of radiological assessments including his/her responsibilities and authority for onsite and offsite dose assessments and recommended protective actions.			X	X
j. The identification by title of the individual, who may authorize reentry into the reactor building or portions of the facility that may have been evacuated during the emergency.	X	X	X	X
k. The identification by title of the individual authorized to terminate an emergency and initiate recovery actions and be responsible for informing the emergency organization of planned organizational actions or changes.	X	X	X	X
l. The identification by title of the individual, who may authorize volunteer emergency workers to incur radiation exposures in excess of normal occupational limits.		X	X	X

4.0 EMERGENCY CLASSIFICATION SYSTEM

PLANNING STANDARD

The emergency plan should describe several classes of emergency situations covering the spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the emergency organization. To provide for improved communications between the licensee, Federal, State and local agencies and organizations, the most severe accidents are standardized in four classes of emergency conditions which group the accidents according to severity of offsite radiological consequences. Each emergency plan should include only those standard classes appropriate for dealing with accident consequences determined to be credible for the specific facility. Most research reactors have potential emergency situations which may occur (e.g., personnel injury with contamination, fire, etc.) that have less severe offsite consequences than the least severe standard class, "Notification of Unusual Events." For some research reactors, no credible accidents are postulated which result in consequence matching the least severe class. However, planning for onsite emergencies is important. Preparedness for the onsite emergencies should be accomplished by identifying them and including in the plan those elements of this standard commensurate with the postulated emergency situations.

Each class of emergency should be associated with particular emergency action levels and with particular immediate actions to provide appropriate graded response. In order of increasing severity, the four standard emergency classes are described in qualitative terms in the following subsections:

4.1 NOTIFICATION OF UNUSUAL EVENTS. Notification of unusal events may be initiated by either man-made events or natural phenomena that can be recognized as creating a significant hazard potential that was previously nonexistent. There is usually time available to take precautionary and corrective steps to prevent the escalation of the accident or to mitigate the consequences should it occur. No releases of radioactive material requiring offsite responses are expected.

One or more elements of the emergency organization are likely to be activated or notified to increase the state of readiness as warranted by the circumstances.

Although the situation may not have caused damage to the reactor, it may warrant an immediate shutdown of the reactor or interruption of nonessential routine functions.

Situations that may lead to this class include: (1) threats to or breaches of security, such as bomb threats or civil disturbances directed toward the reactor; (2) natural phenomena, such as tornados in the immediate vicinity of the reactor, hurricanes, or earthquakes felt in the facility; (3) facility emergencies, such as prolonged fires, fuel damage indicated by high coolant fission product activity, or high offgas activity.

4.2 ALERT. Events leading to an alert would be of such radiological significance as to require notification of the emergency organization and its response as appropriate for the specific emergency situation. Under this class it is unlikely that offsite response or monitoring would be necessary. Substantial modification of reactor operating status is a highly probable corrective action. Protective evacuations or isolation of certain areas within the operations boundary or within the site boundary may be necessary. Situations that may lead to this class include: (1) severe failure of fuel cladding or of fueled experiments where containment boundaries exist to reduce releases or less severe cladding failures in situations where fission products are not well contained, or (2) significant releases of radioactive materials as a result of experiment failures.

4.3 SITE AREA EMERGENCY. A site area emergency may be initiated when events such as major damage of fuel or cladding and actual or imminent failure of other physical barriers containing fission products in reactor fuel or fueled experiments have occurred and projected offsite radiological consequences exceed Appendix I action levels. Monitoring at the site boundary should be conducted to assess the need for offsite protective actions. Protective measures on site may be necessary.

4.4 GENERAL EMERGENCY. A general emergency may be initiated by accidents which result in an uncontrolled release of radioactive material into the air, water, or ground to the extent that protective actions offsite may be necessary. This class of accident is not credible for most research reactors. Therefore, most research reactors would not include this class as part of their emergency plans.

A protective action that may be recommended to offsite authorities may be to shelter the general public within the EPZ. State and local government response organizations have the ultimate responsibility for initiating and implementing any recommended offsite protective actions.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
1. The emergency plan should contain:				
a. An emergency classification system consistent with the planning standard.	X_____	X_____	X_____	X_____
b. In an Appendix to the plan, a listing by title of implementing procedures for each class of emergency.	X_____	X_____	X_____	X_____

5.0 EMERGENCY ACTION LEVELS

PLANNING STANDARD

Because of the wide diversity in research reactors (power level, engineered safety features, site environment, etc.), those conditions which might initiate or signal a radiological incident having particular offsite consequences will vary widely among facilities. Action levels may be specified for effluent monitors or other plant parameters for which the dose rates and radiological effluent releases at the site boundary can be projected. To establish effluent action levels, facilities that have meteorological information available may base the action levels on actual meteorological conditions; otherwise, the criteria for downwind concentration, Section 4 of ANSI/ANS 15.7-1977, "Research Reactor Site Evaluation," should be used. Each emergency plan should establish emergency action levels appropriate for the specific facility and consistent with Appendix I. The emergency plan should include emergency action levels to initiate protective actions for members of the general public onsite. The protective action guide (PAG) shall be 1 rem whole body or 5 rem thyroid.[5]

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. Each licensee's emergency plan should contain:				
a. Emergency action levels which are appropriate to the specific facility and consistent with Appendix I. To the extent possible specify effluent monitors used to project dose rates and radiological effluent releases at the site boundary.	X	X	X	X

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[5]Manual of protective Action Guides and Protective Actions for Nuclear Incidents, EPA-520/1-75001, Sept. 1975, U.S. Environmental Protection Agency.

6.0 EMERGENCY PLANNING ZONES

PLANNING STANDARD

As part of emergency planning, the reactor owner/operator of a facility that identifies radiological emergencies which result in offsite plume exposures exceeding 1 rem whole body or 5 rem thyroid should identify an emergency planning zone (EPZ). The postulated radioactive release from credible accidents provides the basis for determining the need for an EPZ. The size of the EPZ should be established so that the dose to individuals beyond the EPZ is not projected to exceed the PAG. As an alternative to performing such calculations, the EPZ sizes in Appendix II may be adopted according to the power level.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. Ensure that the emergency plan identifies the EPZ.	X	X	X	X
2. If the EPZ is not consistent with Appendix II, the plan shall include an acceptable basis for the EPZ.	X	X	X	X

7.0 EMERGENCY RESPONSE

PLANNING STANDARD

Emergency response measures should be identified for each emergency. These response measures should be related to the emergency class and action levels that specify what measures are to be implemented.

Evaluation Items	Applicability by Reactor Operating Power Levels: ≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. The emergency plan should cover the following notification information for emergency response:				
a. The actions to notify and mobilize the emergency organization and the applicable offsite support organizations for each emergency class.	X	X	X	X
b. The location(s) of current notification lists.	X	X	X	X
c. Describe the contents of initial and followup emergency messages to the NRC and, when applicable, to offsite authorities. To the extent known, these messages should include the following:				
(1) Name, title and telephone number of caller, and the location of the incident and the emergency class.			X	X
(2) Description of emergency event.			X	X
(3) Date and time of incident initiation.			X	X
(4) The type and quantity of radionuclides released or expected to be released.			X	X

Evaluation Items	Applicability by Reactor Operating Power Levels: ≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
(5) Impact of releases and recommended offsite emergency actions.				X
d. A method is established to insure that offsite authorities have received the initial message and that it is authentic.			X	X
2. The emergency plan should cover the following assessment considerations:				
a. A description of methods for gathering and processing information for assessment actions.			X	X
3. The emergency plan should provide a summary description of those actions that could be taken to mitigate or correct the problem for each emergency class.	X	X	X	X
4. The emergency plan should describe protective actions appropriate for the emergency class. The emergency plan should include the following:				
a. Conditions for either partial or complete onsite evacuation, evacuation routes, and primary and alternate assembly areas.	X	X	X	X
b. Methods to ensure personnel accountability and the segregation of potentially contaminated personnel.	X	X	X	X
c. Protective measures and exposure guidelines for emergency personnel.	X	X	X	X

Evaluation Items	Applicability by Reactor Operating Power Levels			
	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
d. Provisions for isolation and access control of facility areas to minimize exposures to radiation and the spread of radioactive contamination.	X	X	X	X
e. The methods for monitoring radiation dose rates and contamination levels, both onsite and offsite, including provisions for transmitting collected information and data to the element of the emergency organization responsible for accident assessment.	X	X	X	X

8.0 EMERGENCY FACILITIES AND EQUIPMENT

PLANNING STANDARD

The emergency plan should briefly describe the emergency facilities, types of equipment and their location.

Evaluation Items	Applicability by Reactor Operating Power Levels: ≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. The emergency plan should describe an emergency support center (ESC).	X	X	X	X
2. Representative types of monitoring and sampling equipment to be used for accident assessment and their location. These should include:				
a. Portable and fixed radiological monitors.	X	X	X	X
b. Sampling equipment.	X	X	X	X
c. Instrumentation for specific radionuclide identification and analysis.	X	X	X	X
d. Personnel monitoring equipment.	X	X	X	X
e. The plan should also describe nonradiological monitors or indicators that may provide pertinent information; for example:				
(1) Reactor instrumentation.				X
(2) Fire detectors, earthquake sensors, etc.	X	X	X	X
(3) Source of meteorological data representative of facility location.				X

Evaluation Items	Applicability by Reactor Operating Power Levels			
	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
3. The emergency plan should identify those measures that will be used to provide necessary assistance to persons injured or exposed to radiation. The capabilities for decontamination, administering first aid, transporting injured personnel, and arrangements for medical treatment should be described. The following items should be included:				
a. Facilities for personnel decontamination.	X	X	X	X
b. Methods for handling and transporting contaminated injured personnel.	X	X	X	X
c. Written agreements with hospitals to ensure that medical services are available and the staff is prepared to handle radiological emergencies.	X	X	X	X
4. The emergency plan should adequately identify the emergency communications systems that will be available to communicate instructions and information both onsite and offsite throughout the course of an emergency.	X	X	X	X
5. Facilities planning for a site area emergency should establish reliable means of communication, e.g., public telephone and radio, that is compatible with local offsite support groups.				X

9.0 RECOVERY

PLANNING STANDARD

This element of the emergency plan should describe the criteria for restoring the reactor facility to a safe status including reentry into the reactor building or portions of the facility that may have been evacuated because of the accident. The operations to recover from most severe accidents will be complex and depend on the actual conditions at the facility. It is not practicable to plan detailed recovery actions for all conceivable situations.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
1. The emergency plan should specify:				
a. That recovery procedure(s) will be written and approved as needed.	X_____	X_____	X_____	X_____

10.0 MAINTAINING EMERGENCY PREPAREDNESS

PLANNING STANDARD

The emergency plan should describe the elements necessary for maintaining an acceptable state of emergency preparedness. A description should be provided of how the effectiveness of the emergency plan will be maintained, including training, review and update of the emergency plan and associated implementing procedures, and maintenance and inventory of equipment and supplies that would be used in emergencies.

	Applicability by Reactor Operating Power Levels			
Evaluation Items	≦100 W	>100 W to <100 kW	≧100 kW to ≦2 MW	>2 MW
1. The emergency plan should describe an initial training and periodic retraining program designed to maintain the ability of emergency response personnel to perform assigned functions for the following:				
a. Personnel responsible for decisionmaking and transmitting emergency information and instructions.	X	X	X	X
b. Personnel responsible for accident assessment.	X	X	X	X
c. Radiological monitoring and analysis teams.	X	X	X	X
d. First aid and rescue personnel.	X	X	X	X
e. Medical support personnel	X	X	X	X
f. Police, security, ambulance and fire fighting personnel.	X	X	X	X
2. The emergency plan should provide for:				

Evaluation Items	Applicability by Reactor Operating Power Levels: ≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
a. Annual onsite emergency drills, to be conducted as action drills.[6]	X	X	X	X
b. Provision for critiques of all drills, including timely evaluation of observer comments and correction of identified deficiencies.	X	X	X	X
c. Development of written scenarios for conducting annual action drills.	X	X	X	X
3. The emergency plan should provide for a biennial review and update of the emergency plan and implementing procedures and agreements with offsite support organizations and agencies including:				
a. Reviews and approvals by those responsible for emergency planning.	X	X	X	X
b. Incorporation of modifications resulting from action drills or changes in the facility or environs.	X	X	X	X
c. Timely forwarding of approved amendments to the plan, agreements, and implementing procedures to authorized individuals, agencies and support organizations.	X	X	X	X

[6]An action drill tests the integrated capability of the emergency plan, or a component thereof, and may include instruction periods to develop and maintain skills in a particular operation.

Evaluation Items	Applicability by Reactor Operating Power Levels			
	≤100 W	>100 W to <100 kW	≥100 kW to ≤2 MW	>2 MW
4. The emergency plan should describe the provisions to ensure operational readiness of emergency communications and emergency health physics equipment by including:				
a. Required maintenance and minimum calibration frequency.	X	X	X	X
b. Functional testing including minimum frequency.	X	X	X	X
c. Minimum frequency of inventory for equipment and supplies.	X	X	X	X

APPENDIX I

EMERGENCY CLASSES

Emergency Class	Action Level[1]	Purpose
Notification of Unusual Events	Actual or projected radiological effluents at the site boundary exceeding 10 MPC[2] when averaged over 24 hours, or 15 mrem whole body accumulated in 24 hours Report or observation of severe natural phenomenon Receipt of bomb threat	(1) Ensure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events information and decision-making
Alert	Actual or projected radiological effluents at the site boundary exceeding 50 MPC[2] when averaged over 24 hours, or 75 mrem whole body accumulated in 24 hours Actual or projected radiation levels at the site boundary of 20 mrem/hr for 1 hour whole body or 100 mrem thyroid dose	(1) Ensure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required, and (2) provide current offsite authorities status information
Site Area Emergency	Actual or projected radiological effluents at site boundary exceeding 250 MPC[2] when averaged over 24 hours, or 375 mrem whole body accumulated in 24 hours Actual or projected radiation levels at the site boundary of 100 mrem/hr for 1 hour whole body or 500 mrem thyroid dose.	(1) Ensure that response centers are manned, (2) ensure that monitoring teams are dispatched, (3) ensure that personnel required for evacuation of onsite areas are at duty stations, (4) provide consultation with offsite authorities and (5) provide information for the public through offsite authorities
General Emergency	Sustained actual or projected radiation levels at the site boundary or 500 mrem/hr whole body. Actual or projected dose at the site boundary in the plume exposure pathway of 1 rem whole body or 5 rem thyroid	(1) Initiate predetermined protective actions for the public, (2) provide continuous assessment of information from licensee and offsite organization measurements, (3) initiate additional measures as indicated by actual or potential releases, (4) provide consultation with offsite authorities, and (5) provide updates for the public through offsite authorities.

[1]The situations that may lead to an emergency class described in the subsections of Section 4.0 may be referenced as emergency actions levels appropriate to the emergency class

[2]Maximum Permissible Concentration (MPC) as listed in Title 10, of the Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation," Appendix B, Table II Column I

APPENDIX II

Alternate Method For Determining The Size of an EMERGENCY PLANNING ZONE[1]

Authorized Power Level	Acceptable EPZ Size
≦2 MW	Operations boundary
>2 MW and ≦10 MW	100 meters
>10 MW and ≦20 MW	400 meters
>20 MW and ≦50 MW	800 Meters
>50 MW	Will be determined on a case-by-case basis

[1]Calculations are based on:
D. Bruce Turner, Work Book of Atmospheric Dispension Estimates, Office of Air Programs. U.S. Environmental Protection Agency, Washington, D.C. (1970)

D. H. Slade, Ed., "Meteorology and Atomic Energy." U.S. Atomic Energy Commission, Washington, D.C. (1968); and

WASH 1400 (NUREG 75/014), "Reactor Safety Study," Appendix VI. U.S. Nuclear Regulatory Commission, Washington, D.C. (1975).

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16 ABSTRACT (200 words or less)

This document provides a Standard Review Plan to assure that complete and uniform reviews are made of research and test reactor radiological emergency plans. The report is organized under ten planning standards which correspond to the guidance criteria in American National Standard ANSI/ANS 15.16 - 1982 as endorsed by Revision 1 to Regulatory Guide 2.6. The applicability of the items under each planning standard is indicated by subdivisions of the steady-state thermal power levels at which the reactors are licensed to operate. Standard emergency classes and example action levels for research and test reactors which should initiate these classes are given in an Appendix.

The involvement of the Federal Emergency Management Agency (FEMA) concerning emergency response of research and test reactors is not required. However, the NRC will request FEMA assistance in the review process if the NRC staff identifies a facility which postulates a credible accident that could result in offsite radiological doses to the general public exceeding the Protective Action Guides of 1 rem whole body or 5 rem thyroid.

17 KEY WORDS AND DOCUMENT ANALYSIS

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17b IDENTIFIERS OPEN ENDED TERMS

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Appendix 12.3

Regulatory Guide 5.59, Revision 1
Standard Format and Content
for a Licensee Physical Security Plan
for the Protection of Special Nuclear Material
of Moderate or Low Strategic Significance

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
February 1983

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 5.59
(Task SG 229-4)

STANDARD FORMAT AND CONTENT FOR A LICENSEE PHYSICAL SECURITY PLAN FOR THE PROTECTION OF SPECIAL NUCLEAR MATERIAL OF MODERATE OR LOW STRATEGIC SIGNIFICANCE

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public. Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

The guides are issued in the following ten broad divisions.

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust and Financial Review
10. General

Copies of issued guides may be purchased at the current Government Printing Office price. A subscription service for future guides in specific divisions is available through the Government Printing Office. Information on the subscription service and current GPO prices may be obtained by writing the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Publications Sales Manager.

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INTRODUCTION

The Atomic Energy Act of 1954, as amended, directed the U.S. Atomic Energy Commission (AEC) to regulate the receipt, manufacture, production, transfer, possession, use, import, and export of special nuclear material (SNM) in order to protect the public health and safety and to provide for the common defense and security. The Energy Reorganization Act of 1974 transferred all the licensing and related regulatory functions of the AEC to the Nuclear Regulatory Commission (NRC).

The principal requirements with respect to the physical protection of licensed activities against industrial sabotage and with respect to the physical protection of special nuclear material in transit are found in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; Part 70, "Domestic Licensing of Special Nuclear Material"; Part 73, "Physical Protection of Plants and Materials"; and Part 110, "Export and Import of Nuclear Equipment and Materials."

Paragraph 50.34(c) of 10 CFR Part 50 and paragraphs 70.22(g), 70.22(h), and 70.22(k) of 10 CFR Part 70 identify the physical protection information that must be provided in a physical security plan as part of a license application. This plan is required in order for the applicant to demonstrate compliance with the specific physical protection requirements of 10 CFR Part 73 and must be submitted with each application for a license to possess or use SNM (or for a license authorizing transport or delivery of SNM), except for a license to possess, use, or transport less than 10 kg of SNM of low strategic significance, in which case a physical security plan is not required. However, for the protection of SNM of low strategic significance, the licensee is required to meet the requirements of § 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance," of 10 CFR Part 73.

Any guidance in this document related to information collection activities has been cleared under OMB Clearance Nos. 3150-0002, 3150-0009, and 3150-0011.

This regulatory guide describes the information required in the physical security plan submitted as part of an application for a license to possess, use, or transport SNM of moderate strategic significance or 10 kg or more of SNM of low strategic significance and recommends a standard format for presenting the information in an orderly arrangement. This standard format will thus serve as an aid to uniformity and completeness in the preparation and review of the physical security plan of the license application. This document can also be used as guidance by licensees possessing or transporting less than 10 kg of SNM of low strategic significance in understanding the intent and implementing the requirements of paragraphs 73.67(a), 73.67(f), and 73.67(g) of 10 CFR Part 73.

Aside from providing guidance for the standard format and content of physical security plans, this regulatory guide explains the intent of the various provisions of the regulation. The intent of each requirement is found in the discussion of each subsection and is implicitly provided by outlining alternative systems that could be used to fulfill the requirements. The

discussion section and list of alternatives should provide the licensee with the sense of the NRC regulations.

This guide is divided into two parts. Part I, "Special Nuclear Material of Moderate Strategic Significance," provides a standard format for preparing the licensee's security plans and provides guidance to licensees who possess, use, or transport SNM of moderate strategic significance. Chapters 1 through 6 of Part I apply to applications for a license to possess or use at any fixed site, or at contiguous sites subject to control by the licensee, SNM of moderate strategic significance. Chapters 7 through 13 of Part I apply to applications for authorization to transport or deliver to a carrier for transport SNM of moderate strategic significance.

Part II, "Special Nuclear Material of Low Strategic Significance," provides a standard format for preparing the licensee's security plan for licensees who possess, use, or transport more than 10 kg of SNM of low strategic significance. It also provides guidance to all licensees who possess, use, or transport SNM of low strategic significance. Chapters 1 through 4 of Part II apply to applications for a license to possess or use at any fixed site, or at contiguous sites subject to control by the licensee, more than 10 kg of SNM of low strategic significance. Included in this category are licensees who have nuclear power reactors under construction and are seeking a license to possess nuclear fuel onsite prior to obtaining their operating license. Chapters 5 through 9 of Part II apply to applications for authorization to transport or deliver to a carrier for transport more than 10 kg of SNM of low strategic significance.

Table 1 shows the type and amount of SNM covered in § 73.67 of 10 CFR Part 73. It should be noted, as stated in the footnote to Table 1, that (1) plutonium with an isotopic concentration exceeding 80 percent or more in Pu-238, (2) special nuclear material that is not readily separable from other radioactive material and that has a total external radiation dose rate in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding, and (3) sealed plutonium-beryllium neutron sources totaling 500 grams or less of contained plutonium at any one site or contiguous sites are exempt from the requirements of § 73.67 of 10 CFR Part 73.

This guide has been prepared to minimize time lost because of incomplete physical security plans and to standardize the review process. Applicants are encouraged to prepare their physical security plans in accordance with this guide and to provide information in each section to support the conclusion that they will be able to operate in accordance with the pertinent regulations. Although conformance with this guide is not required, the format and content presented are acceptable to the NRC staff.

As developments and changes in the nuclear industry occur, the Commission's requirements for information may need modification; revisions to this guide will be made as necessary to accommodate these changes.

Purpose and Applicability

This standard format has been prepared as an aid to uniformity and completeness in the preparation and review of the physical protection section of license applications and to clarify the intent of the regulations. The

Table 1

CATEGORIES OF SPECIAL NUCLEAR MATERIAL

MATERIAL*	ENRICHMENT	MODERATE STRATEGIC SIGNIFICANCE	LOW STRATEGIC SIGNIFICANCE
1. Plutonium	--	Less than 2,000 g but more than 500 g	500 g or less but more than 15 g
2. Uranium-235	20% or more in U-235 isotope	Less than 5,000 g but more than 1,000 g	1,000 g or less but more than 15 g
	10% or more but less than 20% in U-235 isotope	10,000 g or more	Less than 10,000 g but more than 1,000 g
	Above natural but less than 10%	--	10,000 g or more
3. Uranium-233	--	Less than 2,000 g but more than 500 g	500 g or less but more than 15 g
4. Uranium-235, uranium-233, and plutonium in combination	U-235 portion enriched to 20% or more.	Less than 5,000 g according to the formula: grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium) but more than 1,000 g according to the formula: grams = (grams U-235) + 2.0 (grams U-233 + grams plutonium)	1,000 g or less according to the formula: grams = (grams contained U-235) + 2.0 (grams U-233 + grams plutonium) but more than 15 g according to the formula: grams = grams contained U-235 + grams U-233 + grams plutonium.

*The following materials are exempt:

1. Special nuclear material that is not readily separable from the radioactive material and that has a total external radiation dose rate in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding,
2. Plutonium with an isotopic concentration of 80 percent or more in Pu-238, and
3. Sealed plutonium-beryllium neutron sources totaling 500 grams or less of contained plutonium at any one site or contiguous sites.

information this guide contains will help the licensee plan a physical protection system designed to detect the theft of SNM of moderate or low strategic significance. The physical protection subsystems identified are those that will usually be included in a protection system that would normally be capable of meeting the performance requirements of paragraph 73.67(a) of 10 CFR Part 73. However, it is recognized that at any particular site there may be some subsystems and components not needed or additional ones needed to meet these performance requirements. In these cases, applicants are encouraged to address in the license application specific departures of subsystems or components from this guide.

The information requested in this guide is the minimum needed for the review of a physical security plan. Additional information may be required for completing the staff review of a particular plan and should be included as appropriate. It is also the applicant's responsibility to be aware of new and revised NRC regulations. The information provided should be up to date with respect to the state of technology for the physical protection techniques and systems that the applicant proposes to use.

In cases where NRC-approved security plans are already in existence and where the measures included therein meet or exceed the requirements of § 73.67 of 10 CFR Part 73, licensees may reference in their proposed security plans those sections of the NRC-approved security plans that are applicable.

Information and procedures delineated in the regulatory guides in Division 5, "Materials and Plant Protection," that are appropriate to certain sections of the physical security plan may be incorporated by reference.

Applicants should discuss their plans and programs with the NRC staff before preparing the applications. These discussions should give particular emphasis to the depth of information required for the plans.

Upon receipt of an application, the NRC staff will perform a preliminary review to determine whether the application provides a reasonably complete presentation of the information needed to form a basis for the findings required before issuance of a license. The standard format will be used by the staff as a guideline for identifying the type of information needed. If an application does not provide a reasonably complete presentation of the necessary information, further review of an application will be suspended until this needed information is provided.

Use of Standard Format

The applicant should follow the numbering system of the Standard Format down to the level of section (e.g., 3.4). Under some circumstances, certain sections may not be applicable to a specific application. If so, this should be clearly stated and sufficient information should be provided to support that conclusion.

The applicant may wish to submit in support of the application information that is not required by regulations and is not essential to the description of the applicant's physical protection program. Such information could include, for example, historical data submitted in demonstration of certain criteria, discussion of alternatives considered by the applicant, or supplementary data

regarding assumed models, data, or calculations. This information should be provided as an appendix to the application.

Upon completion of the application, the applicant should use the Table of Contents of the Standard Format as a checklist to ensure that each subject has been addressed.

Style and Composition

A table of contents should be included in each submittal.

The applicant should strive for clear, concise presentation of information. Confusing or ambiguous statements and general statements of intent should be avoided. Definitions and abbreviations should be consistent throughout the submittal and consistent with generally accepted usage.

Wherever possible, duplication of information should be avoided. Thus, information already included in other sections of the applications may be covered by specific reference to those sections.

Where numerical values are stated, the number of significant figures should reflect the accuracy or precision to which the number is known. The use of relative values should be clearly indicated.

Drawings, diagrams, and tables should be used when information may be presented more adequately or conveniently by such means. These illustrations should be located in the section where they are first referenced. Care should be taken to ensure that all information presented in drawings is legible, that symbols are defined, and that drawings are not reduced to the extent that they cannot be read by unaided normal eyes.

Physical Specifications of Submittals

All material submitted in an application should conform to the following physical dimensions of page size, quality of paper and inks, numbering of pages, etc.:

1. Paper Size

   Text pages: 8-1/2 x 11 inches.

   Drawings and graphics: 8-1/2 x 11 inches preferred; however, a larger size is acceptable provided the finished copy when folded does not exceed 8-1/2 x 11 inches.

2. Paper Stock and Ink

   Suitable quality in substance, paper color, and ink density for handling and for reproduction by microfilming.

3. Page Margins

   A margin of no less than 1 inch is to be maintained on the top, bottom, and binding side of all pages submitted.

4. Printing

Composition: text pages should be single spaced.

Type face and style: must be suitable for microfilming.

Reproduction: may be mechanically or photographically reproduced. All pages of the text may be printed on both sides, and images should be printed head to head.

5. Binding

Pages should be punched for looseleaf ring binding.

6. Page Numbering

Pages should be numbered by section and sequentially within the section. Do not number the entire report sequentially. (This entire Standard Format has been numbered sequentially because the individual chapters were too short for sequential numbering within each section to be meaningful.)

7. Format References

In the application, references to this Standard Format should be by part, chapter, and section numbers.

Procedures for Updating or Revising Pages

The updating or revising of data and text should be on a replacement page basis.

The changed or revised portion of each page should be highlighted by a vertical line in the margin opposite the binding margin for each line changed or added. All pages submitted to update, revise, or add pages to the report should show the date of the change. The transmittal letter should include an index page listing the pages to be inserted and the pages to be removed. When major changes or additions are made, pages for a revised table of contents should be provided.

Number of Copies

The applicant should submit the appropriate number of copies of each required submittal pursuant to § 70.21, "Filing," of 10 CFR Part 70.

Public Disclosure

The NRC has determined that the public disclosure of the details of physical protection programs is not in the public interest, and such details are withheld pursuant to paragraph 2.790(d) of 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings." Thus the physical protection section of each application should be submitted as a separate enclosure. Other proprietary and classified information should be clearly identified and submitted in separate enclosures. Each such submittal of proprietary information should be accompanied by the applicant's detailed reasons and justifications for requesting exemption from public disclosure as required in paragraph 2.790(b) of 10 CFR Part 2.

PART I

SPECIAL NUCLEAR MATERIAL

OF MODERATE STRATEGIC SIGNIFICANCE

1. USE AND STORAGE AREA AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraphs 73.67(d)(1) and (d)(2), which are as follows:

(d)(1) Use the material only within a controlled access area which is illuminated sufficiently to allow detection and surveillance of unauthorized penetration or activities.*

(d)(2) Store the material only within a controlled access area such as a vault-type room or approved security cabinet or their equivalent which is illuminated sufficiently to allow detection and surveillance of unauthorized penetration or activities.*

A controlled access area (CAA) is defined in paragraph 73.2(z) as "any temporarily or permanently established area which is clearly demarcated, access to which is controlled and which affords isolation of the material or persons within it." "Access control" means measures used to allow only specified personnel, materials, and vehicles ingress into and egress from a given area, while "isolation" refers to measures taken to deter persons, materials, or vehicles from entering or leaving a given area through other than established access control points. Thus, a CAA includes provisions for both isolation and access control. In some cases, isolation or access control systems may also serve the purpose of aiding in the detection of unauthorized penetration or activities* within the CAA. Therefore, these detection-related considerations are alluded to in this chapter.

In the discussion that follows, CAAs intended for use and the possible storage of SNM of moderate strategic significance are discussed separately from those intended solely for the storage of such SNM. Although the requirements for these two different applications of CAA are in most respects similar, it is recognized that the means used to isolate the material and control access may differ markedly for use areas compared with storage areas and therefore warrant separate discussion.

1.1 Area Where Material Is Used [73.67(d)(1)]

Intent

This section discusses CAAs intended primarily for the use of SNM of moderate strategic significance. These may be temporarily established to meet transitory or intermittent SNM use requirements or they may be permanently established. Permanently established CAAs for use of SNM may also be suitable for storage. "Use" means that the material is undergoing processing (e.g., fuel fabrication, irradiation in a reactor) or utilization of its properties in conjunction with experimental equipment (e.g., equipment used for research or educational laboratory experiments). Different isolation/access control measures may be used for periods during which the area is occupied versus unoccupied.

---

*Unauthorized activities are those activities deemed by the licensee to be indicative of or contributory to the possible theft of SNM.

Illumination sufficient to allow detection and surveillance of unauthorized penetration or activities within the CAA where the material is used need not require the use of high-intensity lighting throughout the CAA. What is intended is the use of normal lighting sufficiently uniform throughout the CAA to ensure that material or unauthorized personnel cannot be secreted in a darkened area until a time more convenient for the unauthorized removal of the material. For those facilities where experiments must be conducted in a darkened room, the lighting requirement is exempted for as long as is needed provided access control is ensured and the material is accounted for at the end of the experiment.

1. Temporarily Established CAAs

Temporarily established CAAs for the use of SNM need not have permanent barriers at their boundaries. Isolation of the material and persons using the material may be provided by office partitions, cordons, or other devices used to warn passersby of the restricted nature of the area. Access control can be effected through surveillance or supervision of the area by those who are using the SNM and who are responsible for the material. However, the provision of access control through personal supervision is suitable only for those situations in which the size of the CAA and the number of persons to be admitted are sufficiently small and in which the CAA is suitably configured to make such procedures practical.

When material located in a temporarily established CAA is to be left unattended (i.e., because it is impractical to replace the material in a CAA designed for long-term storage of the material), access control and improved isolation for the temporary CAA may be provided to substitute for the discontinued personal supervision over the material. (In addition, provisions must be made for satisfying the monitoring requirement of paragraph 73.67(d)(3).) Improved isolation can sometimes be provided by increasing the penetration resistance of existing barriers (e.g., locking doors and windows, placing the material in a locked drawer or supply room). Access control may be provided by any of the measures suggested below for permanently established CAAs left unattended; some of these suggestions may be more suitable for temporary use than others. If no suitable barrier is available to provide isolation, material left unattended may be protected by a motion alarm covering the area immediately surrounding the material. This improved detection capability would be considered an acceptable substitute for the rudimentary isolation provided by temporary barricades, cordons, signs, and other less substantial means of isolation that may be used on a temporary basis.

2. Permanently Established CAAs

Permanently established CAAs for the use and temporary storage of SNM of moderate strategic significance would most likely provide isolation for the SNM through the use of permanent barriers. These could consist of fences; gates or freestanding walls for exterior areas; or exterior or interior building walls, locked doors, windows, bars, grillwork; or other barriers for interior areas. Such barriers are not required to meet the more stringent criteria for physical barriers used for the protection of formula quantities of strategic special

nuclear material (SSNM).* However, good security management practice would dictate that the barrier be substantial enough to deter casual passersby from unauthorized penetration. If the barriers are also designed to aid in detection, the criteria for penetration resistance or tamper indication in accordance with the specific monitoring procedures to be used also apply.

Access control for permanently established CAAs during periods they are occupied can be provided by personal supervision. However, when these areas are unoccupied or when the size, configuration, or numbers of persons intended to be admitted to the area make personal supervision impractical, other means of access control may be called for. Some of the additional access control measures that may be used for permanently established CAAs** in these situations are:

- Stationing a watchman at CAA access control points,
- Limiting distribution of keys, keycards, or combinations to doors and gates,
- Using a coded badging system to identify authorized personnel at CAA access control points,
- Controlling locked CAA doors and gates by use of remote surveillance (e.g., intercom or CCTV) by personnel stationed at a centrally located access control facility, or
- Controlling access of personnel, material, and vehicles into CAAs by other means.

It is expected that large facilities will have more complex isolation and access control systems than small facilities in order to achieve comparable levels of protection.

Content

(This section needs to be completed only for CAAs designated exclusively for use of SNM. CAAs designated for both use and storage are to be described as recommended in Section 1.2 of Part I of this Standard Format.)

Describe the CAAs designated exclusively for use of the material. Indicate under what conditions CAAs will be established on a temporary basis. For each CAA identified, provide the following information:

1. A description of the area and its features relative to other facility features, showing the normal routes of ingress to each CAA (including a scale diagram).

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*"Strategic special nuclear material" means uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium.

**Access control requirements for CAAs for the use or storage of SNM of moderate strategic significance are discussed in Chapter 3, "Access Control at a Fixed Site," of Part I of this guide.

2. Descriptions of physical barriers, cordons, walls, partitions, or other means of providing isolation of the CAA and channelling entry through established access control points into the CAA.

3. The means and criteria used for controlling access into the CAA at established access control points.

4. If more than one CAA is designated, the types of material normally used in each CAA; for temporarily established CAAs, the types of activities normally performed within each area.

5. The lighting level and uniformity of lighting provided to allow detection and surveillance of unauthorized penetration or activities within the CAA or in the immediate vicinity of the CAA.

1.2 Area Where Material Is Stored [73.67(d)(2)]

Intent

CAAs normally used only for storage of SNM are described in this section. Such CAAs may be very limited in size since they need only contain the material itself, and access to them is expected to be infrequent. Where small amounts of material are involved, the CAA may consist of relatively small containers such as security cabinets, safes, and locked closets or supply rooms.

The same basic isolation and access control* capabilities that apply to CAAs for the use of SNM of moderate strategic significance also apply to CAAs for the storage of this material. However, the CAA for storage is, in addition, specifically required to be equivalent to either an approved security cabinet or a vault-type room. This additional requirement embodies a tradeoff decision to be made by the licensee between the capability for immediate detection of penetration attempts into a vault-type room and the improved penetration resistance likely to be provided by an approved security cabinet.

A vault-type room is defined in paragraph 73.2(o) of 10 CFR Part 73 as "a room with one or more doors, all capable of being locked, protected by an intrusion alarm which creates an alarm upon the entry of a person anywhere into the room and upon exit from the room or upon movement of an individual within the room." The vault-type room can be a locked laboratory, supply room, closet, or other room equipped with a tamper-resistant motion detector alarm system. The motion detector would simultaneously satisfy the monitoring requirement of paragraph 73.67(d)(3) addressed in the next chapter. The expression "equivalent to a vault-type room" means that a piece of equipment (such as a fission chamber, reactor core, or storage rack), even though it does not resemble a "room," may meet the requirement for a storage-type CAA if there is a means of isolation (e.g., a locked grill, inaccessibility beneath water as in a storage pool, intricate and time-consuming procedures required for removal) and it is protected with a tamper-resistant motion detection system.

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*Access control requirements for CAAs for the use or storage of SNM of moderate strategic significance are discussed in Chapter 3 of Part I of this guide.

Note that, in some cases, material located in such pieces of equipment may be considered in use rather than in storage. In this case, whether the material is being personally supervised or not, the requirement for use of a motion detection system to satisfy the monitoring requirement is removed, and other monitoring devices or procedures may be used instead.

When a container equivalent to an approved security cabinet is chosen to satisfy paragraph 73.67(d)(2), the equivalency is based on the penetration resistance of the container and the difficulty associated with manipulating the lock. The basic premise used to determine whether such a container may be monitored by procedures or devices other than a motion detector is whether it is unreasonable to expect that an external adversary could penetrate the container in a reasonable amount of time without leaving an indication of the penetration. The amount of time that would be required for penetration without such an indication may be used to govern the frequency of patrols when monitoring procedures are used. An approved security cabinet or its equivalent is one whose design has been certified by the General Services Administration or other nationally recognized standards organization (e.g., ANSI) to afford protection against surreptitious entry and lock manipulation equivalent to that provided by a Class-6 GSA rating or better.

Isolation for CAAs intended only for storage usually will be provided by the penetration-resistant features of the perimeter barriers or container walls of the CAA. The level of penetration resistance of such barriers is not required to meet the more stringent criteria for physical barriers used to protect formula quantities of SSNM. Rather, the physical barriers or container walls of the CAA are intended only to deter penetration by unauthorized persons and to aid detection by providing an indication of forced penetration.

In determining the level of isolation/access control required for such CAAs, the time required for extricating SNM from its storage location in relationship to the licensee's time of detection and assessment may be taken into account (e.g., spent fuel assemblies stored in spent fuel storage pools or fuel residing in nonpower reactor cores).

Examples of typical CAAs where special nuclear material of moderate strategic significance may be stored are:

1. Vault. A structure that satisfies the definition for a vault as stated in paragraph 73.2(n) would provide more than adequate isolation protection.

2. Approved Security Cabinets. Those cabinets that are designed to provide delay and resistance against surreptitious entry and lock manipulation.

3. Reactor. Reactors that are so designed that removal of material is difficult.

4. Vault-Type Room. Some typical vault-type rooms where materials are stored and protected with a motion detector are storage pools, a room containing in-process storage racks, and laboratories where material is left unattended. In all cases, movement in the near vicinity of the material or of the material itself generates an alarm signal.

5. Locked Laboratories, Supply Rooms. These areas must be sufficiently penetration resistant to afford a means of isolation/access control and permit the proper functioning of the system for monitoring the storage area as required by paragraph 73.67(d)(3) of the rule and as described in Chapter 2, "Detection Devices and Procedures at a Fixed Site," of Part I of this guide.

The illumination level required for the CAA should be sufficiently uniform and bright (a) in the case of a security cabinet, to detect penetration of or tampering with the CAA containment or (b) in the case of a vault-type room or its equivalent, to detect unauthorized penetration of or activities within the CAA.

Content

Describe the CAAs where the material will be stored and, in some cases, used. For each such CAA, provide the following information:

1. A description of the area and its features relative to other facility features, showing the access control points for the CAA (including a scale diagram).

2. The isolation system such as physical barriers, container walls, or other features that demarcate the perimeter of the CAA and channel entry only through established access control points.

3. The means and criteria used for controlling access to the CAA at established access control points.

4. If more than one CAA is designated for the storage of the material, the types of material normally to be stored in each area.

5. The lighting level and uniformity of lighting provided to allow detection and surveillance of unauthorized penetration or activities within the CAA or in the immediate vicinity of the CAA (where security cabinets are used).

2. DETECTION DEVICES AND PROCEDURES AT A FIXED SITE

This chapter provides guidance for meeting the requirement of paragraph 73.67(d)(3), which is as follows:

(d)(3) Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities.

The purpose of this monitoring activity as stated in paragraph 73.67(a)(2) is to provide "early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area..." and "early detection of removal of special nuclear material by an external adversary from a controlled access area." This should be done to achieve the objective of minimizing the possibilities for unauthorized removal of special nuclear material "consistent with the consequences of such actions," as stated in paragraph 73.67(a) of the effective rule. Thus, the earliness of detection afforded by the licensee's physical protection system may vary depending on the nature and quantity of the material susceptible to unauthorized removal. Further discussion of the earliness of detection required for possible thefts of different types and quantities of special nuclear material of moderate strategic significance is provided in Section 2.1 of Part I of this guide.

2.1 Earliness of Detection [73.67(a)]

Determination of early detection of unauthorized access, activities, or removal of SNM from CAAs will be based on an assessment of the magnitude of the consequences associated with possible misuse of the type and quantity of material that could be removed in a given theft attempt. For SNM of moderate strategic significance, two distinct cases are considered:

1. Theft of strategic special nuclear material (SSNM).*

2. Theft of low-enriched uranium (LEU).**

Thefts from a single facility of SSNM in quantities of moderate strategic significance are limited by definition to quantities that could not be used to construct a nuclear explosive device; thefts of similar quantities of material from several different facilities could, however, lead to the accumulation of an aggregate quantity that would permit such illicit use. (The consequences of a single theft of such material would have minimal impact on the public health and safety.) Therefore, detection of such thefts is required sufficiently early to ensure early notification to the NRC so that the NRC in turn can notify other licensees of the need to implement appropriate responses to prevent the accumulation by a single adversary of a formula quantity of material through multiple thefts from different facilities. The NRC believes that there are detection systems and procedures that would allow the licensee to detect a theft of SSNM within approximately 2 hours.

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* In this case, strategic special nuclear material means all SNM of moderate strategic significance other than low-enriched uranium.

** Low-enriched uranium is uranium enriched above natural percentages but less than 20 percent in the U-235 isotope.

The criteria for early detection of thefts of LEU enriched to more than 10 percent, but less than 20 percent, in the isotope U-235 are the same as for thefts of LEU in quantities comprising SNM of low strategic significance. The reader should refer to Part II of this guide for information regarding appropriate monitoring procedures to achieve early detection for such material.

2.2 Detection Through Monitoring Controlled Access Areas [73.67(d)(3)]

Intent

Licensees possessing quantities of SNM of moderate strategic significance can provide a monitoring system capable of detecting a theft of material within approximately 2 hours of the actual removal of the material. Either detection devices or security procedures may be used to help detect unauthorized activities or penetration of CAAs. Most of the sites at which this material is stored or used are likely to be small research-oriented facilities such as nonpower reactors and educational institutions. At such facilities there are likely to be relatively small quantities of SNM available, and authorized access to it may be infrequent.

Detection devices such as interior motion detector systems, balanced magnetic switches, etc., could be used to monitor such relatively small CAAs where the number of established access control points is very limited.

Security procedures may also be used to protect against thefts of quantities of SNM of moderate strategic significance because watchman patrols inherently allow unguarded intervals during which an external adversary could conceivably obtain access to the CAA, remove a small quantity of material, and leave before being discovered. The inspection procedure may (1) be sufficiently frequent to ensure that an external adversary could not complete a successful theft of the material during the unguarded intervals, (2) provide for an indication of the CAA barrier having been tampered with (as with a seal or by observation of damage done to the barrier), or (3) provide for an item count inventory or other accounting for the material as part of the inspection. Some examples are given below:

- After having left material unattended in an experimental setup in a classroom used as a temporary CAA, the instructor returns after several hours and verifies that the material is still in place.

- A small quantity of material is stored in a security cabinet for several months at a time without authorized access. Every 2 hours a watchman comes around to inspect the cabinet to ensure that the lock is secure and the cabinet has not been tampered with. If a lock is used that is not designed to be resistant to surreptitious and forced entry, the watchman also inspects a tamper-indicating seal to verify that the container has not been tampered with.

- During periods when a nonpower reactor is left unattended, a watchman comes around every 2 hours and checks that none of the fuel elements

have been removed by unauthorized persons. If a person working during off-hours in the vicinity of the nonpower reactor is encountered, appropriate identification is requested.

- While a nonpower reactor is being used, the supervisor ensures that he/she or his/her designee is not away from the reactor for a period longer than 30 minutes or a period he/she estimates would be minimally required for a person to obtain fuel elements in the core that are not self-protecting by virtue of their having been sufficiently irradiated.

- During periods when excess fuel is left unattended in a storage closet at a nonpower reactor facility, an alarm system is activated that would annunciate if anybody tried to penetrate the closet without proper authorization.

During periods when the material is being used under the direct supervision of authorized personnel, the monitoring requirement may be satisfied by virtue of the continuous surveillance exercised by such personnel.

Content

For each of the CAAs designated for the protection of SNM of moderate strategic significance, provide the following information regarding the measures taken to fulfill the monitoring requirement of paragraph 73.67(d)(3):

1. Describe any devices employed by specifying the type of device, its installed location, the type and location of annunciation, the intended area of coverage, and the tamper-resistant features. Refer, if desired, to existing filed security plans, regulatory guides, or NUREG documents.

2. Describe any procedures employed to monitor the CAA or portions thereof, including the categories of persons who will execute the procedures, the frequency of inspections or rounds of patrol, the basis for the determination of such frequencies, and the occasions when the procedures are intended to be implemented. Also indicate the features of the CAA (e.g., barriers, locks, seals) that will affect the way the procedures are used.

3. Explain how the combination of procedures and devices used to monitor each CAA meets the criteria for early detection of theft by an external adversary relative to the type of material found in each CAA.

3. ACCESS CONTROL AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraphs 73.67(d)(4), (d)(5), (d)(6), (d)(7), and (d)(10), which are as follows:

(d)(4) Conduct screening prior to granting an individual unescorted access to the controlled access area where the material is used or stored, in order to obtain information on which to base a decision to permit such access,

(d)(5) Develop and maintain a controlled badging and lock system to identify and limit access to the controlled access areas to authorized individuals,

(d)(6) Limit access to the controlled access areas to authorized or escorted individuals who require such access in order to perform their duties,

(d)(7) Assure that all visitors to the controlled access areas are under the constant escort of an individual who has been authorized access to the area,

(d)(10) Search on a random basis vehicles and packages leaving the controlled access areas.

An access control system is one that controls access to or egress from a CAA through normal routes for personnel, materials, or vehicles.

3.1 Preauthorization Screening [73.67(d)(4)]

Intent

The intent of the requirement for preauthorization screening is to ensure that the licensee will have sufficient knowledge of an individual to determine his/her reliability and need for access prior to granting him/her authorized access to the CAA where the material is used or stored. The selection of procedures for conducting this examination and the criteria employed to make these judgments are the responsibility of the licensee and, of course, should be consistent with all local, State, and Federal laws and regulations regarding the protection of the privacy and other rights of the individual. The screening process may be conducted in the same manner as other investigations customarily conducted by potential employers for similarly sensitive positions. There is no requirement for the licensee to arrange for an NRC clearance or similar clearance from any other government organization. Examples of procedures and criteria that may be employed in the screening process include holding or having recently held a government-sanctioned clearance; examination of past employment or educational records (to determine any unsatisfactory employment or school actions or incidents that would indicate any unreliability or previous breaches of trust between the individual and his/her employer); endorsements or references from previous employers, teachers, or colleagues that would support a decision for granting access or that would attest to the trustworthiness and reliability of the individual; and consideration of the individual's present employment record indicating demonstrated trustworthiness

and reliability over an extended period of employment with the licensee. (This may be considered in the nature of "grandfathering.")

Content

Describe the procedures and criteria that will be used for obtaining sufficient information prior to making a decision on granting unescorted access authorization to an individual to CAAs where the material is used or stored. Identify the types of individuals who will be screened (e.g., process engineers, supervisory personnel, professors, instructors, graduate students) and who will perform the screening process.

3.2 Badging System [73.67(d)(5)]

Intent

The purpose of the badging system is to facilitate the identification of authorized individuals and the control of access to or within the CAA where the material is used or stored. Information on the badge should be such that it is possible to clearly distinguish personnel authorized for access to the CAAs from those requiring an escort. Information on the badge should also uniquely identify the individual possessing the badge. This personalized information can be obtained through the use of photographs, personal vital statistics, signatures, or any means the licensee may wish to use that will uniquely identify the individual.

Content

Describe the badging system used to facilitate control of access to the CAAs. This description should include:

1. The size, shape, color, material, and construction of badges.

2. The distinguishing features of the badge that identify authorized individuals from escorted individuals.

3. How the badges will be used for controlling access. (For example, will all individuals be checked prior to entering the CAAs, will periodic checks be made of individuals within a CAA to determine if they are authorized or under escort, or will the badge itself permit authorized entrance, e.g., a card key.)

4. The system used for issuing, controlling, and accounting for the badges.

3.3 Lock System [73.67(d)(5)]

Intent

Locks used to control access to CAAs should be resistant to manipulation or picking and should not be mastered. Examples of typical lock systems that fit this description are three-position dial-type combination locks, six-pin key locks, and card-key lock systems. The procedures for assigning keys and combinations to individuals is an integral part of the lock system and should

be designed to ensure that only authorized personnel have access to such items. Locks and combinations should be changed when information is obtained that the lock system may have been compromised. Further information may be obtained in Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials."

Content

Describe the locking system used to control access to the CAAs where material is used and stored. This description should include locations of all locks included in the system by type of lock, the pick-resistant and manipulation-resistant characteristics of each lock type used, personnel responsible for issuing keys or combinations and changing combinations or locks, criteria for changing combinations or locks, personnel authorized to be given keys or combinations, and descriptions of types of locks used (references may be made to Regulatory Guide 5.12 for this purpose).

3.4 Personnel Entry Control System [73.67(d)(6)]

Intent

The success of other access control system components such as preauthorization screening, badging, and lock control is dependent upon effective control of personnel access into the CAA. Physical access may be controlled in a number of different ways depending on the actual configuration of the CAA and other site-specific factors. Some examples of these alternatives are:

1. Control by Authorized Person. If the area to be controlled is sufficiently small and free of obstructions, an authorized person performing other activities in a CAA may effect physical access control by monitoring entry of unauthorized persons into the area. A sign posted at the entrance would help deter casual passersby. An example of this approach would be when laboratory instructors are conducting classes and, because they are familiar with each of their students, they could easily recognize unauthorized persons not belonging in the classes.

2. Card-Key, Cipher, Combination, or Key-Lock Control System. A more sophisticated hardware-oriented system involves the use of a card-key, cipher, combination, or key-lock system. Physical access control in this case consists of the use of physical barriers to deter unauthorized persons. A limited number of entrances that are controlled by authorized personnel using a card key, cipher, combination, or key are provided. This system may be more useful when larger numbers of authorized personnel who would not necessarily be familiar with one another need to share the use of the CAA.

3. Control by Security Organization. If security organization personnel are available, physical access control may be accomplished by stationing a person at the entrance to the CAA to check identification and allow only authorized persons into the CAA. This alternative may be unjustifiably expensive unless the security organization member's salary can be justified on other grounds as well. A variation of this system requires persons seeking entrance to the CAA to obtain a key from a properly designated person or security organization for each use.

Content

Describe the system for limiting physical access to each CAA identified in Sections 1.1 and 1.2 of Part I to authorized personnel or those escorted by authorized personnel. Include in this description the names or titles of individuals granting access authorizations, the criteria to be used in granting authorizations, and the procedures used to ensure that only authorized or properly escorted persons are allowed access to the CAA. Reference may be made to Sections 3.1, 3.2, 3.3, 3.5, and 3.6 of this chapter as they apply to this section for the description of locks, barriers, or other hardware that are used to control access.

3.5 Escort System [73.67(d)(7)]

Intent

The requirement that an escort system be established is in recognition of the fact that the licensee may wish to allow access to certain persons or classes of persons on a temporary or infrequent basis or on short notice, thus making the routine process for granting access authorizations impractical or inexpedient. Typical arrangements for escorted access may include escorts for maintenance or repair personnel, laboratory classes, public tours, guests, and visitors as required.

Content

Describe the system that will be used to escort individuals in the CAAs. In its security plan, the licensee should ensure that only properly authorized individuals will be allowed to escort individuals. This description should include:

1. Criteria to be used for granting escorted access,

2. Criteria to be used for escorting others,

3. Procedures for escorting individuals into CAAs (e.g., students under the supervision of laboratory instructor, public tours),

4. The number of escorted individuals per escort, and

5. The responsibilities of the escort (e.g., periodic surveillance of all individuals under escort, accounting for all material prior to leaving the CAA, remaining in general area during the time unauthorized individuals are present).

3.6 Search [73.67(d)(10)]

Intent

The primary intent of the search requirement is to deter and possibly detect attempted thefts of SNM. The search procedures developed by the licensee should take into consideration the environs where the material is used

or stored, the physical characteristics of the material itself, and the frequency of accounting for the material. In some cases, this will require that all vehicles and packages leaving the CAAs be searched in a random manner. The frequency of random searches should be determined by the ease with which the material can be stolen and the length of time it would take to detect a theft. In other cases, only packages that equal or exceed the size of the material being used or stored would have to be searched, taking into consideration the difficulty with which the material could be broken into smaller, more easily concealed parts.

Content

Describe the system to be used for randomly searching vehicles or packages that leave the CAA. Include in the description information as to:

1. The scope of the search. This should identify the criteria that will be used for searching vehicles and packages (e.g., whether all packages and vehicles are subject to search or just those packages or vehicles that are larger than the smallest configuration of material being used or stored).

2. The randomness of the search. The scheme for selecting the packages or vehicles to be searched should be identified (e.g., subjecting each package or vehicle to a search, using a random number generator for determining whether a candidate package or vehicle is to be searched, searching a minimum percentage of all packages or vehicles leaving the CAA each day).

4. SECURITY ORGANIZATION AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraph 73.67(d)(8), which states:

(d)(8) Establish a security organization or modify the current security organization to consist of at least one watchman* per shift able to assess and respond to any unauthorized penetrations or activities in the controlled access areas.

Intent

The intent of this requirement is to ensure that, in the event of a security incident, someone will be available to assess alarms or other unauthorized penetrations or activities and, if warranted, notify the NRC, the local law enforcement authorities, and the responsible person in licensee management. Early detection and notification of any missing material will help facilitate its prompt recovery. In some cases, the licensee may assign additional duties to members of the security organization where procedure-oriented options are chosen to satisfy physical protection requirements (e.g., periodic patrols and inspections of CAAs for storage of SNM). Security organization members are not required to be fully dedicated full-time employees of the licensee. They may include unarmed campus security personnel (watchmen), contract guards, members of the local law enforcement agency (if sufficiently close to the site), etc. No formal or comprehensive training program is required for security organization personnel. However, the licensee should be prepared to demonstrate that each security person understands the particular duties assigned to him/her and is fully qualified and trained to perform them.

Content

Describe the security organization that will be responsible for assessing and responding to security incidents. Indicate the other responsibilities of the security organization such as:

1. Conducting periodic physical security checks of CAAs,

2. Maintaining liaison with the local law enforcement agency,

3. Notifying the local law enforcement agency of any unauthorized penetrations or activities in the CAAs, and

4. Notifying licensee management of any unauthorized penetrations or activities in the CAAs.

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*A "watchman" is defined in paragraph 73.2(d) of 10 CFR Part 73 as "an individual, not necessarily uniformed or armed with a firearm, who provides protection for a plant and the special nuclear material therein in the course of performing other duties."

5. COMMUNICATIONS AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraph 73.67(d)(9), which states:

(d)(9) Provide a communication capability between the security organization and appropriate response force.

Intent

The intent of this regulation is to ensure that a communication capability exists between the licensee and the designated response force. It is implied that, prior to setting up a communication capability, procedures and responsibilities will have been established between the response force and the licensee. (See Chapter 6, "Response Procedures at a Fixed Site," of Part I.) The type of communication system chosen by a licensee should:

1. Provide for full duplex voice communication capability,

2. Be easily accessible to the licensee's security organization, and

3. Be reliable and available for immediate use at any time.

Some communication systems that would provide these capabilities include a dedicated telephone system, a nondedicated public telephone system, radio, or any combination thereof.

Content

Describe the communication system that is used between the security organization and the appropriate response force. This description should include information on:

1. Type of communication system,

2. Location of voice terminals in relationship to CAAs,

3. Availability of communication system on a 24-hour basis, and

4. Reliability of communication system.

6. RESPONSE PROCEDURES AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraph 73.67(d)(11), which states:

(d)(11) Establish and maintain response procedures for dealing with threats of thefts or thefts of such materials.

Intent

The intent of this regulation is to help the licensee to identify those security incidents that could result in the loss of SNM of moderate strategic significance and to develop response procedures to prevent or reduce the likelihood of such a loss. Some types of incidents that should be considered and for which response procedures should be developed are:

1. Situations that could possibly lead to theft of SNM (e.g., civil strife),

2. Discovery that the security system has been breached, and

3. Discovery that some SNM is missing.

Content

Identify those events for which response procedures will be developed. Also, describe the type of response to be accomplished for each event identified and the duties and responsibilities of the security organization and management involved in the response. Ensure that the NRC will be notified immediately in the event of theft or attempted theft of the material. Describe what local law enforcement assistance is available, their response capabilities, and any agreements made with them to respond in the case of theft of the material.

7. MATERIAL TRANSPORTATION REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(1), which are as follows:

(e)(1) Each licensee who transports, exports, or delivers to a carrier for transport special nuclear material of moderate strategic significance shall:

(i) Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification,

(ii) Receive confirmation from the receiver prior to the commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport,

(iii) Check the integrity of the container and locks or seals prior to shipment, and

(iv) Arrange for the in-transit physical protection of the material in accordance with the requirements of paragraph 73.67(e)(3) of this part unless the receiver is a licensee and has agreed in writing to arrange for the in-transit physical protection.

7.1 Advance Notification to Receiver [73.67(e)(1)(i)]

Intent

The intent of this paragraph is to require the shipper to preplan the transportation of material and inform the receiver of his plans prior to shipment. This is the first of the several transportation requirements that will allow the receiver to take delivery of the material as planned or to help ensure traceability of any missing material.

Content

The licensee should ensure in his security plan that, prior to each shipment of material, the receiver will be notified of the impending shipment and provided the following types of information:

1. Mode of transport (e.g., truck, plane, train, or ship),

2. Estimated time of arrival,

3. Location where custody of the material will be transferred to the receiver,

4. Name of carrier, and

5. Transport identification (e.g., truck, train, or flight number; ship name).

7.2 Receiver Confirmation [73.67(e)(1)(ii)]

Intent

The intent of this requirement is that, prior to shipment, the shipper will be assured that the receiver is ready to accept the shipment at the planned time and location and has acknowledged the mode of transport.

Content

Describe what procedures will be used to ensure that shipment of material does not take place until the receiver acknowledges the planned shipment and mode of transport and states that he will be ready to accept the shipment at the planned time and location.

7.3 Inspection [73.67(e)(1)(iii)]

Intent

The intent of this paragraph is to provide a means for confirming the integrity of a shipment through inspections just prior to commencement and during the shipment by ensuring that the material containers and any associated locks or seals are intact at the time the shipment commences.

Content

Describe the procedures to be used to ensure that the integrity of the shipment containers and associated locks or seals is checked just prior to shipment.

7.4 Responsibility for In-Transit Physical Protection [73.67(e)(1)(iv)]

Intent

The intent of this paragraph is to make clear that the licensee shipping the material is responsible for arranging for the physical protection of the material in transit if the receiver is not a licensee. If both the shipper and receiver are licensees, the shipper may allow the receiver to accept this responsibility wholly or in part, provided there is appropriate documentation specifying their respective responsibilities. Where no such documentation exists, both the shipper and receiver are held jointly responsible. (See Section 8.3 of Part I of this guide.)

Content

In its security plan, the shipper should either acknowledge responsibility for the in-transit physical protection of SNM of moderate strategic significance or ensure that a written agreement from the receiver licensee has been received in which the receiver accepts either full responsibility or shared responsibility for the in-transit physical protection of this material in accordance with paragraph 73.67(e)(3) of 10 CFR Part 73.

8. RECEIVER REQUIREMENTS--TRANSPORTATION

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(2), which are as follows:

(e)(2) Each licensee who receives special nuclear material of moderate strategic significance shall:

(i) Check the integrity of the containers and seals upon receipt of the shipment,

(ii) Notify the shipper of receipt of the material as required in Section 70.54 of Part 70 of this chapter, and

(iii) Arrange for the in-transit physical protection of the material in accordance with the requirements of paragraph 73.67(e)(3) of this part unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection.

8.1 Inspection [73.67(e)(2)(i)]

Intent

This requirement is intended to determine whether the material's container has been compromised en route and whether any material has been removed so that immediate recovery procedures may be initiated if required.

Content

Describe the procedures to be used to ensure that the integrity of the containers and seals will be checked upon receipt of the shipment of material.

8.2 Notification to Shipper [73.67(e)(2)(ii)]

Intent

This requirement is intended to provide the shipper with knowledge of when responsibility for the shipment has been assumed by the receiver and when the shipper's responsibility is ended.

Content

Ensure that the receiver will send a completed copy of standard Form NRC-741, "Nuclear Material Transaction Report," to the shipper within 10 days of receiving a shipment of material as required in § 70.54 of 10 CFR Part 70.

8.3 Responsibility for In-Transit Physical Protection [73.67(e)(2)(iii)]

Intent

The intent of this paragraph is to make clear that the licensee receiving the material is responsible for arranging for the physical protection of the

material in transit if the shipper is not a licensee. If both the shipper and receiver are licensees, the receiver may allow the shipper to accept this responsibility either wholly or in part provided there is appropriate documentation specifying their respective responsibilities. Where no such documentation exists, both the shipper and receiver are held jointly responsible. (See Section 7.5 of Part I of this guide.)

Content

In its security plan, the receiver should either acknowledge responsibility for the in-transit physical protection of SNM of moderate strategic significance or ensure that a written agreement from the shipper has been received in which the shipper accepts either full responsibility or shared responsibility for the in-transit physical protection of this material in accordance with paragraph 73.67(e)(3) of 10 CFR Part 73.

9. IN-TRANSIT PHYSICAL PROTECTION REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(3), which are as follows:

(e)(3) Each licensee who arranges for the in-transit physical protection of special nuclear material of moderate strategic significance or who takes delivery of this material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall:

(i) Arrange for telephone or radio communications between the transport and the licensee or its designee: (A) to periodically confirm the status of the shipment, (B) for notification of any delays in the scheduled shipment, and (C) to request appropriate local law enforcement agency response in the event of an emergency.

(ii) Minimize the time that the material is in transit by reducing the number and duration of nuclear material transfers and by routing the material in the most safe and direct manner,

(iii) Conduct screening of all licensee employees involved in the transportation of the material in order to obtain information on which to base a decision to permit them control over the material,

(iv) Establish and maintain response procedures for dealing with threats of thefts or thefts of such material,

(v) Make arrangements to be notified immediately of the arrival of the shipment at its destination, or of any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination,

(vi) Initiate immediately a trace investigation of any shipment that is determined to be lost or unaccounted for after a reasonable time beyond the estimated arrival time.

(vii) Notify immediately the Director of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A to this part of the action being taken to trace the shipment.

9.1 Communications [73.67(e)(3)(i)]

Intent

The primary intent of this requirement is to ensure that procedures are established or facilities made available to permit communications to take place between the transport and the licensee or its designee for the purposes of (1) periodically confirming the status of the shipment while it is in progress, (2) notifying the licensee of delays along the shipment route that may affect the shipment's established schedule, and (3) requesting assistance from local

law enforcement authorities (LLEAs). These communications need not be direct between the licensee and the transport.

1. Confirmation of Shipment Status

Communications for confirming the status of a shipment should be timed to reflect the expected changes in shipment status indicated by the shipment's scheduled itinerary. For general motor freight (involving less-than-truckload quantities), direct communications between the transport and the licensee may be impractical. This requirement may then be satisfied by periodically telephoning the carrier to obtain updated reports on the shipment's progress based on calls the carrier receives from drivers on a regular basis. The frequency with which the licensee should check the status of a shipment should normally be at least every 10 to 14 hours. This corresponds to the frequency with which the drivers normally would call in. Similar indirect communications may be used to satisfy this requirement in the case of air shipments, but they should be more frequent to reflect the shorter air travel times between terminals. When direct radio or telephone communications are practical between the licensee and the transport, the frequency of calls should normally be approximately every 12 hours.

2. Notification of Shipment Delays

Notifications to the licensee should be made by the carrier or transport personnel whenever unforeseen conditions arise that threaten to delay the shipment significantly beyond the estimated arrival time at the next point on the shipment's itinerary. This notification may be made by conventional telephone lines or by radio. Installation of radiotelephone lines or comparable equipment is not required. These communications need not be direct between the transport and the licensee or its designee. The transport personnel may relay the information to the licensee through the carrier's offices.

3. Requests for LLEA Assistance

Procedures or facilities are also required to facilitate the transport personnel in obtaining assistance from LLEAs. Use of radiotelephone or citizens' band radio equipment for this purpose is acceptable but not required. The transport personnel may alternatively use conventional telephone lines. Requests for assistance should be in accordance with the response procedures developed to comply with paragraph 73.67(e)(3)(iv).

Content

Describe the communications facilities and procedures to be used to ensure that communications are established to achieve the three purposes enumerated in paragraph 73.67(e)(3)(i). The names and telephone numbers of responsible individuals should be given where appropriate. Refer to established response procedures (Section 9.4 of Part I of this guide) as necessary.

9.2 Minimum Transit Times [73.67(e)(3)(ii)]

Intent

This requirement is intended to have the shipper or receiver make a reasonable effort to ship the material by the fastest and most direct method possible. It is not intended to require use of dedicated transports or other expensive modes of travel.

Content

Describe the procedures and considerations that apply in the transportation planning process to ensure that a determined effort will be made to minimize transit times.

9.3 Preauthorization Screening [73.67(e)(3)(iii)]

Intent

The intent of the requirement for preauthorization screening is to ensure that the licensee will have sufficient knowledge of an individual to determine his/her reliability and need for access prior to granting him/her authorized access to the material in transit. The selection of procedures for conducting this examination and the criteria employed to make judgments are the responsibility of the licensee and, of course, should be consistent with all local, State, and Federal laws and regulations regarding the protection of the privacy and other rights of the individual. The screening process may be conducted in the same manner as are other investigations customarily conducted by potential employers for similarly sensitive positions. There is no requirement for the licensee to arrange for an NRC clearance or similar clearance from any other government organization. Examples of procedures and criteria that may be employed in the screening process include holding or having recently held a government-sanctioned clearance; examination of past employment records (to determine any unsatisfactory employment or incidents that would indicate any unreliability or previous breaches of trust between the individual and his/her employer); endorsements or references from previous employers or colleagues that would support a decision for granting access or that would attest to the trustworthiness and reliability of the individual; and consideration of the individual's present employment record indicating demonstrated trustworthiness and reliability over an extended period of employment with the licensee. (This may be considered in the nature of "grandfathering.")

Content

Describe the procedures that will be used for obtaining sufficient information prior to making a decision on granting unescorted access authorization to those licensee employees who will be directly involved in the transportation or in the planning and movement control of the material. Identify by title or name those employees who will be screened and those who will perform the screening process.

9.4 Response Procedures [73.67(e)(3)(iv)]

Intent

The intent of this regulation is to help the licensee to identify those transportation incidents for which notification might be expected and that might affect the security of the SNM in transit and to plan response procedures for such situations. For example, if the shipper is informed by the carrier that adverse weather conditions have temporarily prevented further progress of the shipment, the licensee should inform the receiver of a new estimated time of arrival.

Content

Identify those events for which response procedures will be developed. Also, describe types of response to be accomplished for each event identified and the duties and responsibilities of members of the security organization and management for dealing with the response. Ensure that the NRC will be notified immediately in the event of theft or attempted theft of the material.

9.5 Notification [73.67(e)(3)(v)]

Intent

The intent of this requirement is to ensure that the licensee responsible for the physical protection of SNM in transit will have a firm basis for deciding whether or not to initiate response procedures in the event a shipment becomes overdue or is lost.

Content

Describe the arrangements and procedures that will be used for notifying the licensee who arranges for the physical protection of material in transit of the arrival of the shipment at its destination or of any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination.

9.6 Lost Material Notification [73.67(e)(3)(vi)]

Intent

The intent of this requirement is to ensure that, in the event a shipment becomes overdue at its destination or at a scheduled stop on its itinerary and no reasonable explanation has been received by the licensee from the carrier regarding its status, a trace investigation will be initiated.

Content

Describe what procedures will be used to trace any shipment that is lost or has not arrived at a particular point on the shipment's itinerary or at its final destination by the estimated arrival time.

9.7 NRC Notification [73.67(e)(3)(vii)]

Intent

The intent of this requirement is to ensure that after a reasonable amount of time has elapsed with no explanation of the cause for the shipment's being overdue, the NRC will be immediately notified that the shipment is missing or unaccounted for and advised of the actions being taken to locate the missing shipment. The amount of time considered reasonable before these tracing and notification procedures are initiated should not exceed 1 hour in any case.

Content

Ensure that all material determined to be lost or unaccounted for will be reported immediately to the appropriate NRC Regional Office in accordance with § 73.71. Ensure that notification includes specifying what actions are being taken to trace the shipment and that the shipper or receiver, as appropriate, will also be notified.

10. TRANSFER AND CONTROL REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(4), which reads as follows:

(e)(4) Each licensee who arranges the physical protection of strategic special nuclear material in quantities of moderate strategic significance while in transit or who takes delivery of this material free on board (f.o.b.) the point at which it is delivered to a carrier for transport, shall, in addition to the requirements of (e)(1), (e)(2) and (e)(3) above:

(i) Make all shipments of the material either (A) in dedicated transports with no intermediate stops to load or unload other cargo and with no carrier or vehicle transfers or temporary storage in transit, or (B) under arrangements whereby the custody of the shipment and all custody transfers are acknowledged by signature, and

(ii) Maintain the material under lock, or under the control of an individual who has acknowledged acceptance of custody of the material by signature.

10.1 Carrier Transfers (73.67(e)(4)(i))

Intent

The intent of this requirement is to ensure continuity in the assignment and acceptance of personal responsibility for shipments of strategic special nuclear material (high-enriched uranium and plutonium). The requirements of paragraph 73.67(e)(4) do not apply to LEU. This continuity may be provided by (1) shipping the material in a dedicated transport vehicle so that the material would be under the continuous control of the same individual(s) for the duration of the shipment or (2) arranging for each individual who assumes custody of the shipment to indicate acceptance of personal responsibility for the material by signature.

Shipment of the material in a dedicated transport means there should be no intermediate stops to load or unload other cargo and no carrier or vehicle transfers or temporary storage en route. However, material other than the SNM being shipped may be carried in the transport's cargo compartment provided it does not detract from the security of the shipment.

For shipments by general freight, this requirement may be satisfied by arranging for "signature security service" with a high surveillance feature, which is provided by many air and motor freight carriers.

Content

Indicate how this requirement would be met for each of the modes of shipment to be employed. Indicate how this requirement would continue to be met in emergency situations by reference to the emergency response plans

included in Section 9.4 of the plan. If signature service is to be employed, describe the type of service that will be provided by the particular carrier chosen for the shipment. If a dedicated transport is to be used to satisfy this requirement, affirm that the same individual(s) will maintain responsibility for the shipment from the point of origin to its destination.

10.2 Control of Shipments (73.67(e)(4)(ii))

Intent

The intent of this requirement is to ensure that SSNM is not misrouted or mishandled by unauthorized individuals during the course of a shipment and to facilitate early detection and recovery of the material in case of loss or theft. This can be achieved by maintaining the material under lock or under the control of a responsible individual whenever the material is not secured in a locked compartment.

Maintaining the shipment in a locked compartment would normally be possible only for shipments in dedicated transports. In this case, the cargo compartment should be locked upon embarkation and unlocked only after arrival at the shipment's destination. If the locked cargo compartment is left unattended for an extended period that might allow the unauthorized removal of the material, the individual(s) responsible for the shipment should establish upon returning to the transport that the compartment has remained continuously under lock. This can be done by examining the cargo compartment lock for signs of tampering, examining seals on the cargo compartment, or opening the compartment and ensuring that the material is intact.

For general freight shipments, during which the cargo compartment may be opened frequently to load or unload other cargo, locking of the compartment may not be practical on a continuous basis. When the shipment is not maintained under lock, the material is required to be in a sealed cargo compartment or maintained under the control of an individual who has indicated by signature his/her acceptance of responsibility for the shipment. This individual should be the same person who has signed for custody of the shipment in compliance with paragraph 73.67(e)(4)(i). For air shipments in which the cargo compartments are not locked, an individual responsible for the shipment is not required to accompany the shipment during flight provided a responsible individual maintains control of the shipment during the entire time the shipment remains at each air terminal. During unscheduled stops at air terminals not on the shipment's itinerary, the shipment may be considered to be under the control of the flight crew provided the shipment is not offloaded and a member of the flight crew remains with the aircraft at all times.

Maintaining the shipment under control means (1) ensuring the shipment is properly routed at transfer points, (2) keeping the material under lock or under surveillance, which would include periodically (normally at least every 2 hours) checking the shipment to ensure it has not been misplaced or tampered with, and (3) ensuring that the licensee or its designee is notified immediately if the shipment is determined to be missing. The licensee should ensure that the signature security service or other service provided by the

carrier under normal or special arrangements includes all three of the these elements of control.

Content

Indicate the periods during which the shipment will be maintained under lock and those during which it will be maintained under the control of a responsible individual. Describe the instruments to be used for documenting acknowledgment of custody of the material by individuals assigned responsibility for controlling the shipment. Also, describe the arrangements made with carriers to ensure that the individuals they assign to maintain control of the material do so in accordance with the three elements of control described above. Refer to the emergency response procedures developed in accordance with paragraph 73.67(e)(3)(iv) as necessary.

11. EXPORT REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(5), which reads as follows:

(e)(5) Each licensee who exports special nuclear material of moderate strategic significance shall comply with the requirements specified in § 73.67(c), (e)(1) and (e)(3) and (e)(4).

Use Chapters 7, "Material Transportation Requirements," 9, "In-Transit Physical Protection Requirements," and 10, "Transfer and Control Requirements," of Part I of this guide to describe the security procedures that will be used to protect the material up to the point where the receiver accepts physical protection responsibility for the shipment.

12. IMPORT REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(6), which reads as follows:

(e)(6) Each licensee who imports special nuclear material of moderate strategic significance shall:

(i) Comply with the requirements specified in § 73.67(c), (e)(2), (e)(3) and (e)(4), and

(ii) Notify the exporter who delivered the material to a carrier for transport of the arrival of such material.

12.1 Security Requirements [73.67(e)(6)(i)]

Use Chapters 8, "Receiver Requirements--Transportation," and 9, "In-Transit Physical Protection Requirements," of Part I of this guide to describe the security procedures that will be used to protect the material from the first point at which the shipment is picked up inside the United States.

12.2 Notification [73.67(e)(6)(ii)]

Intent

The intent of this requirement is to ensure that the exporter is notified that the material has arrived safely.

Content

Describe the procedures to be used for notifying the exporter of the material that the shipment has been received.

13. ORDERS TO DELAY SSNM SHIPMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(e)(7) and § 73.72 (as it may apply to shipments of special nuclear material of moderate strategic significance). These requirements read as follows:

(e)(7) If, after receiving advance notice pursuant to § 73.72 from a licensee planning to import, export, transport, deliver to a carrier for transport in a single shipment, or take delivery at the point where it is delivered to a carrier, special nuclear material of moderate strategic significance containing in any part strategic special nuclear material, it appears to the Commission that two or more shipments of special nuclear material of moderate strategic significance, constituting in the aggregate an amount equal to or greater than a formula quantity of strategic special nuclear material, may be en route at the same time, the Commission may order one or more of the shippers to delay shipment according to the following provisions:

(i) The shipper shall provide to the Commission, upon request, such additional information regarding a planned shipment as the Commission considers pertinent to the decision on whether to delay such shipment.

(ii) The receiver of each shipment, or the shipper if the receiver is not a licensee, shall notify the Director of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A by telephone, no later than 24 hours after arrival of such shipment at its final destination, or after such shipment has left the United States as an export, to confirm the integrity of the shipment at the time of receipt or exit from the United States.

(iii) The Commission shall notify the affected shippers no later than two days before the scheduled shipment date that a given shipment is to be delayed.

(iv) Shipments of special nuclear material of moderate strategic significance which are protected in accordance with the provisions of §§ 73.20, 73.25, and 73.26 shall not be subject to orders to delay shipment nor considered to constitute a portion of an aggregate formula quantity of strategic special nuclear material for the purposes of determining whether any shipments must be delayed.

§ 73.72 Requirement for advance notice of shipment of special nuclear material.

(a) Each licensee who plans to import, export, transport, deliver to a carrier for transport in a single shipment, or take delivery, at the point where it is delivered to a carrier, the following materials, shall notify the

Director of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A by U.S. Mail, postmarked at least 7 days in advance of the shipping date: (1) formula quantities of strategic special nuclear material, or (2) special nuclear material of moderate strategic significance containing in any part strategic special nuclear material.

(b) The following information shall be furnished in the advance notice: shipper, receiver, carrier(s), estimated dates and time of departure and arrival, transfer point(s), and mode(s) of shipment.

(c) The Director of the appropriate Nuclear Regulatory Commission Regional Office shall also be notified by telephone 7 days in advance of the shipping date that an advance shipping notice has been sent by mail, and of any changes to the shipment itinerary prior to the shipment date. Road shipments or transfers with one-way transit lines of 1 hour or less in duration between installations of a licensee are exempt from the requirements of this section.

It is noted that shipments of SNM of moderate strategic significance consisting entirely of uranium enriched in the isotope U-235 to 10 percent or more but less than 20 percent are not subject to these requirements.

Of the requirements detailed above, only paragraph 73.67(e)(7)(ii) and § 73.72 should be addressed in the licensee's physical security plan. This is discussed in Section 13.1.

1. Criteria for Delaying Shipments

Under the provisions of paragraph 73.67(e)(7) of the regulations, the NRC has authority to order delays in certain shipments of SNM of moderate strategic significance. It is the intention of the NRC to exercise this authority only when necessary to protect the public health and safety from the possible consequences of the theft of a formula quantity of SSNM. Since the number of affected shipments expected annually is quite low, shipment delays are expected to be ordered very rarely. In most cases, licensees will be able to clear shipment dates with the NRC staff far in advance so that potential conflicts in the scheduling of shipments can be avoided. However, shipment delays may be required on an emergency basis if a previous shipment is determined to be missing and cannot be accounted for before the next scheduled shipment is to depart. When a scheduling conflict that cannot be resolved voluntarily arises, the following criteria will be applied to determine which shipments will be delayed:

a. Total time needed to complete the shipment,

b. The impact of a schedule delay on the use of the SNM at the delivery site,

c. Planned routing of the material,

d. The relationship of the material to the national defense or to other essential programs in the nation's interest,

e. The possible impacts on the shipper/carrier relationship with regard to availability of shipment vehicles and other special arrangements that would be difficult to reschedule, and

f. How far in advance of the proposed shipment date the NRC was notified of the intent to make the shipment.

2. NRC Requests for Additional Shipment Information

In order to apply the decision criteria listed above, the NRC has the authority under paragraph 73.67(e)(7)(i) to request additional information from licensees regarding a planned shipment of SSNM that may be subject to a possible delay order. Such information requests may be made at any time prior to a scheduled shipment either by telephone or in writing.

3. NRC Notification of Shipment Delay Orders to Licensees

Under paragraph 73.67(e)(7)(iii), the NRC will normally notify licensees that a shipment is to be delayed no later than 2 days before a shipment is scheduled to depart. In practice, it is expected that potential conflicts will be resolved much before the date of a planned shipment after appropriate NRC consultations with the prospective shippers. However, in an emergency the NRC may order delays in shipments or other responsible action at any time it may deem necessary or desirable to promote the common defense and security, protect health, or minimize danger to life or property.

13.1 Notification of Receipt of SSNM (73.67(e)(7)(ii))

Intent

The intent of this notification requirement is to ensure that the NRC is promptly informed of the receipt of a shipment of SSNM subject to paragraph 73.67(e)(7) or, in the case of export shipments, informed that the shipment has left the United States safely. Based on this information, the NRC can permit subsequent shipments subject to paragraph 73.67(e)(7)(ii) to commence with the knowledge that a formula quantity of SSNM will not be at risk in a small number of simultaneous shipments. This will also ensure that shipment delay orders will not have to be imposed unnecessarily for failure to confirm that a previous shipment is no longer at risk. Compliance with this requirement is not a substitute for compliance with other notification requirements under paragraph 73.67(e)(2)(ii).

Content

Affirm that, in the case of a shipment of SNM of moderate strategic significance containing in any part SSNM, arrangements are made to obtain confirmation of the integrity of the shipment at the point where the material leaves the United States as an export or at its final destination. Describe the procedures for accomplishing this and notifying the NRC of the shipment's status at the time of receipt or exit from the United States. Include the names or titles of individuals who will be responsible for ensuring that the required confirmation is made and the proper notification is provided to the NRC.

PART II

SPECIAL NUCLEAR MATERIAL

OF LOW STRATEGIC SIGNIFICANCE

1. USE AND STORAGE AREA AT A FIXED SITE

This chapter provides guidance on meeting the requirement of paragraph 73.67(f)(1), which is as follows:

(f)(1) . . Store or use such material only within a controlled access area.

A controlled access area (CAA) is defined in paragraph 73.2(z) as "any temporarily or permanently established area which is clearly demarcated, access to which is controlled and which affords isolation of the material or persons within it." "Access control" means measures used to allow only specified personnel, materials, and vehicles ingress into and egress from a given area, while "isolation" refers to measures taken to deter persons, materials, or vehicles from entering or leaving a given area through other than established access control points. Thus, a CAA includes provisions for both isolation and access control. In some cases, isolation or access control systems may also serve the purpose of aiding in the detection of unauthorized penetration or activities* within the CAA. Therefore, these detection-related considerations are also alluded to in this chapter.

In the discussion that follows, CAAs intended for the use and possible storage of SNM of low strategic significance are discussed separately from those intended solely for the storage of such SNM. Although there is no difference in the requirements for these two different applications of the CAA, it is recognized that the means used to isolate the material and to control access may differ markedly for use areas compared with storage areas and therefore warrant separate discussion.

1.1 Areas for Use and Temporary Storage [73.67(f)(1)]

This section discusses CAAs intended primarily for the use and temporary storage of SNM of low strategic significance. These may be temporarily established to meet transitory or intermittent SNM use requirements or they may be permanently established. Permanently established CAAs for use of SNM may also be used for storage for short intervals between periods of usage. "Use" means that the material is undergoing processing (e.g., fuel fabrication, irradiation in a reactor) or utilization of its properties in conjunction with experimental equipment (e.g., equipment used for research or educational laboratory experiments). Different isolation/access control measures may be used for periods during which the area is occupied versus unoccupied.

1. Temporarily Established CAAs

Temporarily established CAAs for the use of SNM need not have permanent barriers at their boundaries. Isolation of the material and persons using the material may be provided by office partitions, cordons, or other devices used to warn passersby of the restricted nature of the area. Access control can be effected through surveillance or supervision of the area by those who are using the SNM and who are responsible for the material. However, the provision of

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*Unauthorized activities are those activities deemed by the licensee to be indicative of or contributory to the possible theft of SNM.

access control through personal supervision is suitable only for those situations in which the size of the CAA and the number of persons to be admitted are sufficiently small and in which the CAA is suitably configured to make such procedures practical.

When material located in a temporarily established CAA is to be left unattended (i.e., because it is impractical to replace the material in a CAA designed for long-term storage of the material), access control and improved isolation for the temporary CAA may be provided to substitute for the discontinued personal supervision over the material. (In addition, provisions must also be made for satisfying the monitoring requirement of paragraph 73.67(f)(2).) Improved isolation can sometimes be provided by increasing the penetration resistance of existing barriers (e.g., locking doors and windows, placing the material in a locked drawer or supply room). Access control may be provided by any of the measures suggested below for permanently established CAAs left unattended; some of these suggestions may be more suitable for temporary use than others. If no suitable barrier is available to provide isolation, material left unattended may be protected by a motion alarm covering the area immediately surrounding the material. This improved detection capability would be considered an acceptable substitute for the rudimentary isolation provided by temporary barricades, cordons, signs, and other less substantial means of isolation that may be used on a temporary basis.

2. Permanently Established CAAs

Permanently established CAAs for the use and temporary storage of SNM of low strategic significance would most likely provide isolation for the SNM through the use of permanent barriers. These could consist of fences; gates or freestanding walls for exterior areas; or exterior or interior building walls, locked doors, windows, bars, grillwork; or other barriers for interior areas. Such barriers are not required to meet the more stringent criteria for physical barriers used for the protection of formula quantities of strategic special nuclear material (SSNM).* However, good security management practice would dictate that the barrier be substantial enough to deter casual passersby from unauthorized penetration. If the barriers are also designed to aid in detection, the criteria for penetration resistance or tamper indication in accordance with the specific monitoring procedures to be used also apply.

Access control for permanently established CAAs during periods they are occupied can be provided by personal supervision. However, when these areas are unoccupied or when the size, configuration, or numbers of persons intended to be admitted to the area make personal supervision impractical, other means of access control may be called for. Some of the additional access control measures that may be used for permanently established CAAs in these situations are:

- Stationing a watchman at CAA access control points,
- Limiting distribution of keys, keycards, or combinations to locks on doors and gates,

*"Strategic special nuclear material" means uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium.

- Using a coded badging system to identify authorized personnel at CAA access control points,

- Controlling locked CAA doors and gates by use of remote surveillance (e.g., intercom or CCTV) by personnel stationed at a centrally located access control facility, or

- Controlling access of personnel, material, and vehicles into CAAs by other means.

It is expected that large facilities (e.g., fuel processing facilities) will have more complex isolation and access control systems than small facilities (e.g., research facilities) in order to achieve comparable levels of protection.

Content

Describe the CAAs where the material will be used. Indicate under what conditions CAAs will be established on a temporary basis. State whether each CAA will also be used for storage. For each CAA identified, provide the following information:

1. A description of the area and its features relative to other facility features, showing the normal routes of ingress to each CAA (including a scale diagram).

2. Descriptions of physical barriers, cordons, walls, partitions, or other means of providing isolation of the CAA, and channelling entry through established access control points into the CAA.

3. The means and criteria used for controlling access into the CAA at established access control points.

4. If more than one CAA is designated, the types of material normally used in each CAA; for temporarily established CAAs, the types of activities normally performed within each area.

1.2 Areas for Permanent Storage [73.67(f)(1)]

Intent

CAAs normally used only for storage of SNM are described in this section. Such CAAs may be very limited in size since they need only contain the material itself, and access to them is expected to be infrequent. Where small amounts of material are involved, the CAA may consist of relatively small containers such as security cabinets, safes, and locked closets or supply rooms.

Isolation for CAAs intended only for storage usually will be provided by the penetration-resistant features of the perimeter barriers or container walls of the CAA. Such barriers and container walls are not required to meet the more stringent criteria for physical barriers used for the protection of formula quantities of SSNM. However, good security management practice would

dictate that the barrier be substantial enough to deter casual passersby from unauthorized penetration. If the barriers are also designed to aid in detection, the criteria for penetration resistance or tamper indication in accordance with the specific monitoring procedures to be used also apply.

In determining the level of isolation and access control for such CAAs, the minimum time required for extricating SNM from its storage location in relationship to the licensee's time of detection and assessment may need to be taken into account (e.g., fuel assemblies stored in shipping casks or non-self-protecting fuel residing in nonpower reactor cores). The bulkiness, massiveness, and difficulty of movement or disassembly of fresh fuel elements or other forms of low-enriched uranium (LEU) in large quantities such as shipping casks or assemblies may pose sufficient difficulty to persons attempting unauthorized removal within a reasonable time period (relative to CAA monitoring procedures) that they may be considered to provide isolation. Control of access to special equipment and vehicles capable of handling and transporting such items may also be included as integral elements of the access control program for the CAA.

Access control procedures for CAAs designated exclusively for storage are expected to be similar to those described in the previous section for periods when use-type CAAs are unoccupied.

It is expected that large facilities (e.g., fuel processing facilities) will have more complex isolation and access control systems than small facilities (e.g., research reactors) in order to achieve comparable levels of protection.

An illumination level is recommended for the CAA that is sufficiently uniform and bright to detect penetration of or tampering with the CAA containment (in the case of a security cabinet) or unauthorized penetration of or activities within the CAA (in the case of a vault-type room or its equivalent).

Content

(This section needs to be completed only for CAAs designated exclusively for storage of SNM. CAAs designated for both use and storage are to be described as recommended in Section 1.1 of Part II of this Standard Format.)

Describe the CAAs where the material will be stored, if different from those previously described for use. For each such CAA, provide the following information:

1. A description of the area and its features relative to other facility features showing the access control points for the CAA (including a scale diagram).

2. The physical barriers, container walls, or other features that demarcate the perimeter of the CAA and channel entry through only established access control points. If access control is provided by virtue of the bulkiness, massiveness, and difficulty of disassembly of a container or object

serving as a CAA, describe the facts relied upon to support such a determination.

3. The means and criteria used for controlling access to the CAA at established access control points.

4. If more than one CAA is designated for the storage of the material, the types of material normally to be stored in each area.

2. DETECTION DEVICES AND PROCEDURES AT A FIXED SITE

This chapter provides guidance for meeting the requirement of paragraph 73.67(f)(2), which is as follows:

(f)(2) Monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities.

The purpose of this monitoring activity as stated in paragraph 73.67(a)(2) is to provide "early detection and assessment of unauthorized access or activities by an external adversary within the controlled access area..." and "early detection of removal of special nuclear material by an external adversary from a controlled access area." This should be done to achieve the objective of minimizing the possibilities for unauthorized removal of special nuclear material "consistent with the consequences of such actions," as stated in paragraph 73.67(a) of the effective rule. Thus, the earliness of detection afforded by the licensee's physical protection system may vary depending on the nature and quantity of the material susceptible to unauthorized removal. Further discussion of the earliness of detection required for possible thefts of different types and quantities of special nuclear material of low strategic significance is provided in Section 2.1 of Part II of this guide.

2.1 Earliness of Detection [73.67(a)]

The criteria used for determining the earliness of detection for unauthorized access, activities, or removal of SNM from CAAs will depend on an assessment of the magnitude of the consequences associated with possible misuse of the type and quantity of material that could be removed jn a given theft attempt. For SNM of low strategic significance, three distinct cases are considered:

1. Gross theft of low-enriched uranium (LEU)*

2. Minor theft of LEU.

3. Theft of strategic special nuclear material (SSNM) (i.e., SNM of low strategic significance other than LEU).

Gross theft of LEU refers to the theft of LEU in a sufficiently large quantity that it could yield upon further enrichment or other processing enough material of the type and quantity needed to construct a clandestine fission explosive (CFE) device. This quantity may vary depending on the enrichment and physical and chemical form of the feed material but may roughly be estimated as an amount containing about 75 kilograms of the isotope U-235. Because the most serious consequences to the public health and safety could occur from a theft of this magnitude, a system that would detect a theft sufficiently early to allow for prompt recovery is called for. The NRC believes that, because of the

*Low-enriched uranium is uranium enriched above natural but less than 20 percent in the U-235 isotope.

large quantity of material involved and with proper detection systems and procedures, the licensee would be capable of detecting a gross theft of LEU between the time of the actual theft and the time it is transported off site.

Minor theft of LEU refers to thefts involving much smaller quantities of LEU such as could be removed by one or two persons in a private vehicle or on one's person. These minor thefts are significant only to the extent that they could be repeated periodically to eventually accumulate an aggregate quantity similar to that which might be obtained in a gross theft. The required degree of earliness of detection of such thefts is related to the time that would be needed by the adversary to obtain a gross amount through repeated thefts. If the amount taken in each of a series of daily thefts is very small, the ultimate time of detection could be as long as 6 months--the interval between material inventories as required by Part 70. For larger quantities that could be removed in a single theft, but still much less than would constitute a gross theft, it would be reasonable to expect that one in a series of such thefts would be detected before a gross amount could be removed.

Licensees who possess less than gross quantities of LEU need not provide for early detection of a single theft of a gross quantity, but the capability for detecting multiple thefts is as important as it is for those licensees possessing gross quantities.

Thefts from a single facility of SSNM (SNM of low strategic significance other than LEU) are limited by definition to quantities that could not be used to construct a nuclear explosive device; thefts of similar quantities of SSNM from several different facilities could, however, lead to the accumulation of an aggregate quantity that would permit such illicit use. (The consequences of a single theft of such material would have minimal impact on the public health and safety.) Therefore, detection of such thefts is required sufficiently early to ensure early notification to the NRC. The NRC in turn can then notify other licensees of the need to implement appropriate response measures to prevent the accumulation by a single adversary of a formula quantity of material through multiple thefts from different facilities. The NRC believes that there are detection systems and procedures that would allow the licensee to detect a theft of SSNM within approximately 2 hours.

2.2 Monitoring Controlled Access Areas [73.67(f)(2)]

Intent

Monitoring CAAs where the material is used will require a capability for early detection of (1) unauthorized penetration of the CAA perimeter, (2) unauthorized activities within the CAA, (3) unauthorized removal of SNM from the CAA, or (4) some combination of these, by an external adversary. For purposes of this guidance, an external adversary is any person not employed by the licensee or one of its contractors to report for work on a regular basis at the licensee's site and who is not directly involved in the processing or other authorized use of the material in connection with his/her assigned duties. For purposes of this definition, graduate and other students assigned by the licensee at an educational or research institution to assume responsibility for

the regular handling of the material would not be considered an external adversary.

The choice of particular safeguards measures to be included in the CAA monitoring system will depend on a number of different site-specific factors such as whether the measure is to function while the area is occupied or unoccupied, the size of the area, the types of functions performed in the area, the level of traffic into and out of the area, etc. Many of the available safeguards measures that may be employed in the CAA monitoring system are described in various regulatory guides and NUREG documents issued by the NRC. Specifically, the Fixed Site Physical Protection Upgrade Rule Guidance Compendium* provides guidance on selection, installation, and implementation of monitoring devices and procedures relative to physical protection of formula quantities of SSNM. This guidance is also applicable to monitoring equipment and procedures for CAAs at fixed sites having less than formula quantities of SNM with regard to individual physical protection measures. It is recognized, however, that from a systems point of view, this body of guidance intended to support the Upgrade Rule may emphasize a much greater level of redundancy and diversity than would be required for the protection of SNM of moderate and low strategic significance.** The following paragraphs give illustrations of various monitoring systems that can be used to provide the earliness of detection discussed in Section 2.1 of Part II of this guide.

1. Early Detection of Gross Theft of LEU

Early detection of gross theft of LEU means detection during the attempted theft. This may be accomplished by using intrusion alarms or other monitoring devices in the following applications:

- Fence or buried line perimeter detection systems at the CAA boundaries.
- Door and window alarms (e.g., balanced magnetic switches) for CAAs within buildings and alarms on emergency exit doors equipped with crash bars.
- Volumetric interior motion detection systems for unoccupied CAAs or portions of CAAs within buildings.

In many cases, security procedures may be employed as substitutes for some or all of the devices that could be employed to fulfill the monitoring requirement for the CAA. Security procedures may be especially attractive in protecting against the theft of gross quantities of LEU located outside

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*The compendium was issued in conjunction with the Physical Protection Upgrade Rule. It is intended to assist the licensee in the development and implementation of safeguards physical protection and transportation protection plans. Copies are available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

**Tamper-safing of devices intended for use at facilities having less than formula quantities of SSNM need protect only against the external adversary; to the extent such a distinction can be made for a given device, this represents a relaxation in tamper-safing requirements relative to protection of formula quantities.

buildings where intrusion alarms may more easily be circumvented by an external adversary. Typical security procedures that may be employed for detection of attempts at unauthorized removal of gross quantities of LEU include the following:

- Periodic patrols of CAA perimeters and areas within the CAA by watchmen.

- Surveillance (escorting) of nonlicensee vehicles capable of transporting gross quantities of LEU away from the site.

- Continuous (without interruption) or continual (intermittent) surveillance of the CAA, or a given portion of the CAA that contains a gross quantity of LEU, by designated supervisory or other licensee employees. (These surveillance responsibilities could be undertaken in addition to the employee's normal functions provided they can both be discharged adequately.)

- Periodic inspections to confirm continued integrity of barriers, gates, loading bay doors, etc., through which unauthorized removal of gross quantities of LEU could be effected.

- Inspection of large vehicles leaving CAAs to ensure they are not used for unauthorized removal of a gross quantity of LEU.

Where procedures involve periodic patrols or inspections, the amount of time that would be required for an adversary to complete the removal of a gross quantity of LEU may be used to govern the interval between inspections. For example, if a given number of a certain type of container of LEU would have to be individually loaded onto a truck in order to effect removal of a gross quantity of LEU and the time required for such loading is determined to be in excess of 2 hours, then a 2-hour guard patrol of the given area would be considered to meet the earliness-of-detection guideline for the subject material.

A combination of these and perhaps other measures could satisfy the early detection criterion for gross theft of LEU if it could be shown that an external adversary could not obtain unauthorized access to the SNM without being detected by at least one intrusion alarm component.

2. Early Detection of Minor Theft of LEU

The criteria for early detection of minor theft of LEU can be satisfied by a monitoring program that provides a sufficiently high probability of detection of one in a series of minor thefts so it is implausible that an adversary would be able to obtain in the aggregate a gross quantity of LEU. Such a capability could be provided in whole, or in part, by the same monitoring program designed to detect gross theft. Additional measures that might be included in the monitoring program to protect against minor theft include the following:

- Exit searches of nonlicensee vehicles capable of transporting hand-carried quantities of LEU (on a random or general basis).

- Administrative control of materials removed from controlled areas where easily concealed or hand-carried items containing LEU are available.

- Item count inventories more frequent than those currently required under Part 70.

- Procedures directing regular employees to be aware of unauthorized persons in their work areas and to report the presence of such persons to supervisory or security personnel. (A coded badging program would support this measure.)

- Escorting of visitors and other unauthorized persons entering CAAs.

- Watchman patrols to detect unauthorized persons within the CAA.

- SNM doorway monitors at all personnel access control points.

These are not required measures but illustrations of how the monitoring program for detecting gross theft of LEU may be extended to include the capability for early detection of minor theft of LEU. One or more of these measures may be employed, but probably not all of them would be needed for any one installation.

3. Early Detection of Theft of SSNM (SNM of low strategic significance other than LEU)

For facilities where it is necessary to protect against thefts of SNM of low strategic significance other than LEU (SSNM), earliness of detection is discussed in Section 2.1 of this chapter. The physical protection system in this case may differ considerably from that designed to protect LEU. The sites are likely to be small research-oriented facilities such as nonpower reactors and educational institutions. Also, there is likely to be much less material available, and authorized access to it may be less frequent. Detection devices such as interior motion detector systems, balanced magnetic switches, etc., could be used to monitor relatively small CAAs where the number of established access control points is very limited.

Procedures may also be used to protect against thefts of SSNM in this subcategory. Since watchman patrols inherently allow unguarded intervals during which an external adversary could conceivably obtain access to the CAA, remove a small amount of material, and leave before being discovered, the inspection procedure may (a) be sufficiently frequent to ensure an external adversary could not complete a successful theft of the material during the unguarded intervals, (b) provide for an irrevocable indication of the CAA barrier having been tampered with (as with a seal or by observation of damage done to the barrier), or (c) provide for an item count inventory or other accounting for the material as part of the inspection. Some examples are given below:

> After having left material unattended in an experimental setup in a classroom used as a temporary CAA, the instructor returns after 2 hours and verifies that the material is still in place.

- A small quantity of material is stored in a security cabinet for several months at a time without authorized access. Every 2 hours a watchman comes around to inspect the cabinet to ensure that the lock is secure and the cabinet has not been tampered with. If a suitable combination or other pick-resistant lock is not used, the watchman also inspects a tamper-indicating seal to verify that the container has not been tampered with.

- During periods when a nonpower reactor is left unattended, a watchman comes around every 2 hours and checks that none of the fuel elements have been removed by unauthorized persons. If a person working during off-hours in the vicinity of the reactor is encountered, appropriate identification is requested.

- While a nonpower reactor is being used, the supervisor ensures that he/she or his/her designee is not away from the reactor for a period longer than 30 minutes, a period he/she estimates would be minimally required for a person to secure fuel elements in the core that were not self-protecting by virtue of their having been sufficiently irradiated.

- During periods when excess fuel is left unattended in a storage closet at a nonpower reactor facility, an alarm system is activated that would annunciate if anybody tried to penetrate the closet without proper authorization.

During periods when the material is being used under the direct supervision of authorized personnel, the monitoring requirement is satisfied by virtue of the continuous surveillance by the authorized personnel.

Content

For each of the CAAs described previously (see Chapter 1 of Part II of this guide), provide the following information regarding the measures taken to fulfill the monitoring requirement of paragraph 73.67(f)(2):

1. Describe any devices employed by specifying the type of device, its installed location, the type and location of annunciation, its intended area of coverage, and its tamper-resistant features. Refer, if desired, to existing filed security plans, regulatory guides, or NUREG documents.

2. Describe any procedures employed to monitor the CAA or portions thereof, including the categories of persons who will execute the procedures, the frequency of inspections or rounds of patrol, the basis for the determination of such frequencies, and the occasions upon which the procedures are intended to be implemented. Also indicate the features of the CAA (e.g., barriers, locks, seals) that will play a role in the way the procedures are used.

3. Explain how the combination of procedures and devices used to monitor each CAA meets the criteria for early detection of theft by an external adversary relative to the type of material found in each CAA.

3. SECURITY RESPONSE AT A FIXED SITE

This chapter provides guidance on meeting the requirement of paragraph 73.67(f)(3), which is as follows:

(f)(3) Assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities.

Intent

The intent of this requirement is to ensure that, in the event of a security incident, someone will be available to assess alarms or any unauthorized penetrations or activities and, if warranted, notify the NRC, the local law enforcement authorities, and the responsible person in licensee management. Early detection and notification of any missing material will help facilitate its prompt recovery. For the purpose of this regulation, an offsite response force can be a local law enforcement agency or a contract guard service.

Content

Describe the security organization that will be responsible for assessing and responding to any unauthorized penetrations or activities. Ensure that at least one guard, watchman, or member of an offsite response force will respond to all unauthorized penetrations or security incidents at the CAAs.

4. RESPONSE PROCEDURES AT A FIXED SITE

This chapter provides guidance on meeting the requirements of paragraph 73.67(f)(4), which states:

(f)(4) Establish and maintain response procedures for dealing with threats of thefts or thefts of such material.

Intent

The intent of this regulation is to help the licensee to identify those security incidents that could result in the loss of SNM of low strategic significance and to develop response procedures to prevent or reduce the likelihood of such a loss. Some types of incidents that should be considered and for which response procedures should be developed are:

1. Situations that could possibly lead to theft of SNM (e.g., civil disturbance).

2. Discovery that the security system has been breached.

3. Discovery that some SNM is missing.

Content

Identify those events for which response procedures will be developed. Also describe the type of response to be accomplished for each event identified and the duties and responsibilities of the security organization and management involved in the response. Ensure that the NRC will be notified immediately in the event of theft or attempted theft of the material. Describe what local law enforcement assistance is available, their response capabilities, and any agreements made with them to respond in the case of theft of the material.

5. MATERIAL TRANSPORTATION REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(g)(1), which are as follows:

(g)(1) Each licensee who transports or who delivers to a carrier for transport special nuclear material of low strategic significance shall:

(i) Provide advance notification to the receiver of any planned shipments specifying the mode of transport, estimated time of arrival, location of the nuclear material transfer point, name of carrier and transport identification,

(ii) Receive confirmation from the receiver prior to commencement of the planned shipment that the receiver will be ready to accept the shipment at the planned time and location and acknowledges the specified mode of transport,

(iii) Transport the material in a tamper-indicating sealed container,

(iv) Check the integrity of the containers and seals prior to shipment, and

(v) Arrange for the in-transit physical protection of the material in accordance with the requirements of § 73.67(g)(3) of this part, unless the receiver is a licensee and has agreed in writing to arrange for the in-transit physical protection.

5.1 Advance Notification [73.67(g)(1)(i)]

Intent

The intent of this paragraph is to require the shipper to preplan the transportation of the material and inform the receiver of his plans prior to shipment. This is the first of several transportation requirements that will allow the receiver to take delivery of the material as planned or to help ensure traceability of any missing material.

Content

The licensee should ensure that, prior to each shipment of material, the receiver will be notified of the impending shipment and provided the following types of information:

1. Mode of transport (e.g., truck, plane, train, or ship),
2. Estimated time of arrival,
3. Location where material is to be transferred to receiver,
4. Name of carrier, and
5. Transport identification (e.g., truck, train, or flight number; ship name).

5.2 Receiver Confirmation [73.67(g)(1)(ii)]

Intent

The intent of this requirement is that, prior to shipment, the transporter will be assured that the receiver is ready to accept the shipment at the planned time and location and has acknowledged the mode of transport.

Content

Describe what procedures will be used to ensure that shipment of material does not take place until the receiver acknowledges the planned shipment and mode of transport and readiness to accept the shipment at the planned time and location.

5.3 Container [73.67(g)(1)(iii)]

Intent

The intent of this requirement is to provide a mechanism or system that will help the receiver detect any tampering with the material's container that may have occurred during shipment. Regulatory Guide 5.15, "Security Seals for the Protection and Control of Special Nuclear Material," provides guidance in this area. If the material is shipped in an exclusive-use carrier, it is acceptable to tamper-seal the carrier itself rather than each individual SNM container.

Content

Describe the types of seals that will be used to secure the material's container during transport.

5.4 Inspection [73.67(g)(1)(iv)]

Intent

The intent of this paragraph is to require the shipper to check the integrity of the material container's seals just prior to shipment so as to be assured that they have not been compromised. Then, upon receipt of the shipment, if the receiver discovers the container's integrity has been compromised and the material is missing, the scope of the recovery operation can focus on the transportation route.

Content

Describe the procedures to be used to ensure that the integrity of the containers or seals is checked just prior to shipment.

5.5 Responsibility for In-Transit Physical Protection [73.67(g)(1)(v)]

Intent

The intent of this paragraph is to make clear that the licensee shipping the material is responsible for arranging for the physical protection of the material in transit if the receiver is not a licensee. If both the shipper and receiver are licensees, the shipper may allow the receiver to accept this responsibility wholly or in part provided there is appropriate documentation specifying their respective responsibilities. Where no such documentation exists, both the shipper and receiver are held jointly responsible. (See Section 6.3 of Part II of this guide.)

Content

In its security plan, the shipper should either acknowledge responsibility for the in-transit physical protection of SNM of low strategic significance or ensure that a written agreement from the receiver has been received in which the receiver accepts either full responsibility or shared responsibility for the in-transit physical protection of this material in accordance with paragraph 73.67(g)(3) of 10 CFR Part 73.

6. RECEIVER REQUIREMENTS--TRANSPORTATION

This chapter provides guidance on meeting the requirements of paragraph 73.67(g)(2), which are as follows:

(g)(2) Each licensee who receives quantities and types of special nuclear material of low strategic significance shall:

(i) Check the integrity of the containers and seals upon receipt of the shipment,

(ii) Notify the shipper of receipt of the material as required in § 70.54 of Part 70 of this chapter, and

(iii) Arrange for the in-transit physical protection of the material in accordance with the requirements of paragraph 73.67(g)(3) of this part, unless the shipper is a licensee and has agreed in writing to arrange for the in-transit physical protection.

6.1 Inspection [73.67(g)(2)(i)]

Intent

This requirement is intended to determine whether the material's container has been compromised en route and whether any material has been removed so that immediate recovery procedures can be initiated if required.

Content

Describe the procedures to be used to ensure that the integrity of the containers and seals will be checked upon receipt of the material shipment.

6.2 Notification [73.67(g)(2)(ii)]

Intent

This requirement is intended to ensure that knowledge of the current location of all SNM is available and to formally inform the shipper that the material has been received.

Content

Ensure that a completed copy of Form NRC-741, "Nuclear Material Transaction Report," will be sent to the shipper within 10 days after a material shipment has been received as required in § 70.54 of 10 CFR Part 70.

6.3 Responsibility for In-Transit Physical Protection [73.67(g)(2)(iii)]

Intent

The intent of this paragraph is to make clear that the licensee receiving the material is responsible for arranging for the physical protection of the

material in transit if the shipper is not a licensee. If both the shipper and receiver are licensees, the receiver may allow the shipper to accept this responsibility either wholly or in part provided there is appropriate documentation specifying their respective responsibilities. Where no such documentation exists, both the shipper and receiver are held jointly responsible. (See Section 5.5 of Part II of this guide.)

Content

In its security plan, the receiver should either acknowledge responsibility for the in-transit physical protection of SNM of low strategic significance or ensure that a written agreement from the shipper has been received in which the shipper accepts either full responsibility or shared responsibility for the in-transit physical protection of this material in accordance with paragraph 73.67(g)(3) of 10 CFR Part 73.

7. IN-TRANSIT PHYSICAL PROTECTION REQUIREMENTS

This chapter provides guidance on meeting the requirements of paragraph 73.67(g)(3), which are as follows:

(g)(3) Each licensee, either shipper or receiver, who arranges for the physical protection of special nuclear material of low strategic significance while in transit or who takes delivery of such material free on board (f.o.b.) the point at which it is delivered to a carrier for transport shall:

(i) Establish and maintain response procedures for dealing with threats of thefts or thefts of such material,

(ii) Make arrangements to be notified immediately of the arrival of the shipment at its destination, or of any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination, and

(iii) Conduct immediately a trace investigation of any shipment that is lost or unaccounted for after the estimated arrival time and report to the Nuclear Regulatory Commission as specified in § 73.71 and to the shipper or receiver as appropriate. The licensee who made the physical protection arrangements shall also immediately notify the Director of the appropriate Nuclear Regulatory Commission Regional Office listed in Appendix A of the action being taken to trace the shipment.

7.1 Response Procedures [73.67(g)(3)(i)]

Intent

The intent of this regulation is to help the licensee identify those transportation incidents that could affect the security of the SNM in transit for which notification might be expected and for which response procedures should be planned.

Content

Identify those events for which response procedures will be developed. Also describe the type of response to be accomplished for each event identified and the duties and responsibilities of the security organization and management involved in the response. Ensure that the NRC will be notified immediately in the event of theft or attempted theft of the material.

7.2 Notification [73.67(g)(3)(ii)]

Intent

The intent of this requirement is to ensure that the licensee responsible for the physical protection of SNM in transit will have a firm basis for deciding whether or not to initiate response procedures in the event a shipment becomes overdue or is lost.

Content

Describe the arrangements and procedures that will be used for notifying the licensee who arranges for the physical protection of material in transit (1) of the arrival of the shipment at its destination or (2) of any such shipment that is lost or unaccounted for after the estimated time of arrival at its destination.

7.3 Lost Material Notification [73.67(g)(3)(iii)]

The intent of this requirement is to ensure that, in the event a shipment becomes overdue and no reasonable explanation has been received from the carrier regarding its status, a trace investigation will be conducted to locate the missing SNM. At this time, the NRC should be notified that the material is missing and informed as to what steps are being taken to recover it. Although the licensee is responsible for notifying the NRC of any missing material and for initiating and assisting in the subsequent investigation, the law enforcement agencies bear the responsibility for physically recovering the material.

Content

Describe what procedures will be used to trace any shipment that is lost or has not arrived by the estimated arrival time. Ensure that all lost or missing material will be immediately reported to the appropriate NRC Regional Office along with what actions are being taken to trace the shipment, that the NRC will be notified as specified in § 73.71, and that the shipper or receiver, as appropriate, will also be notified.

Appendix 12.4

Regulatory Guide 2.5
Quality Assurance Program Requirements for Research Reactors

U.S. NUCLEAR REGULATORY COMMISSION

Revision O-R
May 1977

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

(This guide reissued October 1977)

REGULATORY GUIDE 2.5

QUALITY ASSURANCE PROGRAM REQUIREMENTS FOR RESEARCH REACTORS

A. INTRODUCTION

Paragraph (a)(7) of §50 34, "Contents of Applications, Technical Information" of 10 CFR Part 50, "Licensing of Production and Utilization Facilities" requires that each applicant for a construction permit to build a production or utilization facility include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design and construction of the structures, systems and components of the facility Paragraph (b)(6)(ii) of §50 34 requires that each applicant for a license to operate a facility include in the final safety analysis report a description of the managerial and administrative controls to be used to assure safe operation This guide describes a method acceptable to the NRC staff of complying with the Commission's regulations with regard to overall quality assurance program requirements for research reactors

B. DISCUSSION

Work Group ANS-15 8 of Subcommittee ANS-15, Research Reactors, of the American Nuclear Society Standards Committee has developed a standard that describes a quality assurance program for use in research reactors The standard was approved by the American National Standards Committee N17, Research Reactors It was subsequently approved and designated ANSI N402-1976 by the American National Standards Institute (ANSI) on August 19, 1976

C. REGULATORY POSITION

The general requirements for establishing and executing a quality assurance program for the design, construction testing modification, and maintenance of research reactors that are included in ANSI N402-1976 "Quality Assurance Program Requirements for Research Reactors" provide an acceptable method for complying with the program requirements of 10 CFR §50 34

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals in connection with applications for operating licenses, construction permits, or proposed amendments thereto docketed after January 16, 1978, unless this guide is revised as a result of suggestions from the public or additional staff review

If an applicant wishes to use this regulatory guide in developing submittals for applications docketed on or before January 16, 1978, the pertinent portions of the application will be evaluated on the basis of this guide

[1] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue LaGrange Park, Illinois 60525

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents or to provide guidance to applicants Regulatory Guides are not substitutes for regulations and compliance with them is not required Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission

Comments and suggestions for improvements in these guides are encouraged at all times and guides will be revised as appropriate to accommodate comments and to reflect new information or experience However the staff's consideration of comments received during the initial public comment period for this guide has resulted in the determination that there is no need for a revision at this time

Comments should be sent to the Secretary of the Commission U S Nuclear Regulator, Commission Washington D C 20555 Attention Docketing and Service Branch

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13 ACCIDENT ANALYSES

In the other chapters of the SAR, the applicant should discuss and analyze the safety considerations and functional requirements at a non-power reactor facility for the design bases that ensure safe reactor operation and shutdown and acceptable protection for the public, the operations and user staff, and the environment. In those chapters, the applicant should not only discuss potential equipment malfunctions, deviations of process variables from normal values, and potential effects of external phenomena on the facility, but should also describe how equipment will work when needed in accident situations. In Chapter 13 of the SAR, the applicant should submit information and analyses that show that the health and safety of the public and workers are protected and that the applicant has considered potential radiological consequences in the event of malfunctions and the capability of the facility to accommodate such disturbances. The major purpose of this chapter is for the applicant to demonstrate that the facility design features, safety limits, limiting safety system settings, and limiting conditions for operation have been selected to ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

The issue of what standards to use in evaluating accidents at a research reactor was discussed in an Atomic Safety and Licensing Appeal Board (ASLAB) decision issued May 18, 1972, for the research reactor at Columbia University in New York City. The ASLAB stated that "as a general proposition, the Appeal Board does not consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The staff has not found it necessary to follow the board's recommendation to develop separate criteria for evaluating research reactor accidents because most research reactors to date have been able to conform to the conservative criteria of 10 CFR Part 20.

The principal safety issues that differentiate test reactors from research reactors are the reactor site requirements and the doses to the public that could result from a serious accident. For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 (10 CFR 20.1 through 20.602 and appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results are less than 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301.

In several instances, the staff has accepted very conservative accident analyses with results greater than the 10 CFR Part 20 dose limits discussed above.

If the facility conforms to the definition of a test reactor, the results should be compared with 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses given in 10 CFR Part 100 are reference values and are not intended to imply that the dose numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, they are values that can be used for evaluating reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of exposure of the public to radiation.

The accidents analyzed should range from such anticipated events as a loss of normal electrical power to a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible. This limiting accident is named the maximum hypothetical accident (MHA) for non-power reactors; the details are reactor specific. Because the MHA is not expected to occur, the scenario need not be entirely credible. The initiating event and the scenario details need not be analyzed, but the potential consequences should be analyzed and evaluated.

The information on credible postulated accidents should achieve the following objectives:

- Ensure that enough events have been considered to include any accident with significant radiological consequences. Rejection of a potential event should be justified in the discussions.

- Categorize the initiating events and scenarios by type and likelihood of occurrence so that only the limiting cases in each group must be quantitatively analyzed.

- Develop and apply consistent, specific acceptance criteria for the consequences of each postulated event.

Each postulated event should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor:

- MHA
- insertion of excess reactivity (ramp, step, startup, etc.)
- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel

- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

The accident events in each group should be evaluated systematically to identify the limiting event selected for detailed quantitative analysis. Limiting events in each category should have potential consequences that exceed all others in that group. As noted above, the MHA selected should bound all credible potential accidents at that facility, yet should be an event that is not likely to occur during the life of the facility.

13.1 Accident-Initiating Events and Scenarios

In this section of the SAR, the applicant should describe potential accident-initiating events and scenarios for non-power reactors. For documents on general accident scenarios and analysis, radiological consequences, and fuel types, see Section 13.4. The following sections contain suggestions for selecting and categorizing postulated accidents:

13.1.1 Maximum Hypothetical Accident

In general, the escape of fission products from fuel or fueled experiments and their release to the unrestricted environment would be the most hazardous radiological accident conceivable at a non-power reactor. However, non-power reactors are designed and operated so that a fission product release is not credible for most. Therefore, this release under accident conditions can reasonably be selected as the MHA, which bounds all credible accidents and can be used to illustrate the analysis of events and consequences during the accidental release of radioactive material. The applicant may choose to perform sensitivity analysis of the assumptions of the MHA. For example, reactor operating time before accident initiation may be examined to determine the change in MHA outcome if a more realistic assumption is made. However, these assumptions may form the basis for technical specification limits on the operation of the facility. The MHA could be any of the following:

- A specified fraction of fuel in the core melts. (How this occurs may or may not be specified.)

- Cladding is stripped from a specified fraction of the core fuel plates or elements.

- The fuel encapsulation bursts, releasing gaseous fission products to the pool or the air. (The failure of one fuel element in air is the MHA for a TRIGA reactor.)

- A fueled experiment melts or fails catastrophically in the pool or in the air.

13.1.2 Insertion of Excess Reactivity

In some cases, the insertion of excess reactivity can be an initiating event that leads to fuel or fueled experiment melting, which is the MHA. Insertion-of-excess-reactivity accidents can also be used to show that limiting conditions for operation on reactivity are justified. Some insertion-of-excess-reactivity events are the following:

- Rapid inadvertent insertion of a portion of all excess reactivity loaded into the reactor.

- Rapid removal of the most reactive control rod or shim rod.

- Rapid insertion of a fuel element into a vacancy in the core at the most reactive position.

- Ramp insertion of reactivity by drive motion of the most reactive control rod or shim rod, or ganged rods, if possible. (This event could occur during reactor startup procedures or when the reactor is at power.)

- Failure or other malfunction of an experiment that inserts excess reactivity. (This can be used to justify movable experiment reactivity limits.)

- Rapid increase in reactivity as a result of a change in operating parameters, such as a surge of cold coolant.

13.1.3 Loss of Coolant

In many non-power reactor designs, the loss-of-coolant accident (LOCA) is of no consequence because decay heat in the fuel is so small as to be incapable of causing fuel failure. In some higher power reactors (normally greater than 2 MW), an engineered safety feature, such as an emergency core cooling system, may need to be operable for some time after reactor shutdown to remove decay heat in the event of a LOCA. Some initiators of LOCAs are the following:

- failure or malfunction of some component in the primary coolant loop
- failure or malfunction of an experimental facility, such as a beam tube
- failure or leak of the reactor coolant boundary

13.1.4 Loss of Coolant Flow

This accident is usually most limiting for forced convection-cooled non-power reactors, where the forced flow is downward through the reactor core. The effects of loss of coolant flow should be considered for all non-power reactors. Upon loss of forced downward coolant flow through the core, coolant flow in the core must reverse to upward natural-convection cooling. During the flow reversal, heat transfer may be inadequate in the core. Loss of coolant flow may also occur if a foreign object obstructs a coolant flow path. Some initiators of loss of coolant flow are the following:

- loss of electrical power
- failure of a pump or other component in the primary coolant system
- blocking or significant decrease in flow in one or more fuel coolant channels

13.1.5 Mishandling or Malfunction of Fuel

This class of accidents represents fuel damage less severe than the MHA. Operation with water-logged fuel is an important consideration for pulsing reactors where the sudden addition of energy to the fuel due to a pulse may cause the water to turn quickly to steam and damage the fuel cladding. Initiating events in this class are the following:

- overheating of fuel during steady-power or pulsed operation
- dropping or otherwise damaging fuel in any location
- dropping, impact, or other malfunction of a non-fueled component
- operation (including pulsing) with damaged fuel, such as water-logged pin- or rod-type fuel

13.1.6 Experiment Malfunction

The conduct of experiments is one of the important functions of a non-power reactor. Experiments may contain fuel, explosives, and highly reactive materials. Failure or malfunction of experiments may initiate accidents. In some cases, particularly for lower power non-power reactors, failure or malfunction of an experiment may be the MHA, especially if fueled experiments are allowed by the facility license. Initiating events for this class of accidents include the following:

- loss of cooling capability or other malfunction in a fueled experiment resulting in liquefaction or volatilization of the fissile component

- loss of cooling capability in a strongly absorbing non-fueled experiment resulting in absorber failure and rapid increase in reactivity

- placement of an experiment component in an unplanned location, causing effects that were not evaluated

- failure of an experiment containing highly reactive contents

- failure of an experiment and release of corrosive materials into the reactor coolant

- detonation of an explosive experiment

13.1.7 Loss of Normal Electrical Power

This accident initiator could result from onsite or offsite power interruptions. Emergency power supplies, if provided, are assumed to operate. However, the applicant may want to analyze the effects of failure of emergency power.

13.1.8 External Events

This class of accident initiators represents some outside effect on the facility, be it natural or caused by humans. Some initiating events in this category are the following:

- meteorological disturbance, such as hurricane, tornado, or flood
- seismic event
- mechanical impact or collision with building
- event caused by humans, such as explosion or toxic release near the reactor building

13.1.9 Mishandling or Malfunction of Equipment

This class of accident initiators represents failures or errors that do not fall into one of the other categories. Some initiators in this category are the following:

- operator error at the controls

- other operator errors

- malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies
- electrical fault in control/safety rod systems
- malfunction of confinement or containment system
- rapid leak of contaminated liquid, such as waste or primary coolant

13.2 Accident Analysis and Determination of Consequences

In this section of the SAR, the applicant should discuss each event giving information consistently and systematically for gaining a clear understanding of the specific reactor and making comparisons with similar reactors. Many of the steps used to select the limiting event in each category may be semiquantitative. However, the analyses and determination of consequences of the limiting events should be as quantitative as possible. The following steps are suggested:

(1) State the initial conditions of the reactor and equipment. Discuss relevant conditions depending on fuel burnup, experiments installed, core configurations, or other variables. Use the most limiting conditions in the analyses.

(2) Identify the causes that initiate the event; the causes may include equipment malfunction, operator error, or a natural phenomenon or one caused by humans. Base the scenario on a single initiating malfunction, rather than on multiple causes.

(3) List the sequence of events, assumed equipment operation and malfunction, and operator actions until a final stabilized condition is reached. Discuss functions and actions assumed to occur that change the course of the accident or mitigate the consequences, such as reactor scrams or initiation of such engineered safety features as emergency core cooling. If credit is taken for mitigation of the accident consequences, discuss the bases used to determine that the systems are operable and discuss the system functions.

(4) Classify damage that might occur to components during the accident until the situation is stabilized. Discuss all components and barriers that could affect the transfer of radiation and radioactivity from the reactor to the public and that ensure continued stability of conditions after the accident.

(5) Prepare realistic analyses to demonstrate a detailed, quantitative evaluation of the accident evolution, including the performance of all barriers and the

transport of radioactive materials to the unrestricted area. Include the assumptions, approximations, methodology, uncertainties, degree of conservatism, margins of safety, and both intermediate transient and ultimate radiological conditions. Justify the methods used. Further, make sure the information is complete enough to allow the results to be independently reproduced or confirmed. Demonstrate the validation of the computational models, codes, assumptions, and approximations by comparison with measurements and experiments when possible. Describe in detail computer codes that are used as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operating non-power reactors. Include estimates of the accuracy of the analytical methods. In Chapter 11, "Radiation Protection Program and Waste Management," of the SAR, discuss the methods and assumptions used to analyze the release and dispersion of radioactive materials from normal operations. Adapt those methods as appropriate for accident analyses.

(6) Define and derive the radiation source terms, if any are involved. Include in the source terms the quantity and type of radionuclides that could be released, their physical and chemical forms, and the duration of potential releases. Describe potential radiation sources that could cause direct or scattered radiation exposure to the facility staff and the public.

(7) Evaluate the potential radiological consequences using realistic methods. Discuss the degree of conservatism in the evaluation. For example, include a discussion of the degree of conservatism introduced by the use of postulated release fractions or assumption of an infinite hemispherical cloud.

Include environmental and meteorological conditions specific for the facility site to illustrate consequences. Exposure conditions should account for the facility staff until the situation is stabilized (including staff evacuation and reentry), the most exposed member of the public in the unrestricted environment until the accident conditions are terminated or the person is moved, and the integrated exposure at the facility boundary and the nearest permanent residence. The radiological consequences should include external and internal exposures. Address contamination of land and water where applicable; include exposure control measures to be initiated.

13.3 Summary and Conclusions

In this section of the SAR, the applicant should summarize the important conclusions about the postulated accidents and the potential consequences. The applicant should compare the projected radiological consequences with the acceptance criteria discussed previously in this chapter The information should demonstrate that all reasonable measures have been incorporated into the facility

design bases to prevent undue radiation exposures and contamination of the unrestricted environment. The discussions should show that engineered safety features have been incorporated where necessary to limit consequences to acceptable levels.

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14 TECHNICAL SPECIFICATIONS

In this chapter of the SAR, the applicant should discuss the development of the facility technical specifications. This chapter of the format and content guide discusses the contents of Chapter 14 of the SAR and presents guidance in Appendix 14.1 on the format and content of technical specifications for non-power reactors.

NRC requires each applicant for a license to operate a non-power reactor to develop technical specifications that state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36. The technical specifications are typically derived from the facility descriptions and safety considerations contained in the SAR and represent a comprehensive envelope of safe operation.

Applications for construction permits or operating licenses and renewals of operating licenses must contain proposed technical specifications that will be incorporated in the operating license. During its review of the application, the NRC staff will review the SAR and proposed technical specifications to ensure they are complete and comprehensive and that the environment and public health and safety will be protected. After final acceptance by the NRC staff, the technical specifications will be included as Appendix A to the operating license.

The format and content of the technical specifications discussed in Appendix 14.1 follow the format of the 1990 revision to American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1. Examples of the format and content of technical specifications can be found in previously accepted and approved technical specifications for non-power reactors of similar design, operating characteristics, site and environmental conditions, and use.

This chapter of the SAR normally is very short. The applicant should be able to state conclusively that the technical specifications were prepared following an accepted format, that normal operation of the reactor within the limits of the technical specifications will not result in offsite radiation exposure in excess of 10 CFR Part 20 guidelines, and that the technical specifications limit the likelihood and consequences of malfunctions. The reader is referred to the technical specifications, which are in a document separate from the SAR. The technical specifications are neither derived nor justified in this chapter of the SAR. They are determined by the analyses that appear in the other chapters of the SAR. Each of the technical specifications should be supported by the SAR, and it is useful to refer to the supporting SAR analysis in the basis of each technical specification.

In Appendix 14.1, every section of ANSI/ANS 15.1 is addressed. If ANSI/ANS 15.1 should be modified or clarified to provide acceptable technical specifications,

additional guidance is given. Sections that provide acceptable guidance as written are noted.

The guidance in this chapter and in Appendix 14.1 pertains to technical specifications for all types of non-power reactors. Not all of the guidance given is applicable to a particular reactor type. The applicant should propose and justify those technical specifications that are applicable to the reactor design and utilization under consideration.

The standard format and content of technical specifications for non-power reactors are presented in Appendix 14.1. The numbering system (Sections 1 through 6.8) corresponds to the numbering system in ANSI/ANS 15.1–1990.

Reference

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

Appendix 14.1

Format and Content of Technical Specifications for Non-Power Reactors

Format and Content of Technical Specifications for Non-Power Reactors

The numbering system in this appendix (Sections 1 through 6.8) corresponds to the numbering system in ANSI/ANS 15.1–1990.

1 INTRODUCTION

1.1 Scope

NRC accepts the guidance provided in this section of ANSI/ANS 15.1. This section confirms that the technical specifications for non-power reactors should include all the categories in 10 CFR 50.36 for production and utilization facilities.

1.2 Application

1.2.1 Purpose

NRC accepts the guidance provided in this section of ANSI/ANS 15.1. Technical specifications represent a set of operating requirements for a reactor that the applicant and NRC have agreed on. The specifications become part of the operating license.

1.2.2 Format

Sections of the technical specifications should be numbered as indicated in Section 1.2.2 of ANSI/ANS 15.1. Subsections may be left out if not applicable for a particular reactor or may be altered if necessary, but the subsections included should be arranged in consecutive numerical order.

For individual specifications in Sections 2, 3, and 4, applicability, objective, specification, and basis information should be included in the specified format. For Sections 5 and 6 of the technical specifications, ANSI/ANS 15.1 suggests that the specifications be stated without providing applicability, objective, or basis. Although the ANSI format is preferred, NRC will accept these sections even if they include applicability, objective, or basis statements.

Technical specifications that use the SAR as a basis should explicitly reference the SAR section number. In addition, any other sources used to support the technical specification should be explicitly referenced.

1.3 Definitions

NRC and the non-power reactor community have agreed on most of the definitions given in this section of ANSI/ANS 15.1. Those applicable to a particular facility should be included verbatim. Facility-specific definitions may be added to clarify terms referred to in the technical specifications. Modifications and additional definitions presented below help clarify the meaning of terms used in ANSI/ANS 15.1.

The following definitions should be modified as indicated:

- **Class A reactor operator.** The term acceptable to NRC is senior reactor operator.

- **Class B reactor operator.** The term acceptable to NRC is reactor operator.

- **Reactor shutdown.** The reactor is shut down if it is subcritical by at least 1 dollar both in the reference core condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

- **Reference core condition.** The reference core condition is the reactivity condition of the core when it is at 20 °C and the reactivity worth of xenon is zero (i.e., cold, clean, and critical).

- **Shutdown margin.** Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition. It should be assumed that the most reactive scrammable rods and all non-scrammable rods are in their most reactive position and that the reactor will remain subcritical without further operator action.

  Note: Shutdown margin has a single value mutually acceptable to NRC and the applicant, to be determined on a case-by-case basis.

The following definitions should be added:

- **Secured shutdown.** Secured shutdown is achieved when the reactor meets the requirements of the definition of "reactor secured" and the facility administrative requirements for leaving the facility with no licensed reactor operators present.

- **Shutdown reactivity.** Shutdown reactivity is the value of the reactivity of the reactor with all control rods in their least reactive positions (e.g., inserted). The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

All reactor licensees are required by 10 CFR 50.36(c) to specify safety limits in the technical specifications. These safety limits should be placed on important process variables identified in the SAR as necessary to reasonably protect the integrity of the primary barrier against the uncontrolled release of radioactivity. For non-power reactors, the radioactivity of concern is generally the fission products in the fuel. For heterogeneous-core non-power reactors, the primary barrier is the cladding of fuel plates, rods, or pins. Cladding integrity could be lost by softening, melting, blistering, or yielding to excessive internal pressure, all of which are dependent on temperature and operating history. For homogeneous-core reactors, this primary barrier may be the fuel matrix, the primary vessel, or some other component that contains the fuel and the fission products.

Reactor conditions and safety limits should be developed to avoid failure of the fuel and should be supported by SAR analyses. Manufacturers have studied failure modes and failure parameters during fuel development programs. NRC has issued staff reports in the NUREG series approving the use of some types of low-enriched uranium fuel in non-power reactors. The applicant should make maximum use of the appropriate references, some of which are listed at the end of this appendix.

The applicant should consult NUREG-1313 for evaluating aluminum-clad, aluminum matrix plate-type fuels, using both highly enriched uranium (HEU) and new uranium-silicide low-enriched uranium (LEU). The report discusses temperatures from experimental irradiation tests at which plate blistering has been observed, a possible forerunner of failure. NRC finds 530 °C an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation. NUREG-1313 also references tests with HEU plate fuel that have led to similar conclusions (Beeston et al., 1980; Gibson, 1967; Nazare et al., 1975; Stahl, 1982).

There are several reports on training reactor and isotope production, General Atomics (TRIGA)-type fuels (NUREG-1282; Simnad et al., 1976 and 1981;

Simnad and West, 1986; West et al., 1986). For stainless steel–clad $UZrH_{1.65}$ LEU 8.5 uranium weight percent (w/o) TRIGA fuel, stainless steel–clad $UZrH_{1.65}$ HEU (70% U-235 enriched) 8.5 w/o fuel lifetime improvement program (FLIP) TRIGA fuel, and stainless-steel-clad $UZrH_{1.65}$ LEU 20 w/o and 30 w/o TRIGA fuel, General Atomics has shown and NRC has accepted that integrity is not compromised under the following cases and conditions:

- for cladding temperature at or less than 500 °C, peak fuel temperature at or less than 1150 °C
- for cladding temperature greater than 500 °C, peak fuel temperature at or less than 950 °C

For aluminum-clad $UZrH_{1.0}$ LEU 8 w/o TRIGA fuel, NRC has accepted that the peak fuel temperature should not exceed 500 °C.

For pulsed training assembled reactor (PULSTAR) types, NRC has accepted that the UO_2 fuel temperature should not exceed 2400 °C and the Zircaloy-2 cladding temperature should not exceed 1500 °C.

For Aerojet-General Nucleonics (AGN)-201 reactor types, NRC has accepted that the fuel temperature should not exceed 200 °C.

The applicant should base SAR analyses on the applicable fuel developer's reported test results to ensure fuel integrity under all operating conditions.

2.1.1 Important Process Variables

ANSI/ANS 15 1 proposes a list of parameters that may be acceptable as process variables for non-power reactors and states that safety limits will be measurable parameters. However, as discussed in Sections 2.1.2 and 2.1.3 (below), not all safety limits for non-power reactors must be monitored and actually measurable. Safety limits could be inferred from limitations on other process variables.

It is convenient in discussing fuel integrity to divide the non-power reactors into two groups: those with engineered cooling systems (forced-convection cooling) and those without engineered cooling systems (natural-convection cooling or no active cooling system). Safety limits for these reactors are discussed in the next two sections

2.1.2 Criteria—Reactors With Engineered Cooling Systems

NRC modifies this section of ANSI/ANS 15.1 as follows. Operation of the cooling system for reactors with forced-convection cooling maintains fuel

temperature within acceptable limits to ensure cladding integrity. Important parameters include fuel temperature, coolant flow rate, coolant inlet temperature, height of coolant above core, and reactor power level. These parameters should be controlled and measured. When all values are jointly maintained within the limits determined by the safety analyses, fuel cladding integrity will not be lost. These parameters are important process variables on which safety limits should be established and specified in the technical specifications. Safety limits should preclude flow instabilities in the hottest channel and ensure that the minimum departure from nucleate boiling ratio (DNBR) is at least 2.0 (which has been an acceptable margin to the onset of nucleate boiling). The analyses should range over all physical and engineering parameters of the fuel components, the core configurations, and the coolant systems, and should also include consideration for uncertainties.

For reactors that will operate with both natural-convection cooling and forced-convection cooling, safety limits should be specified for appropriate process variables in both modes of operation (see Section 2.1.3 below). All non-power reactors should be designed so that both fission heat and decay heat can be dissipated without fuel damage. The analyses also should show that the safety limits are not exceeded during all anticipated modes of operation.

2.1.3 Criteria—Reactors Without Engineered Cooling Systems

For reactors that will be licensed to operate without forced-convection cooling, only Section 2.1.3(2) of ANSI/ANS 15.1 is acceptable according to 10 CFR 50.36(c). NRC modifies Section 2.1.3(2) of ANSI/ANS 15.1 as follows. For reactors that either circulate coolant by natural thermal convection or have no specific coolant or cooling systems, thermal-hydraulic coolant parameters are not separately controllable. The applicant should ensure fuel cladding integrity and should note appropriate parameters chosen for safety limits in the technical specifications.

High fuel temperature is the likely precursor of fuel failure. Therefore, a maximum allowable fuel temperature safety limit should be established below which fuel integrity is ensured. On the basis of this fuel temperature, a power level should be calculated using an appropriate margin which ensures that the fuel remains below the fuel temperature safety limit.

If the license will contain a provision to measure fuel temperature, the maximum fuel temperature in the core would be the parameter on which a safety limit is established. The SAR should show the relationship between the measured fuel temperature and the maximum fuel temperature for the proposed reactor conditions. If there is no provision for measuring fuel temperature directly, the calculated power level should be selected as the safety limit, on the basis of the

maximum allowable fuel temperature and appropriate margin. However, the basis for the safety limit still should be the maximum allowable fuel temperature.

Because most TRIGA-fueled cores have at least one instrumented fuel rod, NRC has accepted fuel temperature alone as the safety limit for these reactors. Because the point of measured fuel temperature is normally not the point of maximum fuel temperature, the SAR should show the relationship between the measured fuel temperature and the maximum fuel temperature.

For plate-type fuel, the capability to measure fuel temperature is generally not available. Therefore, the applicant should determine a fuel cladding temperature below which cladding damage (softening or blistering) can be precluded. The applicant should then establish a corresponding power level, reactor conditions, and uncertainties that limit cladding temperature below the damage limit.

For reactors without fuel elements, such as the homogeneous AGN-201 reactors, safety limits should be based on considerations similar to those for plate-type fuel. The power level established in the SAR as a safety limit must provide reasonable assurance that fission products will not be released from their confining barrier, which could be either the fuel matrix or the fuel canister.

Safety limits should be based on the SAR. The technical specifications should discuss the mechanism and magnitude of the fuel limitation, including a primary reference for the fuel development studies that support the safety limit presented (e.g., NUREG-1282 and -1313; Simnad et al., 1976). The analysis should address authorized core configurations and limiting thermal power levels and conditions for the reactor.

Safety limits acceptable to NRC for various reactor fuels are discussed in Section 2.1 (above).

2.2 Limiting Safety System Settings

NRC accepts the guidance of this section of ANSI/ANS 15.1. The SAR should address normal operating conditions, off-normal operations, and all pertinent postulated accident scenarios. For each parameter on which a safety limit is established by the SAR, a protective channel should be identified that prevents the value of that parameter from exceeding the safety limit. The calculated setpoint for this protective action, providing the minimum acceptable safety margin considering process uncertainty, overall measurement uncertainty, and the transient phenomena of the process instrumentation, is defined as the "limiting safety system setting (LSSS)." Because the LSSSs are analytical limits, the protective channels may be set to actuate at more conservative values. The more conservative values may be established as limiting conditions for operation (LCOs).

Such LCOs may be determined on the basis of experience, which has shown that safety system channels can be set readily within 20 percent of the normal operating value for a measured parameter, if the LSSS is not exceeded, without undue interference to operations. In many cases, the LCO can even be within 10 percent of the operating value. The SAR justification for LSSSs and LCOs should be referenced.

2.2.1 Criteria—Reactors With Engineered Cooling Systems

NRC accepts the guidance of this section of ANSI/ANS 15.1 for the forced-convection cooling mode of operation. For reactors licensed to operate with forced-convection cooling, this specification should list the LSSS derived in the SAR for each reactor parameter for which a safety limit was established. The bases part of this specification should indicate the SAR assumptions and limits of uncertainty for each analyzed LSSS. For reactors licensed to operate in forced- and natural-convection cooling modes, appropriate LSSSs should be listed for both modes.

2.2.2 Criteria—Reactors Without Engineered Cooling Systems

NRC substitutes the following guidance for Section 2.2.2 of ANSI/ANS 15.1. Section 2.1.3 (above) requires that safety limits be established by SAR analysis for all licensed reactors; therefore, channels should be established on the basis of SAR analysis to not violate each of these safety limits. Calculated LSSSs defined in Section 2.2 of ANSI/ANS 15.1 and in this appendix should be provided as technical specifications.

3 LIMITING CONDITIONS FOR OPERATIONS

LCOs are derived from the safety analyses in the SAR, which provide the bases for the LCOs. LCOs are implemented administratively or by control and monitoring circuitry to ensure that the reactor is not damaged, that the reactor is capable of performing its intended function, and that no one suffers undue radiological exposures because of reactor operations.

NRC accepts the guidance of this section of ANSI/ANS 15.1 as amplified in the sections that follow. Many of the LCOs have evolved from experience. Many are facility specific, depending on reactor type, operating characteristics, and site location. NRC accepts the LCOs discussed in this section provided that the applicant justifies them and shows the applicability to the specific facility. Additional specifications may be appropriate for unique facility designs or experimental features or for additional conservatism in operations required by the applicant or NRC. As noted above, LCOs can in many cases be set within 10 percent of the normal operating level of a parameter. Specifications on surveillance intervals for

LCOs and other parameters and facility design features are given in Sections 4 and 5, respectively, of ANSI/ANS 15.1 and this appendix. LCOs should be provided as outlined in the remainder of this section.

3.1 Reactor Core Parameters

(1) Excess Reactivity

The upper limit for allowed excess reactivity should be specified. The referenced SAR analyses should discuss all operations that require excess reactivity and the safety implications for the excess reactivity proposed. The discussions should include operational flexibility, potential accidents, and relationship to shutdown margin. The SAR (Chapter 4, "Reactor Description," and Chapter 13, "Accident Analyses") should contain a discussion of the safety implications of the excess reactivity, including the following:

- resultant shutdown reactivity with all control rods inserted
- effects on the reactor of any credible rapid removal of a control or safety rod
- potential effects of other maximum credible rapid additions of excess reactivity
- possible reactivity changes caused by experiment failure or displacement
- interrelationship between shutdown margin and excess reactivity

If none of the postulated events would lead to loss of fuel integrity or to uncontrolled release of radioactivity, the proposed excess reactivity would be acceptable.

(2) Shutdown Margin

A single value for the shutdown margin, as defined in Section 1.3 (above), should be specified. The specification should state that compliance with the shutdown margin takes precedence over the excess reactivity specification. In addition, other reactor parameters that apply to the shutdown margin should be stated. These should include cold, clean core reactivity conditions (e.g., temperature and poisons), core configuration (e.g., fuel and control rods), and the status of experiments (e.g., movable experiments in their most reactive state). The value of the shutdown margin should be large enough to be readily determined experimentally, for example, $\geq 0.5\%\ \Delta k/k$ or ≥ 0.50 dollar.

(3) Pulse Limits

Because the TRIGA design is the only pulsing reactor design NRC licenses at this time, the specific values given in the discussion below apply only to TRIGA design pulsing reactors. However, the general design criteria discussed below may be applicable to other potential pulsing non-power reactor designs.

The maximum reactivity addition for a pulse is a license condition similar to maximum thermal power and is determined case by case. The value should be based on the SAR analysis for maintaining fuel integrity, which considers fuel type, limiting core configurations, reactivity feedback coefficients, operating history, heat capacity, and peak fuel temperature limitations. This LCO on the maximum reactivity addition administratively gives assurance that the maximum pulse reactivity addition license condition and the safety limit on maximum fuel temperature will not be exceeded.

The SAR should show that the maximum reactor pulse for a TRIGA reactor with stainless steel-clad $UZrH_{1.65}$ fuel would not raise the peak fuel temperature of any element above 1000 °C (Simnad et al., 1976). (This is a conservative limit, proposed by General Atomics and accepted by NRC, that is not to be confused with the safety limit temperature value.) For a TRIGA reactor with aluminum-clad $UZrH_{1.0}$ fuel, the analysis should show that the peak fuel temperature will not exceed 500 °C for the maximum reactor pulse. The analysis should be applicable to the specific reactor considering its core size, operating history, fuel types, feedback coefficients, temperature gradients and the power peaking of all authorized core configurations. The potential effects of pulsing on in-core experiments or detectors should be included in the analysis. Any required limitations on experiments should be noted in Section 3.8 of the technical specifications.

The report by Simnad et al. (1981) discussing fuel damage at the Texas A&M University TRIGA reactor should be reviewed to determine if reactor operating history and power level would require a lower peak pulse fuel temperature because of damage to the fuel during pulsing operation.

There should be a limit on the total worth of the pulse rod to ensure that the worth of the pulse rod could not exceed the maximum reactivity insertion limit and allow an amount of reactivity to be inserted into the core that could damage the fuel. There should be a steady-state power level above which pulses should not be initiated. For TRIGA reactors, NRC has accepted a power level of 1 kW in conjunction with an interlock that prevents movement of the steady-state control rods when the reactor is in the pulse mode.

For other pulsing reactors, the proposed limiting fuel temperature and reactivity insertion should be justified by reference to appropriate tests and analyses.

(4) Core Configurations

The applicant should specify special core configurations, experimental facilities internal to the core, special neutron reflectors, burnable poisons, or mixed fuel types assumed in the SAR. The following specifications should be included in the LCO for core configurations:

- If analysis shows the reactivity effects of waterholes in the core need to be limited due to reactivity insertion accidents, the reactor should have a closely packed core and acceptable vacancies in the core center and the periphery should be described. This does not prevent the use of in-core experimental facilities. Reactors with thermal power levels in excess of 1 MW and a cross-sectional area of a core experimental facility greater than 16 square inches will be licensed as testing facilities.

- No fuel should be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive fuel element.

- If control rods need to be removed from the reactor core for inspection, an LCO should state the negative reactivity necessary in the core before a control rod can be removed.

- If analysis shows that power peaking or power density is a concern in mixed cores, then core geometry may need to be restricted to allow certain types of fuel assemblies or elements only in certain core positions.

Non-power reactors should be designed with reactivity and void coefficients and a power defect sufficiently negative that many reactor transients are inherently counteracted to avoid loss of fuel integrity. Although the individual reactivity coefficients and power defect are addressed in the specification below, this LCO should be used to develop specifications on allowed core configurations to ensure the assumptions used in the development of limits on those parameters are met.

The specified conditions of core configuration are acceptable to NRC if the SAR shows that none of the conditions analyzed could lead to loss of fuel integrity, uncontrolled release of radioactivity, or potential exposures exceeding 10 CFR Part 20.

(5) Reactivity Coefficients (Added by NRC)

Non-power reactors should specify reactivity coefficients for fuel temperature, moderator temperature, and void volume and a power defect if these parameters could vary unacceptably with reactor operation. In many cases, these measurements need only be made during the startup testing program. The net effect of the coefficients and the power defect should be negative over most of the range of reactor operations. The SAR analyses of both routine operation and potential accident scenarios should show that the net negative effect of these core characteristics is sufficient to mitigate any anticipated event or postulated accident scenario. Reasonable values should be designed into the reactor (e.g., by under-moderation of the neutron spectrum). Values for surveillance should be specified for those negative reactivity coefficients and the negative power defect that can be measured. The values of the coefficients and the power defect are acceptable if they ensure that the assumptions and initial conditions of the analyses are enveloped to prevent compromise of the fuel integrity during reactor transients and other applicable accident scenarios.

(6) Fuel Parameters (Added by NRC)

An LCO should be specified for certain fuel parameters or characteristics.

Design features of the approved fuel should be included in Section 5 of the technical specifications. Fuel-related LCOs include the following:

(a) All Fuel Types

- No operation with damaged fuel except to locate such fuel. The definition of damaged fuel should specify limits on longitudinal growth, bowing, or bending, and limits on detectable amounts of fission products that could escape through the primary barrier.

- Periodic visual inspection of fuel. This specification should be clear and explicit and should reference fuel manufacturers' guidance or recommendations for detecting deterioration. The intervals and methods of fuel inspection should be specified in Section 4 of the technical specifications The purpose of inspection is to detect cladding deterioration that results from erosion, corrosion, or other damage.

(b) TRIGA Fuel

Additional technical specifications limit fuel rod elongation, bowing, and uranium burnup. Limits listed below were proposed by General Atomics

and accepted by the Atomic Energy Commission (AEC) during initial licensing of pulsing TRIGA reactors. If these limits are reached, the fuel element is defined as "damaged fuel." Acceptable specifications for TRIGA fuel for both steady-state and pulsed operation include the following:

- Bowing—For stainless steel-clad $UZrH_{1.65}$ TRIGA fuel, the sagitta shall not exceed 0.125 in. (0.318 cm) over the length of the cladding in a hexagonal-grid core arrangement or 0.0625-in. (0.159-cm) elongation over the original length of the cladding in a circular-grid core arrangement. For aluminum-clad $UZrH_{1.0}$ fuel, the limit on the sagitta should be 0.125 in. (0.318 cm).

- Elongation—For stainless steel-clad $UZrH_{1.65}$ TRIGA fuel, the total length of the fuel element shall not exceed its original length by more than 0.125 in. (0.318 cm). For aluminum-clad $UZrH_{1.0}$ fuel, the limit on elongation should not exceed 0.5 in. (1.27 cm).

- Burnup—The burnup of uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration (NUREG -1282 and Simnad and West, 1986).

(c) Materials Testing Reactor (MTR)-Type Fuel

To prevent fuel swelling there should be burnup limitations on the fuel. Aluminum-clad aluminum-matrix MTR-type fuel plate non-power reactors should have technical specifications that limit uranium-235 burnup or fission density. The specifications are acceptable if they are consistent with the SAR, which accounts for all relevant thermal-hydraulic and metallurgical considerations. NRC is specifically concerned with the maximum burnup limit for plate-type fuels because of the buildup of oxide on the fuel cladding. This can be a concern when applicants apply to increase the maximum acceptable burnup. The increased resistance to heat transfer to the coolant may affect consequences considered in Chapter 13 of the SAR on accident analyses. NRC has accepted uranium burnup densities in fuels based on a uranium aluminide matrix up to a fission density of 2.3×10^{21} fissions/cm^3 and up to 50 percent of the initial concentration of uranium-235 (Beeston et al., 1980; Gibson, 1967; Nazare et al., 1975; Stahl, 1982).

(d) PULSTAR Fuel

Burnup of pin-type PULSTAR fuel should be limited by a specification based on testing and the SAR analysis. NRC has accepted burnup limits up to 20,000 MWd/ton uranium.

LCOs are acceptable if they are analyzed in the SAR and consistent with the values given above. The analyses should verify for these fuel parameter conditions that the fuel will not exceed safety limits for normal and off-normal operations.

3.2 Reactor Control and Safety Systems

(1) Operable Control Rods

The number and type of operable control and safety rods should be specified. No minimum number of operable control and safety rods is prescribed for non-power reactors. The specification regarding the number of operable control rods is acceptable if the excess reactivity and shutdown margin specifications required by the SAR analyses can be ensured for all operating conditions. The individual or total reactivity worths need not be specifically listed. A rod of lesser worth might be designated the "regulating rod" and is used as a fine power adjustment mechanism. In some cases, the worth of a control rod(s) connected to an automatic control system (which can add reactivity) may be limited to a maximum amount that was assumed in the SAR in this LCO. This regulating rod need not have scram capability, but rods without scram capability should not be used when showing compliance with shutdown margin requirements. Other rods of greater worth, with an automatic protective (scram) function, should be capable of achieving the specified shutdown margin.

The maximum scram time should be specified for each scrammable rod. The specification should ensure that the drop times are consistent with the SAR analysis of reactivity required as a function of time to terminate a reactivity addition event accounting for measurement and calculational uncertainties. In most non-power reactors, for rods 2 to 3 feet long, full rod insertion time in the absence of excess mechanical friction or interference is less than 1 second. If a specification proposes a longer scram time, it requires appropriate SAR analysis. NRC finds it acceptable to shut down a non-power reactor by intentionally scramming the control and safety rods.

(2) Reactivity Insertion Rates

The maximum rates of adding positive reactivity should be specified for the control and safety rods. The specification should explicitly state that gang or multiple rod withdrawal is allowed. Control rod(s) connected to an automatic control system may have maximum rates of reactivity addition that differ from the rest of the control rods. The acceptable rates should be based on the SAR, including inadvertent addition of ramp reactivity at the maximum rate for the most conservative power, rod position, and reactor conditions.

(3) Pulsed Operation

Limitations on reactivity additions are discussed in Section 3.1(3) above and need not be repeated here. If any hardware systems require special limitations for pulsing, they should be discussed in this section of the technical specifications. Examples might include (a) special core configuration, (b) specific location of pulse rod, (c) number of pulse rods, and (d) removal of in-core fueled experiments. These specifications are acceptable if the assumptions of the SAR are ensured and damage to the reactor by authorized pulses is precluded.

If an experiment containing fissionable material could be damaged by reactor pulsing, limiting specifications must be provided in Section 3.8 of the technical specifications to preclude the event.

(4) Scram Channels

A table should specify all required scram channels and setpoints, the minimum number of channels, other functions performed by the channel, and reactor operating mode, such as steady-state power or pulsed, and cooling method, such as forced- or natural-convection coolant flow. The safety limits that the scram protects should be discussed in the basis for the table. Table 14.1 shows how the information could be displayed. Reactor scrams should be based on the SAR. There should be at least two completely independent power level scram channels and they should provide diversity and redundancy.

Historically, there have been cases in which NRC has accepted power level scrams higher than the licensed power (1.2 times licensed power level is common) if supported by the safety analysis. This power level is the only non-power reactor scram setpoint that, if reached, violates the license (maximum power level). Some licensees have incorrectly interpreted this scram setpoint at higher than the licensed power level as allowing limited operation above the license power level. Although this operation is generally not a safety concern, the NRC staff recognizes it as a regulatory problem. For example, if the reactor power measuring channels are out of calibration, it is possible that the reactor has been operated at several percent above the maximum licensed power level for a period of time. To ensure that licensed power levels are not exceeded for non-power reactor operation, the applicant may consider the option of having an LCO which has a power level scram set below the licensed power level. The NRC staff has accepted another option and, upon submittal of a license amendment request and supporting safety analysis, has approved license amendments for non-power reactors that raise the licensed power 10 percent above the power level at which the reactor will be operated. The applicant can then set the reactor scram in this 10-percent power band. This allows the operating power to remain the same while retaining the scram setpoints within the license power limit. Safety limits and associated LSSS

Table 14.1 Typical required scrams and power reverses*

Channel	Set point** and Function	Minimum Number Required
Period safety	Scram if period ≤ 3 sec	1
Period reverse	Rod run in if period ≤ 10 sec	1
Power level safety (linear and safety)	Scram if power > 100%	2
Power level reverse (safety)	Rod run in if power > 97%	1
High power/no coolant flow	Scram if flow < 56.8 l/sec (900 gpm) and power > 100 kW	1
High power/flapper open	Scram if power > 100 kW and flapper is open	1
Flapper closed/no coolant flow	Scram if flow < 56 8 l/sec (900 gpm) and flapper is closed	1
Software (digital) malfunction	Scram upon malfunction	1
Loss of high voltage to detectors	Scram if voltage is lost	1
Pool water level	Scram if level < 4.88 m (16 ft) above core top	1
Bridge not clamped	Scram when clamps are released	1
Bridge radiation level and building exhaust air radiation level	Scram if radiation ≥ 50 mrem/hr and concentration ≥ 2×10^{-6} µCi/ml	1 1
Manual scram switch	Scram when switch is depressed	1
Rod magnet power keyswitch	Scram when magnet power is turned off	1
Fuel temperature	Scram if temperature ≥ 550 °C (1022 °F)	2
Reactor coolant exit temperature	Scram if temperature ≥ 55 °C (131 °F)	1
Automatic control system out of limit	Rod run in if out of specification	1
Experiment	Scram if setpoint is violated	1
Loss of site power	Scram if power is lost	1

*As illustrative values, the setpoints and channels listed do not apply to any one reactor
**Values listed are limiting setpoints For operational convenience, setpoints may be changed to more conservative values

based on power are still determined by the results of the analysis in the SAR and should not change.

The applicant can choose to continue to have power level scram setpoints above the licensed power level and can meet the license power level requirements by such other means such as administrative limits.

(5) Interlocks

Required interlocks that inhibit or prevent control rod withdrawal or reactor startup should be specified by a table (see Table 14.2 as an example). Interlocks should be specific to the facility and should be based on the SAR. These interlocks include the following:

- operability of area or other radiation monitors
- experimental facilities
- confinement and ventilation systems
- initial conditions for pulsing
- detected neutrons for startup
- operability of measuring channel components, such as ion chamber power supplies and recorders as discussed in the SAR

Table 14.2 Typical required interlocks*

Channel	Minimum Number Required	Function
Recorders not operating	3	Prevent rod withdrawal (startup inhibit)
Neutron count rate (startup)	1	Prevent rod withdrawal (startup inhibit) if count rate ≤ 2 cps
Simultaneous rod withdrawal	5	Prevent withdrawal of 2 or more rods
Nonpulse condition	1	Prevent movement of pulse rod in steady-state mode
Pulse withdrawal	4	Prevent movement of standard control rods in pulse mode
Transient withdrawal	1	Prevent movement of pulse rod with reactor power above 1 kW

*Values listed are limiting setpoints For operational convenience, setpoints may be changed to more conservative values

If the reactor will be licensed to operate in more than one mode, the specification should include the mode for which the interlock is required. If permanent interlocks are established for special experiments, shields, or access control, they should be included in the technical specifications, as described in the SAR.

(6) Backup Shutdown Mechanisms

Most non-power reactors are required to use only control and safety rods for shutdown. If the SAR identifies a need for backup mechanisms (e.g., moderator dump in a critical facility), they should be specified with appropriate requirements placed on their operability (LCOs).

(7) Bypassing Channels

Any individual channels identified in items 4, 5, or 6 (above) for which bypassing is allowed during reactor operation should be justified in the SAR and specified under this item. Only minimal bypassing should be permitted in safety systems and never in a system that could compromise scram capability of the other channels. Bypassing temporary scrams or interlocks associated with experiments need not be included in the technical specifications but should be addressed in specific experiment protocol.

(8) Control Systems and Instrumentation Requirements for Operation (Added by NRC)

Technical specifications for non-power reactors should have redundant and accurate power level monitors that cover the range from subcritical source multiplication to above the full power level. Not all monitors are required to include scram capability (see Table 14.3 for a typical minimum set). These include a startup channel, linear power monitor, logarithmic power monitor, and safety channel(s). In addition, most non-power reactors have a period channel (meter), including a period scram. One should be specified as analyzed in the reactor transient response section of the SAR.

Some non-power reactors with forced-convection cooling have a channel that displays the radiation level of nitrogen-16 in the primary flow. Although the nitrogen-16 channel is not required in the SAR for mitigating transients, it can be an important channel because (a) it has greater stability than delta temperature across the core during power changes and (b) it is not affected by changes in core flux distribution caused by fission product buildup in the core that can affect the ionization chambers. If it is necessary for the operators to use the nitrogen-16 channel in reactor operations, it should be on the list of specified channels.

Table 14.3 Typical required minimum measuring channels*

Channel	Minimum Number Required	Function
Startup	1	Monitor subcritical multiplication for startup
Power level	2	Input for safety power level scram
Pulse power	1	Input for pulse power level scram
Fuel temperature	2	Input for fuel temperature scram
Log N/period	1	Wide range power level and input for period meter and period scram
Linear power level	1	Display power for control
N-16	1	Display power level

*As illustrative values, these channels do not apply to any one reactor. Minimum channels for a particular facility are determined from the SAR analysis.

In past cases where digital control and safety instrumentation was used, an analog reactor protection system was specified in the technical specifications in addition to the digital system to provide diversity and redundancy. The technical specifications for digital systems (including the degree of diversity and redundancy needed) are based on the analysis in Chapter 7 of the SAR, "Instrumentation and Control Systems."

Specifications in this section should cover the entire channel, including readout meters and recorders and the protective functions they perform, such as to prevent an LSSS from being exceeded.

Each non-power reactor should have more than one power level channel indication in the control room when operating at full power. However, because sensors and channel electronics might not be identical, the channels may indicate slightly different power levels. Power level is a principal license condition, and each applicant may consider designating a primary channel for power level monitoring. That channel should be calibrated for thermal power in the region of maximum licensed power and should be recorded in a way that allows auditing for later proof of authorized operation within the license condition. Facility procedures should identify this designated channel and allow for alternative designations using analytic comparisons to achieve operational flexibility, if necessary. Technical specifications or facility procedures that do not include this concept are acceptable to NRC.

3.3 Coolant Systems

The basic systems required for cooling the fuel and other components, for limiting corrosion, and for monitoring coolant radioactivity in non-power reactors should be specified in this section. All non-power reactors should have the capability to remove both fission and decay heat to ensure fuel integrity under all potential conditions. All reactors having forced-convection cooling systems should have specifications ensuring operability of systems and reactor configurations for fail-safe changeover from normal forced-convection to emergency forced-convection cooling or natural-convection cooling. An adequate heat sink, as described in the SAR, is a necessary component of such a system.

For reactors licensed to operate in both forced- and natural-convection cooling modes, the appropriate coolant system configurations and the relevant power levels for both modes should be specified as analyzed in the SAR.

Not all of the following items apply to all types of non-power reactors. However, when applicable, they should be limited by technical specifications on the basis of the analyses and justifications in the SAR.

(1) Shutdown Cooling or Pump Requirements

At a minimum, the requirements for natural-convection cooling and the operability and status of related systems required for shutdown should be specified as LCOs. If additional requirements are necessary for temporary forced-convection cooling following reactor shutdown from extended high-power operation, the technical specifications should state them, using the SAR as the basis.

(2) Isolation Valves

The existence, location, operability, and status of any valves required to isolate subsystems or components for operational needs, including removal of decay heat, should be specified as LCOs.

(3) Coolant Level Limits

Both the coolant pressure (boiling temperature at the fuel) and adequate natural-convection flow depend on the level of water above the core. In addition, vertical and horizontal radiation shielding by the coolant might be necessary. Pool water level should be an LCO for both reasons, using the SAR as the basis.

(4) Detection of Leakage or Loss of Coolant

If primary system leakage or other loss of coolant could lead to an uncontrolled release of radioactive material (see items 5 and 8 below) to the environment, an LCO should state the need for operability of leakage detection systems. Depending on reactor design, the applicant may also want to detect loss of coolant to prevent fuel damage. An example of this type of system would be instrumentation that monitors the pressure difference between the primary and secondary cooling systems at the heat exchanger to detect conditions that would allow loss of primary coolant in the event of a heat exchanger leak. Another example is heavy water systems that may have detectors located on pumps and piping to detect leakage.

(5) Detection of Fission Product Activity

The technical specifications should provide for prompt detection of fission products escaping from the fuel barrier. The method could be a radiation detector placed in the primary coolant flow loop or a strategically located continuous air monitor in the reactor room or in a ventilation duct. Temporary substitutions, in case the fission product monitor is inoperable, should follow guidance in Section 3.7.1 of ANSI/ANS 15.1. This specification may be combined with the specification discussed in Section 3.7.1(2) on fission product monitors.

The specified fission product monitor should be able to initiate action, such as a reactor scram, reactor room isolation, or an alarm, as appropriate. The SAR should provide the bases and describe how fission products are distinguished from other waterborne or airborne radioactivity.

(6) Hydrogen Concentration (Off-Gas) Limits

If the SAR has shown any of the isotopes of hydrogen (hydrogen, deuterium, and tritium) to be a significant risk to personnel or the facility, an LCO should provide for detection or adequate control, as discussed in the SAR.

(7) Emergency Core Cooling Systems

If the SAR indicates a need for supplemental core cooling to mitigate a loss-of-primary-coolant event, the technical specifications should contain an LCO requiring an operable and adequate system. The system should satisfy the cooling requirements for the SAR scenario and should not depend on continued availability of normal electrical service.

(8) Secondary and Primary Coolant Radioactivity Limits

In addition to the prompt detection of fission products from failed fuel or experiment malfunctions [see Section 3.3(5)], LCOs should limit radioactivity in the coolant. The technical specifications should require periodic sampling and appropriate analyses to detect and quantify radioactivity in both the primary and secondary coolant. The coolant should be sampled for gross activity at a short interval, for example, weekly, and sampled for isotope identification at a longer interval, for example, quarterly. Trending the output of the fission product monitor or other primary coolant continuous radiation monitor may substitute for weekly gross activity sampling of primary coolant if justified by the applicant. The purpose of this LCO is to detect deterioration of components in the primary coolant loop, such as a control element, and leakage in a heat exchanger into the secondary coolant loop These specifications should be stated in such a way that significant changes in radioactivity, as defined in the SAR, trigger remedial action.

(9) Water Chemistry Requirements

To control (a) corrosion of such components as the reactor fuel, structure, and pool and (b) activation of impurities in the reactor coolant, and to maintain visual clarity of the reactor coolant, there should be LCOs on both electrical conductivity and pH of the primary coolant. These specifications also should apply to any water that comes into contact with the fuel, such as water in fuel storage tanks and pits. The explicit limits and ranges of values should be given and should be consistent with recommended values given by both fuel vendors and in water chemistry guidelines The conductivity should be monitored continuously. There should be a definite schedule for measuring pH, during both operating and shutdown periods. The bases should clearly address the appropriate ranges and give meaningful references. The SAR should justify the values for conductivity and the pH for the particular reactor. Acceptable ranges for these process variables have traditionally been ≤ 5 μmhos/cm for conductivity and between 5.0 and 7.5 for pH. These values can usually be achieved by demineralization, filtration, and good housekeeping practices, but chemical methods should be described and specified, if applicable.

3.4 Containment or Confinement

Because accidents that result in both release of steam and building overpressure are uncommon, most non-power reactors are housed in a confinement, not a containment. There should be an LCO requiring that the system specifically described in the SAR exist as stated. The system should be operable during operation and for other applicable times such as before operation and following shutdown, as noted in Sections 3.4.1 and 3.4.2 of ANSI/ANS 15 1. If interlocks or administrative controls to ensure operability are required, there should be

appropriate specifications. Whether the facility has a confinement or a containment depends on the reactor design, operating characteristics, and facility location. Specifications should require nominal exhaust rates for air under the operating and accident conditions analyzed in the SAR. Specifications should limit building leak rates to those described in the SAR.

3.5 Ventilation Systems

Ventilation and exhaust flow rates and the systems to achieve the controlled release of effluents, as analyzed in the SAR, should be specified as LCOs. These LCOs should be established to achieve controlled release of effluents. Automatic failsafe closure of vents should be specified for confinement systems. Provisions to initiate controlled, filtered, and monitored exhaust and ventilation for radiological accidents should be included. In some cases, depending on the results of the analysis, minimum airflow rates may be LCOs.

The ventilation system should maintain a lower air pressure in the reactor room than in adjacent spaces. Air in the reactor room should not be distributed to other occupied spaces within buildings. The location and height of the air exhaust system stack or release point should be specified as an LCO here or as a design feature in Section 5 of the technical specifications. The dimensions of the stack should be consistent with the assumptions used in the SAR to predict potential radiation doses in the unrestricted environment. It is acceptable that the concentration of airborne radioactivity at the point of exhaust for normal operation be higher than the regulatory limit for restricted areas, provided that this point is not readily accessible to the public, the analyzed doses to the public are well below regulatory limits for unrestricted areas, and the potential doses to the facility staff are within regulatory limits. The as low as is reasonably achievable (ALARA) program should be applied in all analyses (see Section 3.7 below).

3.6 Emergency Power

Any requirement for emergency electrical power for non-power reactor facilities should be analyzed in the SAR on a case-by-case basis. Any necessary facility functions, such as radiation monitoring, emergency core cooling, or isolating the containment or the confinement, that need to be maintained if normal electrical power is lost should be described in the SAR. If emergency power is required, an LCO should ensure operability of the system. The technical specification should specify automatic startup of emergency electrical power if automatic startup is indicated in the SAR.

3.7 Radiation Monitoring Systems and Effluents

Monitoring systems and effluents may be addressed in the technical specifications under separate principal headings. The following discussion is consistent with the corresponding sections of ANSI/ANS 15.1.

3.7.1 Monitoring Systems

A separate table in the technical specifications (see Table 14.4) should list the required radiation monitors, the function each performs (e.g., scram or containment isolation), the approximate location of each, the type of radiation detected, and the alarm and/or automatic action setting, as analyzed in the SAR. The setpoints and calibrations should be listed in terms of radiation exposure rates and concentrations rather than as count rates that can change with calibration. Specific count rates for alarms and action settings can be presented in a facility procedure that can be amended in accord with the procedures section in the technical specifications. For specified monitors that become inoperable, the specification should state that reactor operations may continue only if the monitor is replaced by a substitute or portable monitor. The replacement monitor should perform essentially the same function until the original monitor is repaired or replaced (generally not to exceed 1 work week unless justified in the SAR). The specification also should state that if the specified monitor was displayed in the control room, the operator on duty should also be able to observe the temporary monitor. The applicant should provide a table applicable to the specific facility on the basis of the SAR.

(1) Air Monitors (Gas and Particulate)

Monitors should be specified for both radioactive gas and those radioactive particulates that might be airborne in the reactor room. There should be at least one continuous air monitor (CAM) with an audible alarm and data recorders. These monitors should be capable of alerting facility personnel to the presence of radioactivity. They should be calibrated for anticipated radioactive species. Potential sources of airborne radioactivity should be analyzed in the SAR.

There should be specifications requiring operability of properly calibrated effluent monitors, preferably with recorded outputs for long-term records that provide documentation of the concentration and total quantity of radioactive effluents, as discussed in Section 3.7.2 below.

For reactors operated at power levels below a few hundred kilowatts, the concentrations of airborne radionuclides may be too low to measure during normal operation. For these, calculated concentrations of released quantities are acceptable as specifications, using the SAR as the basis.

Table 14.4 Typical required radiation measuring channels*

Channel	Minimum Number Required	Function	Setpoint Equal to or Less Than
Area radiation monitors	4	Alarm	0.15 mSv/hr 15 mrem/hr
Hot cell monitor	1	Alarm and door interlock	1 mSv/hr 100 mrem/hr
Reactor bridge	1	Alarm (isolates containment with building particulate)	0 5 mSv/hr 50 mrem/hr
Primary coolant	1	Alarm	0.5 mSv/hr 50 mrem/hr
Building particulate	1	Alarm (isolates containment with reactor bridge)	2×10^{-9} µCi/cm^3 2-hr particulate
Building gas (Argon-41)	1	Alarm	2×10^{-5} µCi/cm^3 daily release
Stack particulate	1	Alarm	2×10^{-8} µCi/cm^3 2-hr particulate
Stack gas (Argon-41)	1	Alarm	2×10^{-4} µCi/cm^3 daily release 4×10^{-8} µCi/cm^3 annual average

*As illustrative values, these channels and setpoints do not apply to any one reactor. Setpoints for a particular facility must be determined in the SAR analysis

(2) Fission Product Monitors

The specified fission product monitor could be the CAM or the primary coolant monitor, depending on the release scenarios analyzed in the SAR. Release of fission products from both fuel and fueled experiments should be included. This specification may be combined with the specification discussed in Section 3.3(5) above.

(3) Area Monitors

There should be a specification requiring operable area monitors in and near the reactor room. The type of radiation detected, such as gamma rays or neutrons, should be specified. Brand names, efficiencies, and specific designs should be

avoided as specifications, but the range of exposure rates monitored may be specified. These area monitors should give information on the potential exposure rates from reactor-related radiation. Alarm and automatic action setpoints should be specified to ensure that personnel exposures and potential doses remain well below limits of 10 CFR Part 20 and are consistent with the facility ALARA program.

(4) Environmental Monitors

There should be at least one environmental monitoring station near the facility, preferably at the site boundary or at other areas of concern, such as at population centers or student dormitories. These monitors should be specified to match the types of radiation anticipated and should be either in the line of sight from the air exhaust point or downwind in the prevailing wind, as appropriate. The types of monitors should be specified [see Section 3.7.1(4) of ANSI/ANS 15 1]. The location and method of determining background readings should be discussed in the SAR and in the basis of the specification. The specification should state that environmental monitors are used to verify that the potential maximum dose, annual or other, in the unrestricted environment is within the values analyzed in the SAR. The specification should address both potential accident scenarios and normal operations.

3.7.2 Effluents

All radioactive species listed in Section 3.7.2 of ANSI/ANS 15.1 that are released by the facility should be addressed for normal operations, and the releases should be limited by technical specifications. NRC accepts the proposed concentration limits, provided the SAR shows that potential doses from these concentrations comply with 10 CFR Part 20 for the maximum exposed member of the public on a facility-specific basis. If the applicant proposes to limit release to 10 CFR Part 20 limits at the point of release, then the analysis of effluents in the safety analysis report is sufficient and no technical specifications need be proposed.

Argon-41 is the principal radionuclide released by most non-power reactors. Even though the doses related to argon-41 are generally small, a specification should address the average and maximum concentrations in both the restricted and unrestricted areas and the total curies (becquerels) released during a calendar year. The calculated potential doses to the most exposed persons in restricted and unrestricted areas must conform with 10 CFR Part 20 and the facility ALARA program.

Because of diffusion and dispersion of the release, if the point of release is inaccessible to the public and generally not accessed by facility staff, the maximum normal concentration at that location may be higher than the concentration allowed

in 10 CFR Part 20 for restricted areas. The SAR should show that the diffused and dispersed release at the point of contact with members of the public is within 10 CFR Part 20 limits. The calculations in the SAR for diffusion and dispersion should be realistic but conservative, and should be based on logical models and specified effluent levels. Because an infinite cloud assumption is extremely conservative for argon-41 releases, a finite cloud should be considered, as discussed in NUREG-0851 and accepted by NRC.

3.8 Experiments

Experimental facilities should be described in the SAR, and their basic features should be included in Section 5 ("Design Features") of the technical specifications. The experiments to be performed in the experimental facilities need only be noted briefly, if at all, in the SAR, unless they could present a hazard to the reactor facility, the public, or facility staff. Any LCOs for experiments should be performance based to ensure that no regulations are violated, that experiment safety analysis limits are not exceeded, and that the reactor is not damaged by experiment failure or malfunction.

Regulatory Guide (RG) 2.2, provides detailed guidance to applicants on the scope of the discussions for experiments to be included in the SAR. The regulatory guide also provides guidance on the technical specifications needed to govern the experiments performed. The technical specifications should follow the guidance of Section 3.8 of ANSI/ANS 15.1 and Section C of RG 2.2, as supplemented by the guidance that follows.

3.8.1 Reactivity Limits

Limits should be specified on absolute values of reactivity associated with each type of experiment: secured, unsecured, and movable (see ANSI/ANS 15.1 and RG 2.2 for definitions). Generally, the limits on secured experiments should be approximately twice the limits on unsecured and movable experiments, where the latter should be no more than 1 dollar. The 1-dollar limit is such that inadvertent prompt criticality is avoided even if the experiment were to fail. Movable experiments must be clearly defined to include those to be inserted or removed while the reactor is operating. Unsecured experiments include those installed before reactor startup that change position or change other conditions while the reactor continues to operate. Reactivity limits of experiments that change position while the reactor is operating should not exceed the ability of the reactor operator or automatic servo system to maintain control of the reactor. The specified reactivity limits on movable experiments should not permit the violation of the shutdown margin specification.

The specified sum of the absolute values of the reactivity worths of all experiments should not be more than twice the limit on individual secured experiments. The value should be consistent with the SAR analysis of inadvertent reactivity insertions, as explained in Section C.1.a. of RG 2.2.

There should be a specification requiring that the reactor be shut down during the changing or moving of any secured experiment.

3.8.2 Materials

For fissile materials in experiments, limits should be specified on the allowed thermal power and on the equilibrium or maximum inventory of specific fission products, such as iodines and strontium. Specifications such as those indicated in Section C.2.a of RG 2 2 are acceptable.

A specification should require double encapsulation of potentially corrosive materials. All liquid and gas samples should be analyzed to determine if they require double encapsulation considering such factors as (1) the effect of failure on the reactor, (2) the radiological consequences of failure on the facility staff, the public, and the environment, and (3) the possibility that the failure would result in an industrial hazard that could affect safe reactor operation. The failure of an encapsulation of material that could damage the reactor should require removal and physical inspection of potentially damaged components.

Specifications should limit the quantity of explosive material permitted in the experimental facilities and elsewhere in the reactor facility. For experimental facilities, the upper limit should be 25 mg TNT or its equivalent, as indicated in Section C.2.d of RG 2.2. For the overall reactor facility, the upper limit should be no higher than 100 mg TNT or its equivalent, unless a larger quantity is analyzed in the SAR and approved by NRC. An additional specification should require prior testing or analyses of explosive material encapsulations to ensure no reactor damage in the event of detonation, regardless of the limit.

A specification should limit the quantities of unknown materials that could be placed in certain experimental facilities for exploratory studies. Conformance with Section C.2.i of RG 2.2 would be acceptable.

3.8.3 Failure and Malfunctions

Specifications that address the failure and malfunction of an experiment and limit the experiment parameters should be included on a case-by-case basis, as discussed in the SAR. The guidance of Section 3.8.3(2) of ANSI/ANS 15.1 should be followed, but specifications that require compliance with regulations are redundant and are unnecessary [see Section 3.8.3(1) of ANSI/ANS 15.1].

For experiments that may off-gas, sublime, volatilize, or produce aerosols, standard assumptions are often specified for calculating the activity that could be released under normal operating conditions, accident conditions in the reactor, and accident conditions in the experiment. Such specifications ensure conservatism in the safety analysis of the experiment. These specifications have contained such assumptions as the following: (1) if an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100 percent of the radioactive gases or aerosols escape; (2) if an effluent holdup tank isolates on a high radiation signal, at least 10 percent of the radioactive gases or aerosols escape; (3) if the effluent exhausts through a filter with 99-percent efficiency for 0.3-micron particles, at least 10 percent of the vapors escape; and (4) if an experiment fails that contains materials with a boiling point above 130 °F (54 °C), the vapors of at least 10 percent of the materials escape through an undisturbed column of water above the core. Any particular assumptions used should be derived from the SAR.

Applicable limits for specific experiments are normally not part of the technical specifications and should be derived from the experiment safety review discussed in Section 6.5 below.

3.9 Facility-Specific LCOs

The LCOs discussed above apply to most non-power reactors. Each reactor may also have technical specifications containing facility-unique LCOs. These should be based on the SAR and facility design.

4 SURVEILLANCE REQUIREMENTS

Certain LCOs established in Section 3 of the technical specifications should be accompanied by a surveillance requirement in Section 4. These surveillance-related specifications should clearly identify the parameter or function to be measured or tested, the method, the frequency, and the acceptable deviation or error. Acceptable deviations could be limited by license conditions (such as thermal power level) or by regulations (such as 10 CFR Part 20).

NRC accepts the surveillance frequencies stated in this section of ANSI/ ANS 15.1 as amplified in the following sections. The actual wording of the specifications should not be ambiguous. Wording in ANSI/ANS 15.1 has been interpreted incorrectly by some licensees to allow the extended interval (interval not to exceed statement) as the average. If the extended interval is used for a particular surveillance test, a shorter interval should be used as soon afterwards as possible to adhere to the average. The intervals of ANSI/ANS 15.1 should be listed in the applicant's technical specifications.

In addition to surveillance verification of LCOs, other surveillance activities should be specified. These include such specifications as periodic pulse rod maintenance and cleaning, thermal power level calibration, preventive maintenance and inspection of control/safety rod drive systems, fuel element inspections, preventive maintenance on other important components to provide assurance of operability, and calibration of effluent monitoring systems.

If a surveillance is not required for safety while the reactor is shut down, it may be deferred, but must be performed before reactor startup. If the reactor is not to be operated in a particular mode (e.g., pulse mode) for an interval that exceeds the surveillance intervals for that particular mode, surveillances not required for safety (an example is the requirement for a standard pulse to be performed every year) while the reactor is operated in other modes may be deferred, but must be performed before the reactor is considered operational in the mode in which surveillances were deferred. Scheduled surveillances that cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown. Surveillances that may be deferred and the reasons for deferment should be clearly stated in the technical specifications, justified in the SAR, and noted in the basis of the specification.

In general, any time that a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due. This special surveillance may change the due date of the next regularly scheduled surveillance of that type.

4.1 Reactor Core Parameters

The excess reactivity and shutdown margin LCOs specified in Section 3 of the technical specifications are applicable for all authorized operating conditions. As an example, for a movable experiment, the specifications for excess reactivity and shutdown margin surveillance measurements should be based on that experiment being in its most reactive location In addition, other reactor parameters that affect reactivity during operation should be explicitly specified. For the following specified surveillance requirements, the parameters may be determined by an appropriate combination of measurements and calculations.

(1) Excess Reactivity

Excess reactivity should be determined at least annually and after changes in either the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

(2) Shutdown Margin

The shutdown margin should be determined at least annually and after changes in either the core, in-core experiments, or control rods.

(3) Pulse Limits

The relationship between peak fuel temperature and inserted reactivity for pulses should be determined when changes are made in the core (see item 1 above).

(4) Core Configuration

Limitations on core configurations are intended to ensure that reactor physics and thermal-hydraulic parameters specific to the core are within the limits analyzed in the SAR. Core configuration parameters specified in Section 3.1(4) or in Section 5 of the technical specifications should be met during reactor operations. Therefore, an acceptable surveillance specification is to verify compliance with all applicable specifications in those sections when any change occurs in the reactor core configuration.

(5) Reactivity Coefficients

Section 3.1(5) of the technical specifications limits reactivity coefficients, which are largely determined by reactor design and fuel type. Measuring and verifying reactivity coefficients can be a difficult task. An acceptable schedule for surveillance of reactivity coefficients is at initial reactor startup and when any change in the reactor core configuration or fuel type requires changes in the specifications of Section 5.

(6) Fuel Parameters

All TRIGA fuel should be inspected for damage and all TRIGA non-instrumented fuel should be measured for length and bend at the following frequencies:

- For non-pulsing TRIGA reactors, the fuel should be inspected and measured on at least a 5-year cycle. Approximately 20 percent of the fuel could be inspected and measured annually. If an annual inspection identifies damaged fuel, then the entire core should be inspected and measured.

- For pulsing TRIGA reactors, the fuel should be inspected and measured annually. If the reactor is pulsed infrequently (fewer than 10 pulses annually), the annual inspection requirement may be relaxed if the relaxed scheduled is analyzed and justified in the SAR. If the reactor is pulsed to reactivity insertions over 4 dollars, additional inspection requirements based

on the number of pulses may be necessary. Facilities in this situation should present and justify inspection frequency requirements determined by the fuel vendor.

Routine inspections of fuel used in AGN-201 and PULSTAR reactors have not been required by technical specifications. If the opportunity is presented to conduct an inspection of fuel, such as core disassembly of an AGN-201 or disassembly of a PULSTAR fuel element to replace fuel pins, the licensee should consider taking advantage of this opportunity. This type of inspection need not be a technical specification requirement. NRC may require fuel inspection as a license condition to increase burnup limits on such fuels. This would be determined on a case-by-case basis. These inspections are qualitative, to detect evidence of excessive corrosion/erosion and mechanical wear or damage.

Inspections for reactors with plate fuels have not been required by the technical specifications except for higher power reactors that refuel frequently. However, for reactors that remove plate fuel from service because the fuel has reached its burnup limit, there should be a requirement to inspect representative fuel elements (e.g., 1 in every 10) for excessive corrosion/erosion, mechanical wear or damage, or plate swelling. The surveillance procedures should follow guidance provided by the fuel supplier, if available. In all cases, the specification should describe briefly how the inspection will be performed.

For reactors with technical specification limits or SAR analyses imposing limits on uranium burnup or fission density, confirmatory estimates should be made at intervals during the life of the fuel, such as at 50, 60, and 70 percent of the fuel life or semiannually when the NRC/Department of Energy Form 742 is submitted. The SAR should justify the surveillance method and intervals which ensure that the limit is not exceeded.

For reactors with HEU fuel and subject to the 10 CFR 73.6(b) exemption (self-protection), confirmatory radiation measurements or analyses should be made at intervals justified in the SAR.

4.2 Reactor Control and Safety Systems

(1) Reactivity Worth of Control Rods

The integral and differential worths of all control and safety rods should be determined at initial fuel loading. Integral and differential worths should be determined at least annually and after changes of the core or control rods, as noted in Section 4.1(1) above.

(2) Rod Withdrawal and Insertion Speeds

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(3) Transient Rod and Associated Mechanism

This system should be inspected, disassembled, cleaned, and, if applicable, lubricated annually. The reactor should be pulsed at least annually and after changes to core or control rods, as indicated in Section 4.1(1) above, with a well-documented reactivity insertion. If the reactor is not routinely pulsed, this standard pulse (a reactivity addition whose results are well know) may be deferred for more than a year, but it should be performed before resumption of normal pulsing.

(4) Scram Times of Control and Safety Rods

A specific interval should be stated for the surveillance intervals given in Section 4.2(4) of ANSI/ANS 15.1.

(5) Scram and Power Measuring Channels

Channel tests of all scram and power measuring channels required by technical specifications, including scram actions with safety rod release and interlocks, should be performed before each reactor startup following a shutdown of more than 24 hours or following each secured shutdown. If the reactor operating schedule calls for no secured shutdowns, the channel tests should be performed at least quarterly. Many facilities perform these tests before each reactor startup and NRC recommends this practice.

(6) Operability Tests

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(7) Thermal Power Calibration for Reactors Cooled by Forced Convection

Thermal power should be calibrated at least annually; heat balance should be verified at least monthly.

(8) Thermal Power Calibration for Reactors Not Cooled by Forced Convection

Thermal power should be calibrated at least annually. The basis should indicate the method to be used.

(9) Rod Inspection

The rod-drive and scram mechanisms of each control and safety rod should be inspected annually. The poison sections of control and safety rods should be inspected biennially for indications of deterioration or damage. This can be a visual inspection or an inspection that requires the rod to pass through a measuring device which detects swelling.

4.3 Coolant Systems

Only a small fraction of the licensed non-power reactors will need to consider all of the following surveillance items. The applicant should discuss the applicability and identify the parameters that should be tested. The applicable parameters should be specified, and the functions should be explicitly stated in the specification.

(1) Starting Function of Emergency Shutdown and Sump Pumps

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(2) Test of Emergency Coolant Sources and Systems

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(3) Inservice Inspections

If any inservice inspections of cooling system components are identified in the SAR, they should be performed according to the manufacturer's recommendations. If the manufacturer's recommendation is not available, the frequency should be as established in the SAR from engineering judgment and similar component inservice inspection requirements and experience.

(4) Analysis of Coolants for Radioactivity

Analyses for isotope identification of primary and, if applicable, secondary coolant should be performed by sampling quarterly. Sampling weekly for gross analysis should be considered to establish trends to quickly identify fuel or heat exchanger failure.

(5) Hydrogen Concentration in Off-Gas

If applicable, this test should be performed at least annually and after maintenance or repair that could affect the system or measurement instrumentation.

(6) Conductivity and pH

When the reactor is operating on a routine schedule, conductivity and pH should be measured at least weekly. This requirement could be met by a system that monitors conductivity and pH continuously while the reactor is operating.

If the reactor is not operated for long periods, the interval between conductivity and pH measurements may be increased to monthly if reasonable justification is provided in the SAR.

If fuel is stored in water in separate fuel storage from the reactor pool, the pH and conductivity of this water should be measured at regular intervals as determined and justified in the SAR.

(7) Primary Coolant Level

If the primary coolant level above the core is not continuously displayed during reactor operation, the primary coolant level in the pool or tank should be verified daily if the reactor is operating or before reactor startup.

(8) Primary Coolant Sensors and Channels

Channel tests of sensor operability and channels not included elsewhere in the technical specifications that are identified in the SAR should be performed quarterly and before startup after maintenance.

All channels should be calibrated annually and before startup after major modification or component replacement.

4.4 Containment or Confinement

4.4.1 Containment

Few licensed non-power reactors are required to have a containment. For those required by the SAR, the surveillance intervals given in ANSI/ANS 15.1 are acceptable.

4.4.2 Confinement

Confinement is a system that provides a temporary holdup or controlled release of radioactive effluents to the environment. Most non-power reactors are equipped with confinements and should have a functional test of the overall system described in the SAR quarterly. In addition, an efficiency test of the filters should be

performed annually or in accordance with manufacturer recommendations and acceptance criteria.

4.5 Ventilation Systems

Ventilation systems at most licensed non-power reactors are an integral part of the containment or confinement system, and surveillance activities may be interrelated. An operability check, including dampers and blowers, should be performed quarterly and following repair or maintenance to declare the system operable. The function and efficiency of filters should be tested annually or in accordance with manufacturer's recommendations and acceptance criteria and following repair or maintenance to declare the system operable.

4.6 Emergency Electrical Power Systems

For all emergency electrical power systems, channel checks or other operability checks should be performed before reactor startup and after maintenance. Maintenance should be performed according to the manufacturer's recommendations. If the manufacturer's recommendations are not available, the frequency should be as determined in the SAR.

4.6.1 Diesels and Other Devices

The shorter of the surveillance intervals given in this section of ANSI/ANS 15.1 is acceptable.

4.6.2 Emergency Batteries

The shorter of the surveillance intervals given in this section of ANSI/ANS 15.1 is acceptable.

4.7 Radiation Monitoring Systems and Effluents

4.7.1 Monitoring Systems

A channel check should be performed daily before reactor startup. Where physically possible, a channel test using a radiation source should be performed at least monthly. The SAR should describe such capability.

All required radiation monitoring systems, including effluent monitors, should be calibrated at least annually according to the manufacturer's recommendations. Individual systems should have separate specifications.

4.7.2 Effluents

Quantities of radioactive effluents released to the environment are LCOs. If the SAR states that it is not feasible to monitor such effluents from low power reactors in real time at the point of release, calculated releases may be substituted. The SAR should specify surveillance methods and intervals for confirming these releases or for verifying upper limits.

For gaseous airborne radioactive effluent, it is acceptable to confirm annual upper limits by integrating dosimeters such as thermoluminescence dosimeters (TLDs) or film.

For particulate airborne or waterborne radioactive effluent, it is acceptable to confirm annual upper limits by surveillance of environmental factors given in this section of ANSI/ANS 15.1.

4.8 Experiments

If any experiment discussed in the SAR is designed to operate with emergency systems or with connections to the reactor protective systems, a channel check should be specified both daily and before reactor startup when the particular experiment is being performed. Surveillance activities for experiments that are included in the experiment protocol and the review and approval process need not be included explicitly in the technical specifications.

4.9 Facility-Specific Surveillance

There should be applicable surveillance specifications for any facility-specific LCOs in Section 3.9 of the technical specifications not explicitly included in Section 4. These surveillances should be performed to verify significant safety features from the SAR.

5 DESIGN FEATURES

The SAR forms the basis for NRC to issue an operating license for a non-power reactor. Essential information includes the type and enrichment of fuel, core and fuel configurations, fuel storage facilities, thermal power level, potential accident scenarios and mitigating features, environmental conditions at the site, and other factors. To ensure that the issued license remains valid, design features should not be changed without prior NRC review and approval. These major design features are noted in Section 5 of the technical specifications, if they have not already been specified in Section 2 or 3.

The NRC accepts the guidance in this section of ANSI/ANS 15.1. The applicant should provide concise but explicit information on all noted features.

6 ADMINISTRATIVE CONTROLS

The specified information and controls on staffing and operations of the reactor facility will ensure that the management and staff of the facility are acceptably knowledgeable and aware of the technical requirements to operate a safe facility, to comply with regulations and the license conditions, and to practice a meaningful ALARA program, which will protect the environment and the health and safety of the public, the facility users, and the staff.

Not all owners and operators of non-power reactors will have the same management organization or office titles. Regardless of the details of the management organization, or of the complexity of the facility, the administrative functions presented in this section of ANSI/ANS 15.1 should be established and specified. The NRC accepts the ANSI/ANS 15 1 position as modified in the sections that follow.

6.1 Organization

6.1.1 Structure

The information recommended by ANSI/ANS 15.1 should be clearly stated, including how and when the radiation safety staff communicates with the facility manager and level 1 management to resolve safety issues.

6.1.2 Responsibility

NRC accepts the guidance provided in this section of ANSI/ANS 15 1.

6.1.3 Staffing

Applicants should use the terms "reactor operator (RO)" and "senior reactor operator (SRO)" instead of "Class B" and "Class A," respectively (see Figure 1 in ANSI/ANS 15.1).

6.1.4 Selection and Training of Personnel

Compliance with 10 CFR Part 55 is required of the licensee and licensed operators, unless NRC has issued an exemption ANSI/ANS 15 4–1988 provides additional guidance for non-power reactors.

6.2 Review and Audit

The committee established to perform the review function may be assigned approval authority by the facility manager or the facility manager may retain that authority. Section 6.2 of the technical specifications should explicitly state who holds the approval authority and should specify the committee's authority and how it communicates and interacts with management levels 1 and 2.

6.2.1 Composition and Qualifications

One or more voting members of the committee should be from organizations other than the one operating the reactor.

6.2.2 Charter and Rules

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

6.2.3 Review Function

The fact that this section of ANSI/ANS 15.1 addresses the review function required by 10 CFR 50.59 should be explicitly stated in the technical specifications.

6.2.4 Audit Function

In addition to the emergency plan, all other required plans, such as physical security and operator requalification, should be specified for auditing. The requirement to audit these plans may be part of the plan itself. If that is the case, the requirement to audit does not need to be repeated in the technical specifications.

6.3 Radiation Safety

The technical specifications should state that 10 CFR Part 20 establishes requirements that the radiation safety program must achieve. Additional guidance for radiation safety programs at non-power reactors may be found in ANSI/ANS 15.11.

The authority of the radiation safety staff to interdict or terminate safety-related activities should be stated. The technical specifications should state management's commitment to practice an effective ALARA program. This program should apply to facility staff, facility users, the general public, and the environment.

6.4 Procedures

Procedures in addition to those in ANSI/ANS 15.1 should be proposed at facilities to address operational situations recognized in the SAR. For example, if byproduct material whose possession is authorized under the reactor license is used in facility laboratories that are part of the reactor license and/or transferred to other licensees, procedures for control and transfer of this byproduct material should be part of the set of minimum procedures required by the technical specifications. The specifications should be written to ensure a minimum necessary set of procedures, but should allow for future additions as necessary.

The minor modifications and temporary deviations allowed by ANSI/ANS 15.1 should not be spelled out in the technical specifications. However, the methodology for establishing and changing procedures should be stated in the specifications.

6.5 Experiments Review and Approval

In addition to the guidance of ANSI/ANS 15.1, the review and approval of experiments should be consistent with the guidance provided in Section C.3 of RG 2.2 and RG 2.4. The specifications should make clear that "established and approved procedures" means written procedures, properly reviewed and approved. Any changes made to these procedures should conform to Section 6.4 of ANSI/ANS 15.1.

6.6 Required Actions

6.6.1 Action To Be Taken in Case of Safety Limit Violation

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

6.6.2 Action To Be Taken in the Event of an Occurrence of the Type Identified in Sections 6.7.2(1)(b) and 6.7.2(1)(c)

The first sentence of this section of ANSI/ANS 15.1 states that "reactor conditions shall be returned to normal or the reactor shall be shut down." The specification should be written to provide that the applicant establish, in advance, specific criteria for the two alternative actions: shutdown or return to normal. For example, a return-to-normal event is a reactor scram resulting from a known cause, such as an electric transient.

6.7 Reports

6.7.1 Operating Reports

The technical specifications should state that operating reports should be sent to the NRC Document Control Desk and that a copy should be sent to the appropriate regional administrator.

Section 6.7.1(4) of ANSI/ANS 15.1 refers to the reporting required by 10 CFR 50.59. The specification should make reference to the rule.

6.7.2 Special Reports

The technical specifications should state that (1) special written reports of events should be sent to the NRC Document Control Desk and that a copy should be sent to the appropriate regional administrator and (2) special telephone reports of events should be made to the NRC Operations Center and the regional staff.

6.8 Records

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

7 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors." ANS, LaGrange Park, Illinois, 1988.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1987

Beeston, J. M., R. R. Hobbins, G. W. Gibson, and W. C. Francis, "Development and Irradiation Performance of Uranium Aluminide Fuels in Test Reactors," *Nuclear Technology* 49, pp. 136–149, June 1980.

Gibson, G. W., *The Development of Powdered Uranium-Aluminide Compounds for Use as Nuclear Reactor Fuels*, IN–1133, Idaho Nuclear Corporation, Idaho Falls, Idaho, December 1967.

Nazare, S., G. Ondracek, and F. Thummler, "Investigations on UAlx-Al Dispersion Fuels for High-Flux Reactors," *Journal of Nuclear Materials* 56, pp. 251–259, 1975.

Simnad, M. T., et al., "Fuel Elements for Pulsed TRIGA Research Reactors," *Nuclear Technology* 28, p. 31, 1976

Simnad, M. T., et al , "Interpretation of Damage to the FLIP Fuel During Operation of the Nuclear Science Center Reactor at Texas A&M University," GA–A16613, GA Technologies, Inc., San Diego, California, December 1981.

Simnad, M. T., and G. B. West, "Postirradiation Examination and Evaluation of TRIGA LEU Fuel Irradiated in the Oak Ridge Research Reactor," GA–A18599, GA Technologies, Inc., San Diego, California, May 1986

Stahl, D., *Fuels for Research and Test Reactors, Status Review*, ANL-85-5, Argonne National Laboratory, Argonne, Illinois, December 1982.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," November 1973.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.4, "Review of Experiments for Research Reactors," Revision 0-R, July 1976.

U.S. Nuclear Regulatory Commission, NUREG-0851, "Nomograms for Evaluation of Doses From Finite Noble Gas Clouds," January 1983.

U.S. Nuclear Regulatory Commission, NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," August 1987.

U.S. Nuclear Regulatory Commission, NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors," July 1988.

West, G. G , M. T. Simnad, and G. L Copeland, "Final Results From TRIGA LEU Fuel Postirradiation Examination and Evaluation Following Long-Term Irradiation Testing in the ORR," *International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, Tennessee, November 3–6, 1986*, ANL/RERTRI-TM9, pp. 259–266, Argonne, Illinois, May 1988.

15 FINANCIAL QUALIFICATIONS

In this chapter of the SAR, the applicant should present the financial information submitted to NRC for a non-power reactor license to establish that the applicant is financially qualified to own, construct, operate, and decommission a non-power reactor. This information should be submitted along with the application for a construction permit and an initial operating license or along with the application for license renewal. Financial qualifications cover three areas:

- financial ability to construct the non-power reactor facility authorized by the construction permit
- financial ability to safely operate the facility
- financial ability to safely decommission the facility so that NRC can terminate the facility license at the end of the facility's use

Financial information to be submitted by the applicant is discussed in 10 CFR 50.33(f) and (k). The cover letter for the construction permit and operating license application, or for the license renewal application, can refer to this chapter of the SAR for complete financial information. If the applicant considers its financial information to be proprietary, an affidavit in accordance with 10 CFR 2.790 shall be submitted to request that the information be withheld from the public. If possible, a non-proprietary version of the financial information should also be submitted.

15.1 Financial Ability To Construct a Non-Power Reactor

An applicant for a construction permit to build a non-power reactor should submit information which demonstrates that the applicant possesses, or has reasonable assurance of obtaining, the funds necessary to cover estimated construction costs and related fuel cycle costs. This information should include estimates of construction and fuel cycle costs and should identify the sources of funds to cover these costs.

The applicant can obtain estimates of construction costs from the facility designers as part of the design contract, from construction bids received from contractors to build the facility, or from costs, adjusted for inflation, for similar completed projects. However, the number of non-power reactors constructed in recent years is limited. Construction costs to install a non-power reactor in an existing building should take into account the cost of the reactor and support systems and any modifications to the existing building. The regulations in 10 CFR 50.10 allow the construction of multipurpose buildings (e.g., construction of a college laboratory building that will house a non-power reactor) without the issuance of a reactor

construction permit. If the building funding has been committed, the applicant should specify the costs for that section of a multipurpose building used to house the non-power reactor and should indicate that the costs are considered covered costs.

The applicant can obtain fuel cycle cost estimates from analysis of the proposed operations, as well from analysis of proposals from fuel vendors and providers of other services needed for the fuel cycle. The applicant can also quote recent costs of operating similar reactors. The applicant should estimate fuel cycle costs even if the non-power reactor facility receives fuel assistance from the U S. Department of Energy (DOE).

The applicant should discuss the sources of funds to cover these estimated costs. If the source of funding is not committed, the applicant should discuss the probability of acquiring the funds and the potential source(s) of the funds. The applicant should discuss the options available to secure funding that is not committed through the completion of the project. The applicant should provide supporting documentation for funding that is committed. For example, if university funds are to be used to construct the facility, the applicant could provide a statement signed by the chief financial officer of the university. If gifts or grants are to be used, the applicant could submit copies of these documents. This section also applies to fuel cycle costs. If DOE will be supplying fuel for the reactor and support for the fuel cycle, the applicant should submit a letter from DOE stating this fact or a copy of the DOE grant that supports the fuel cycle costs. Section 1.7 of this document contains information on fuel disposal costs.

15.2 Financial Ability To Operate a Non-Power Reactor

An applicant for an operating license or for renewal of an operating license for a non-power reactor should submit information which demonstrates that the applicant possesses, or has reasonable assurance of obtaining, the funds necessary to cover estimated operating costs for the duration of the license. The applicant should provide estimates of operating costs for each of the first 5 years of operation of the facility or for the first 5 years of the renewal period. The applicant should also indicate the sources of funds covering these costs.

The applicant can obtain estimates of operating costs from an analysis of the proposed operation that takes into account the operating time and the experimental program. The applicant can exclude from the analysis those overhead services that are provided to all departments of the university or company without internal transfer of funding (e.g., cleaning, utilities, and in some organizations, health physics coverage), but the applicant should indicate that these costs are excluded and should discuss the reasons for the exclusion. The applicant should include in the costs the overhead that is allocated to departments (e.g., a certain percentage

of direct salaries for benefits or a percentage of the total budget). The applicant for a new facility can use similar operating facilities to develop cost estimates. Applicants seeking to renew licenses have actual costs from which to develop future estimates. The 5-year estimates should be sufficiently detailed to show categories of spending, such as salaries, benefits and overhead, equipment, and supplies. If possible, the applicant should break the estimates down by functional area, such as reactor operations, utilization, health physics, and administration.

The applicant should discuss the sources of funds covering these estimated costs. If the source of the funding is not committed, the applicant should discuss the probability of acquiring the funds and the potential source of the funds. The applicant should also discuss the possibility of operating the facility without this funding. The applicant should provide supporting documentation for funding that is committed. The applicant should submit the latest financial statements of the university or the company as part of the evidence of financial solvency and the ability to fund the facility. If gifts or grants are to be used to fund operations, the applicant should submit copies of these documents if they are available.

A non-power reactor may be considered a commercial reactor, as discussed in 10 CFR 50 22. If more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training, the facility would be considered a commercial non-power reactor. Note that the key is not *where* funding comes from, but *the cost* of owning and operating the facility and *the percentage of the cost* devoted to commercial activities. It is possible for a non-power reactor to be involved in commercial activities that provide a large portion of the budget (e g , 90 percent), but if the cost of conducting the commercial activity is less than 50 percent of the cost of owning and operating the facility, the facility may be licensed as a Class 104 facility. This arrangement allows facilities to use commercial activities to fund research and development A commercial non-power reactor generally would be licensed as a Class 103 facility, in accordance with 10 CFR 50.22, and the licensing process would be similar to that for a power reactor.

The NRC staff determines whether an activity is a commercial activity on a case-by-case basis. The staff has determined that some activities are commercial, such as irradiation of gemstones to enhance color and the irradiation of silicon to make semiconductors; other activities may similarly be classified as commercial

In the application, the applicant should discuss any specific activities it is involved in or plans to be involved in and should specify which are commercial activities and which are not. The applicant should show the percentage of cost devoted to commercial activities.

15.3 Financial Ability To Decommission the Facility

The information in this section applies primarily to new applications for operating licenses. As part of the operating license application, 10 CFR 50.33(k) requires a report indicating how reasonable assurance will be provided that funds will be available to decommission the facility. The information to be submitted to the NRC for decommissioning is discussed in 10 CFR 50.75(d). The decommissioning report shall contain a cost estimate for decommissioning the facility, an indication of which of the method or methods described in 10 CFR 50.75(e) is to be used to provide funds for decommissioning, and how the cost estimate and funding levels will be adjusted periodically over the life of the facility to account for changes in the costs of such items as labor and waste disposal charges.

To identify costs and develop the cost estimate, the applicant may have to carry out preliminary decommissioning planning to identify the amount of radioactive waste created, the labor required to decommission the facility, and the other supplies and costs associated with decommissioning the facility. Chapter 17, "Decommissioning and Possession-Only Amendments," of this document contains additional information on decommissioning planning and decommissioning plans. Section 1.7 of this document discusses disposal of high-level radioactive waste and spent fuel.

Acceptable methods of providing financial assurance for decommissioning of non-power reactors are discussed in 10 CFR 50.75(e)(2). These methods include prepayment, an external sinking fund, and a surety method, insurance, or other guarantee method. Federal, State, or local government applicants may submit a statement of intent containing a cost estimate for decommissioning and indicating that funds for decommissioning will be obtained when necessary. State university officials may use this method for non-power reactors they own. The statement of intent must be signed by an official who has the authority to commit to spending the necessary funds to accomplish decommissioning. It should be clearly asserted in the statement that the official signing the statement has the authority to commit to spending the funds.

The applicant can determine the estimated cost of decommissioning from an analysis of the facility design, as well as from an analysis of estimates and actual costs of decommissioning similar facilities. NUREG/CR-1756 contains information on estimating the costs of decommissioning nuclear non-power reactors.

The applicant should discuss how the cost estimate and funding levels will be adjusted periodically over the life of the facility to account for changes in the costs of such items as labor and waste disposal charges. The applicant can use actual changes in costs such as waste disposal charges or indices of costs such as changes

in labor and energy costs to adjust the cost estimate. This process requires that the original cost estimate of decommissioning be in sufficient detail to allow categories of costs that can be adjusted. Funding levels should also be adjusted. This adjustment may require an increase in the funds needed to decommission the facility and a change to the prepayment, external sinking fund, or surety method, insurance, or other guarantee method. Applicants using a statement of intent should ensure that the officials responsible for the statement of intent are aware that the cost of decommissioning the facility has changed.

15.4 Reference

U.S. Nuclear Regulatory Commission, "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors," NUREG/CR-1756, March 1982; Addendum, July 1983.

16 OTHER LICENSE CONSIDERATIONS

In this chapter of the SAR, the applicant should discuss license considerations that do not belong elsewhere in the SAR. One of these considerations is the way reactor components were used in the past. A recent consideration discussed in this chapter is the medical use of non-power reactors. There may be other topics that should appear in this chapter, but the applicant should determine these on a case-by-case basis.

16.1 Prior Use of Reactor Components

This section applies primarily to applications for license renewal, since the facility has a history of operation. However, new facilities at which components are being used that came from other reactor facilities may have to consider prior use of components. An example could be control rod drives from a shutdown facility that are being installed in a new facility. Fuel provided by the Department of Energy (DOE) for a new facility or for a facility already in operation could come from DOE storage and have a history of prior use that must be considered. Prior use need not have taken place at the reactor for which the applicant is seeking a license, but the applicant should consider how the component was used in the past.

In SAR Chapter 13, "Accident Analyses," the applicant discusses various accident events considered for the facility and the components or systems that prevent or limit the release of radioactive material to the facility confinement or containment and to the environment. The applicant should consider whether prior use of components or systems could significantly degrade their capability to continue to perform their safety functions.

Some components and systems for which prior use should be considered are fuel (fuel cladding), reactivity control systems, and engineered safety features. This list is not all inclusive, and the applicant should review the design of the facility to ensure that all prior use of important components and systems has been considered.

For components or systems that are identified for analysis of prior use, the applicant should discuss the possible mechanisms that could lead to deterioration, along with the potential effect of the mechanism on the component or system. Some prominent deterioration mechanisms are the following:

- radiation
- high temperature or temperature cycling
- corrosion
- erosion
- mechanical damage

The applicant may show by analysis that the deterioration mechanisms are not sufficient to cause damage that would interfere with the performance of safety functions. Components should be demonstrated to be within design-life requirements. For example, manufacturer design specifications, recommendations, and requirements could be used to establish acceptable performance standards for use. Analysis could show that the stress in a component caused by temperature cycling is not sufficient to damage the component. As part of the analysis process, the applicant may consider the performance of similar components or systems in other non-power reactors that have longer operating histories. Component attributes or system performance may be measured to show that deterioration mechanisms have not damaged a component or system to a point that impairs its safety function. Two examples are the ultrasound measurement of pipe thickness or the measurement of control rod drop time. In some cases, periodic measurements may be needed to determine trends in component or system deterioration to ensure acceptable performance. The timing of periodic measurements and component or system performance limits may form the basis of limiting conditions for operation in the technical specifications.

The applicant should describe the regular preventive and corrective maintenance program that provides for replacement or repair of important components or systems as necessary. The description should include a discussion of how components are chosen for maintenance, how maintenance intervals are chosen for some key components, and if the manufacturer's suggestions are heeded in the maintenance program. The success of the program in preventing malfunctions and other failures of equipment should be discussed. If components and systems in the maintenance program have malfunctioned or otherwise failed, the applicant should show that these problems do not indicate that the program has broken down.

16.2 Medical Use of Non-Power Reactors

The authors of the Atomic Energy Act of 1954, as amended (AEA), saw that there might be a use for special nuclear material in medical therapy. Section 104a of the AEA allows licenses to be issued for the use of utilization facilities in medical therapy. Two regulations in 10 CFR Part 50 carry out the provisions of the AEA: 10 CFR 50.21 authorizes the issuance of Class 104a licenses for medical therapy, and 10 CFR 50.41 states that the Commission will allow the greatest amount of therapy possible with the special nuclear material that is available. Also, regulations in 10 CFR Part 35, "Medical Use of Byproduct Material," control the use of byproduct material introduced into the human body, such as medical isotopes, and byproduct sources, such as teletherapy.

The use of neutron beams to treat the cancer glioblastoma multiforme and other brain tumors was identified in the 1950s as a potential therapeutic use for non-power reactors. This treatment is called boron neutron capture therapy (BNCT)

and consists of a pre-irradiation administration of a boron compound to the patient and the concentration of the boron compound in the tumor. The patient is then exposed to a neutron beam from the non-power reactor, causing the boron to fission into lithium and an alpha particle, which are heavy charged particles. In theory, these particles cause secondary ionization that kills the tumor cells.

Although the regulations for medical use of byproduct material have expanded and matured over the years with expanded use of byproduct material in medicine, there was no development in the area of regulations for the use of special nuclear material and non-power reactors in medical therapy. The reason for this lack is the fact that, until the current renewed interest in BNCT, the use of non-power reactors in the United States for medical therapy was limited to a few BNCT trials in the late 1950s and early 1960s that did not seem to hold much promise. There was no need to consider regulating non-power reactors for medical therapy, until Massachusetts Institute of Technology (MIT) researchers proposed almost 30 years later that new patient trials using the MIT reactor be conducted.

Improvements in the drugs that concentrate boron in tumor cells and improvements in beam technology toward the use of epithermal beams have convinced researchers to reconsider BNCT and the medical use of non-power reactors. The NRC staff considered a number of approaches to the regulation of non-power reactors for medical use; these ranged from doing nothing (allowing 10 CFR 50.21 and 10 CFR 50.41 to govern medical use of non-power reactors) to conducting full-scale rulemaking to create a 10 CFR Part 35 equivalent in 10 CFR Part 50. The staff decided to place the requirements for the conduct of human irradiations in both the medical provider's license and the non-power reactor technical specifications.

For regulatory purposes, the NRC staff considers medical therapy at non-power reactors to comprise two components: (1) a medical use licensee authorized to use the neutron beam from the facility to irradiate patients and (2) the non-power reactor that provides the neutron beam. Licensees may also have to meet Food and Drug Administration (FDA) requirements and receive approval from FDA to conduct medical therapy. These requirements are beyond the scope of this document.

The medical use licensee is responsible for the patient, preparation of a treatment plan and written directive (as defined in 10 CFR Part 35), administration of the boron pharmaceuticals and the neutron beam to the patient, supervision of the setup and irradiation of the patient, and control of the byproduct material formed in the patient's body as a result of the treatment. The regulatory requirements for the medical use licensee are not discussed in this chapter other than to state that the licensee's license must list the non-power reactor medical therapy treatment facility as an additional place of use. For further details on this aspect of BNCT,

contact the Medical, Academic, and Commercial Use Safety Branch of the Division of Industrial and Medical Nuclear Safety in the Office of Nuclear Material Safety and Safeguards at NRC.

The non-power reactor component comprises the medical therapy treatment facility, production of the neutron beam, physical characterization of the beam at its interface with the patient, all health physics considerations associated with the beam, radioactive contamination and activation of the medical therapy treatment room and its contents, and adherence to a quality management program for the conduct of human therapy. These are the responsibility of the non-power reactor licensee under the reactor license. The regulatory approach for this component is modeled after the one in 10 CFR Part 35, Subpart I, concerning teletherapy.

Licensees who want to irradiate patients at non-power reactors need to receive a Class 104a license from NRC. The application for this license should describe any modifications to the reactor and should discuss the safety aspects of the modifications, the design and installation of a medical therapy treatment facility, and additions to the technical specifications and the facility license

The need for an application for a construction permit (CP) for an existing facility depends on the amount of modification and construction required at a facility to initiate patient irradiations. In general, if major modifications do not need to be made to the reactor core or structure, a CP is not required . However, the applicant should recognize that failure to address safety issues at the start of a modification could lead to delays or additional modifications at a later time. Medical therapy treatment facilities can be installed, filters can be added to existing beam tubes, and in most cases, additional beam tubes can be installed in the reactor without a CP.

Modifications to the facility to conduct animal studies or tests and experiments to develop beam characteristics do not require a Class 104a license. These changes to the facility or procedures described in the SAR, however, must conform to the requirements of 10 CFR 50.59. The licensee shall seek NRC approval for modifications that require a change to the existing technical specifications. However, there is no assurance that modifications made to the facility before applying for a Class 104a license will be acceptable to NRC for the conduct of patient therapy. Although not a requirement, the NRC staff suggests that final modifications to a facility progressing from a configuration for animal studies and beam development to human therapy be included in the application for the Class 104a license, and that these final modifications be made after the Class 104a license is issued.

If no issues of reactor safety are involved, NRC does not review and approve the details of beam design. Placing filters in a beam to control beam characteristics is

allowed in most existing licenses and is not unique to medical therapy. The filter construction and materials must be in accordance with the license and SAR or changes made under 10 CFR 50.59, or the licensee shall seek NRC approval of the changes. The characteristics of the beam necessary to conduct treatment are decisions that are medical in nature and are the responsibility of the physician and scientists developing the beam.

In an application for a Class 104a license, the applicant should address the following general guidelines:

- A commitment to deliver neutrons to treat patients only pursuant to a written directive from a physician authorized user who is specifically authorized to perform BNCT by an NRC- or Agreement State-issued medical use license.

- A commitment to record events equivalent to "recordable events" in 10 CFR 35.2, report events equivalent to "misadministrations" in 10 CFR 35.2, and establish a written quality management program for use of the neutron beam using criteria similar to criteria specified in 10 CFR Part 35 for teletherapy.

- Methodology to ensure that the neutron flux, fluence, and spectrum delivered to the patient are delivered as requested by the physician authorized user.

- Design aspects of the neutron beam delivery system that are important to patient or user safety to prevent changes to these aspects without a license amendment.

- A reactor operator and physician communication system and a method for terminating the treatment exposures.

- Consideration of the anticipated activities that may alter beam characteristics and may require spot checks before beam use, and the spot checks that will be performed in these situations.

- Interlock systems and safety precautions to prevent personnel from being accidentally exposed to the beam in the medical therapy treatment room and safety precautions to be followed before, during, and after treatment exposures to limit occupational exposure to ionizing radiation. (Information on surveillances should be included, if appropriate.)

The applicant can use a combination of design features and administrative requirements and procedures for the medical therapy treatment facility to meet the

above general guidelines. Some of the design features and administrative requirements and procedures may become the basis for the technical specifications.

Some design features that the applicant should consider for the medical therapy treatment facility are the following:

- Personnel should be able to scram the reactor from the medical therapy treatment facility control area, from within the medical therapy treatment room, and by contacting the reactor control room.

- Personnel should be able to communicate from the medical therapy treatment facility control area to the reactor control room. They should also be able to communicate from the medical therapy treatment control area to inside the medical therapy treatment room.

- The medical therapy treatment room shielding design shall meet the requirements of 10 CFR Part 20 and should meet the guidelines of the facility ALARA program for radiation exposure of persons not undergoing treatment.

- The medical therapy treatment room should be designed so that radiation fields inside the room from activation of the walls, floor, and ceiling and the equipment used for treatment (e.g., treatment table and patient supports) are held to a minimum.

- The medical therapy treatment should be designed with two independent, redundant systems that have the capability to cut off the beam within a short time. These systems may be shutters, each of which has the capability to cut off the beam. Controls for the systems should be located both outside and inside (beam cutoff only) the medical therapy treatment room. If shutters are used, they should be designed with redundant sources of motion (e.g., electrical and pneumatic), or each shutter should have a different method of motion. The systems should be designed so that one of them can be manually operated. They should be designed so that one system will cut off the beam if the capability to operate the other system is lost. For example, for redundant shutters, the system should be designed so that electrical failure or low air pressure will cause the shutters to close if they have dual sources of motion or the operable shutter will close if the source of motion is lost to the other shutter. The systems should be interlocked with the medical therapy treatment room entrance so that the beam cannot be turned on unless the room is configured to prevent entry. If the beam is on and the medical therapy treatment room is entered, the systems should be interlocked so that this action will cause the beam to be

cut off automatically. There should be positive indication of the actuation of the systems at the medical therapy treatment facility control area.

- The medical therapy treatment room should have a shielded door, labyrinth with unshielded door, or some other method to prevent entry during treatment. If the door is opened or the room is entered during treatment, the systems used to cut off the beam should automatically operate. If the room has a door that is motor operated, opening that door manually should also be possible.

- If the medical therapy treatment room does not have a method for directly viewing the patient (such as through a lead glass window), the medical therapy treatment facility should be designed with redundant methods (e.g., multiple television cameras and monitors) for viewing the patient. A method of emergency lighting in case of power failure should be provided.

- The medical therapy treatment room should contain a radiation monitoring system. The primary purpose of this system is to indicate if the systems used to cut off the beam have worked properly by monitoring radiation levels in the medical therapy treatment room. This system should have visual and audible alarms in the medical therapy treatment facility control area and inside the medical therapy treatment room. It should be possible to bypass these alarms during treatment to prevent continuous alarming. The system should have a backup power supply.

- A method should be established for monitoring the beam to determine the neutron dose being administered to the patient.

Other design approaches may also meet the general guidelines. The applicant should describe its design and explain how the design conforms with the general guidelines.

Administrative requirements are needed along with design features to control the conduct of medical therapy. The following are some administrative requirements that the applicant should consider for the conduct of BNCT and for inclusion in the technical specifications:

- An administrative procedure should be considered that requires patients to be referred for treatment by a written directive from a medical use licensee authorized by NRC or an Agreement State to use the non-power reactor providing the irradiation.

- The responsibilities of the non-power reactor licensee and the physician authorized user should be stated. It should be clearly stated that medical

treatment is the responsibility of the physician in charge of the therapy and the medical physicist of the medical use licensee, the medical use licensee is responsible for the treatment plan, and the non-power reactor licensee is responsible for delivering the radiation fluence requested in the written directive and for providing current and accurate beam parameters to the medical use licensee.

- Both the non-power reactor licensee and the medical use licensee shall agree that a patient irradiation can be started. However, either the medical use licensee or the non-power reactor licensee should have the authority to terminate a patient irradiation.

- Procedures should provide for removing the patient from the medical therapy treatment room in case of medical complications or equipment failures.

- If the reactor scram in the medical therapy treatment facility control area is not operable, an alternative method to shut down the reactor by contacting the reactor control room may be used for a limited time to allow repair of the medical therapy treatment control area scram circuits. The applicant should justify this alternative method, if requested for use.

- If the usual method of indicating the status of the systems used to cut off the beam fails, a temporary alternative method of indicating status may be used for a limited time to allow for repairs of the primary status indication. The applicant should justify this alternative method, if requested for use.

- The radiation monitor in the medical therapy treatment room should be calibrated at regular intervals as recommended by the manufacturer or in an accepted national standard. The system should be checked for operability before patient irradiation. Alarm setpoints should be stated and justified. If the normal radiation monitor fails, a temporary alternative method of measuring radiation fields in the medical therapy treatment room may be used for a limited time to allow for repairs of the radiation monitoring system. The applicant should justify this alternative method, if requested for use.

- If one of the redundant methods of viewing the patient fails during irradiation, the irradiation in progress may continue if the medical use licensee and the non-power reactor licensee agree.

- Limits on the maximum amount that the radiation fluence given to the patient can exceed the fluence prescribed in the patient treatment plan should be stated. Definitions should be developed for recordable events

and misadministrations. The applicant should discuss when and how these events will be reported to NRC.

- The scrams, beam cutoff systems interlocks, status indication of beam cutoff systems, radiation monitors, communications equipment, and any other systems important to the safety of the medical therapy treatment process should be subject to periodic surveillance requirements if used for therapy. Any systems undergoing maintenance should be successfully tested for operability before patients are treated.

- A requirement for a calibration check of the beam and a functional check of beam monitors at regular intervals should be stated. Requirements for a calibration check of the beam or a functional check of beam monitors or both in the event of beam maintenance or modification should be discussed. A requirement for calibrating the beam monitors at regular intervals should be stated. A requirement for characterizing the beam at regular intervals should be stated. A calibration check of the beam should verify that beam intensity or neutron spectrum has not changed. Characterization of the beam is the determination of the dose delivered by the various components of the beam at a certain depth in the patient. This characterization can be done using phantoms that represent the portion of the body to be irradiated. The phantoms are constructed of materials that interact with radiation in a way that is similar to the way the human body interacts and allow placement of dosimetry to determine doses. A functional check of the beam monitors should verify that the monitors are responding to radiation within specified accuracy. The beam monitors may be calibrated against instruments calibrated by a secondary calibration laboratory.

- Other reactor facilities that perform measurements associated with the conduct of medical therapy (e.g., neutron activation analysis for determination of boron concentration in blood or tissue) should have calibration and surveillance requirements.

- A requirement that repair, maintenance, or modification of the medical therapy treatment facility be properly reviewed and conducted by qualified personnel should be stated. It should be clear what equipment (e.g., medical instruments, patient positioning system) is not considered part of the facility for purposes of this requirement.

- Requirements for qualification and training of personnel to operate the controls for the medical therapy treatment facility should be stated. The applicant should discuss minimum instructions that must be available at the medical therapy treatment facility control area. The applicant should discuss the training for making changes in the operation of the medical

therapy treatment facility that could affect the reactivity of the reactor. NRC-licensed reactor operators and senior reactor operators should be trained and requalified in the operation and maintenance of the facility and associated procedures, as appropriate, including aspects of the reactor's operation that could affect the treatment beam. The applicant should also discuss requirements for retaining training records.

- A requirement to adhere to a quality management program for the conduct of human therapy should be stated. The requirements for the program should parallel those of 10 CFR 35.32, "Quality Management Program." The program should meet the following guidelines, which are similar to the objectives stated in 10 CFR 35.32:

  - written directive before treatment is administered

  - verification of the patient's identity by more than one method before treatment

  - verification that the treatment plan and related calculations agree with the written directive

  - verification that the treatment as administered agrees with the written directive

  - assurance that unintended deviations from the written directive are identified and evaluated and that appropriate action is taken in response to any such unintended deviation

- Written procedures required by the technical specifications should be in place for the conduct of medical therapy treatments before starting human irradiations.

The appendix to this chapter is a copy of the technical specifications for human therapy at the Massachusetts Institute of Technology.

Appendix 16.1

Amendment No. 27
to Facility Operating License
No. R-37
Massachusetts Institute of Technology

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 16, 1993

Docket No. 50-20

Dr. John A. Bernard
Director of Reactor Operations
Nuclear Reactor Laboratory
Massachusetts Institute of Technology
138 Albany Street
Cambridge, Massachusetts 02139

Dear Dr. Bernard:

SUBJECT: ISSUANCE OF AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. R-37 - MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR (TAC NO. M82958)

The Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. R-37 for the Massachusetts Institute of Technology Research Reactor (MITR-II). The amendment consists of changes to the Technical Specifications in response to your submittal dated March 10, 1992, as supplemented on July 30, 1992, August 31, 1992, September 23, 1992, December 22, 1992, December 30, 1992, and January 22, 1993.

The amendment adds requirements to the Technical Specifications (TS) for the use of the MITR-II Medical Therapy Facility beam for human therapy.

Please provide us with a final approved copy of your procedures for the conduct of human irradiations with the MIT medical therapy facility no later than thirty days prior to the start of human irradiations.

A copy of the related Safety Evaluation supporting Amendment No. 27 is enclosed.

Sincerely,

Alexander Adams, Jr.

Alexander Adams, Jr.
Senior Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 27
2. Safety Evaluation

cc w/enclosures:
See next page

Massachusetts Institute of Technology

Docket No. 50-20

cc:

City Manager
City Hall
Cambridge, Massachusetts 02139

Assistant Secretary for Policy
Executive Office of Energy Resources
100 Cambridge Street, Room 1500
Boston, Massachusetts 02202

Department of Environmental
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Robert G. Zamenhof, Ph.D.
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New England Medical Center
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C 20555

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

DOCKET NO. 50-20

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. R-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Facility Operating License No. R-37 filed by the Massachusetts Institute of Technology (the licensee), dated March 10, 1992, as supplemented on July 30, 1992, August 31, 1992, September 23, 1992, December 22, 1992, December 30, 1992, and January 22, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4) and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. R-37 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 27, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard F. Dudley, Jr. for

Seymour H. Weiss, Director
Non-Power Reactors and Decommissioning
Project Directorate
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: February 16, 1993

ENCLOSURE TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. R-37

DOCKET NO. 50-20

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
iii	iii
	6-21
	to
	6-33
7-32	7-32

Page

6. EXPERIMENTS

6.1 General Experiment Criteria 6-1

6.2 In-Core Cryostat 6-8

6.3 Sodium Subassembly for Fast Neutron Spectrum Facility 6-14

6.4 Closed-Loop Control Systems 6-17

6.5 Generation of Medical Therapy Facility Beam for Human Therapy 6-21

7. ADMINISTRATIVE CONTROLS

7.1 Responsibility 7-1

7.2 Reactor Staff Organization 7-4

7.3 Reactor Staff Qualifications 7-6

7.4 Retraining and Replacement Training 7-8

7.5 Review 7-9

7.6 Action to be Taken in the Event of an Abnormal Occurrence 7-18

7.7 Action to be Taken if a Safety Limit is Exceeded 7-19

7.8 Operating Procedures 7-20

7.9 Experiment Approval Procedures 7-22

7.10 Radiation Protection Program 7-24

7.11 Security Program 7-25

7.12 Records Retention 7-26

7.13 Plant Reporting Requirements 7-28

6.5 Generation of Medical Therapy Facility Beam for Human Therapy

Applicability

This specification applies solely to the generation of the medical therapy facility beam for the treatment of human patients. It does not apply to any other use of the medical therapy facility and/or its beam. Surveillances listed in this specification are only required if human therapy is planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of human therapy during the interval in question.

Objective

To provide for the protection of the public health and safety by ensuring that patients are treated in accordance with the treatment plan established by the physician authorized user and that the ALARA principle is observed for all non-therapeutic radiation exposures.

Specification

1. Patients accepted for treatment shall have been referred by written directive from a physician authorized user of NRC Medical Use Licensee No. 20-03857-06 or of any other medical use licensee that has been similarly authorized by NRC to utilize the MIT Research Reactor's Medical Therapy Facility beam for neutron capture therapy for humans.

2. All medical treatments, including irradiations and analyses of the neutron capture agents in the patients, are the responsibility of the physician authorized user in charge of the therapy and the medical physicists from the NRC-licensed medical center. The Massachusetts Institute of Technology is only responsible for providing current and accurate beam characteristic parameters to the medical use licensee and for delivery of the desired radiation fluence as requested in the written directive. Before the start of a therapy, both the certified medical physicist and the Director of the Nuclear Reactor Laboratory, or his designate, must agree that the therapy can be initiated. The physician authorized user is responsible for monitoring the therapy and for directing its termination. However, a radiation therapy can also be terminated at any time if

either the physician authorized user or the NRL Director, or their designates, judge that the therapy should be terminated.

3. It shall be possible to initiate a minor scram of the reactor from a control panel located in the medical therapy facility area. In the event that the medical facility minor scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.

4. Access to the medical therapy facility shall be controlled by means of the shield door located at its entrance.

5. The following features and/or interlocks shall be operable:

   (a) An interlock shall prevent opening of the shutters that control beam delivery unless the medical therapy facility's shield door is closed.

   (b) The shutters that control beam delivery shall be interlocked to close automatically upon opening of the medical therapy facility's shield door.

   (c) The shutters that control beam delivery shall be designed to close automatically upon failure of either electric power or on low air pressure if the shutter is operated pneumatically.

   (d) Shutters that control beam delivery and that are normally pneumatically-operated shall, in addition, be designed for manual closure.

   (e) It shall be possible to close the shutters that control beam delivery from within the medical therapy facility.

6. Each of the shutters that controls beam delivery shall be equipped with a light that indicates the status of the shutter. These lights shall be visible at the medical therapy facility's local control panel. In the event of a status light malfunction, it shall be acceptable to use the affected shutter provided that an alternate means of verifying position is available. Use of this alternate means of shutter position verification is limited to seven consecutive working days.

7. The medical therapy facility shall be equipped with a monitor that provides a visual indication of the radiation level within the facility, that indicates both within the facility and at the local control panel, and that provides an audible alarm both within the facility and at the local control panel.

   (a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.

   (b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any patient irradiation.

   (c) This radiation monitor shall be calibrated quarterly.

   (d) The audible alarm shall be set at or below 50 mR/hr. This monitor and/or its alarm may be disabled once the medical therapy room has been searched and secured, such as is done immediately prior to initiation of patient therapy. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the medical therapy facility's shield door.

   (e) In the event that this monitor is inoperable, personnel entering the medical therapy facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the MIT Research Reactor's radiation protection program and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.

8. An intercom or other means of two-way communication shall be operable both between the medical therapy facility control panel and the reactor control room, and also between the medical therapy facility control panel and the interior of the facility. The latter is for the monitoring of patients.

9. It shall be possible for personnel monitoring a patient to open the medical therapy facility's shield door manually.

10. It shall be possible to observe the patient through both a viewing port and by means of a closed-circuit TV camera. Both methods of patient visualization shall be operable at the outset of any patient irradiation. Should either fail during the irradiation, the treatment may be continued at the discretion of the physician authorized user. Adequate lighting to permit such viewing shall be assured by the provision of emergency lighting.

11. The total radiation fluence delivered by the medical therapy facility beam as measured by on-line beam monitors shall not exceed that prescribed in the patient treatment plan by more than 20%. The treatment is normally delivered in fractions in accordance with standard practice for human therapy. The 20% criterion applies to the sum of the radiation fluences associated with all fractions in a given treatment plan. A criterion of 30% applies to the difference between the administered and prescribed fluence for any given week (seven consecutive days). Finally, if the treatment consists of three or fewer fractions, then a criterion of 10% shall apply.

12. The following interlocks or channels shall be tested at least monthly and prior to treatment of human patients if the interlock or channel has been repaired or deenergized:

	Interlock or Channel	Surveillance
a)	Medical therapy facility minor scram	Scram test
b)	Shutters will not open unless shield door is closed	Operational test
c)	Shutters close upon both manual and automatic opening of shield door	Operational test
d)	Shutters close on loss of electrical power and reduction of pressure in pneumatic operators, if applicable	Operational test
e)	Manual closure of pneumatic shutters	Operational test

f)	Shutters can be closed manually from within the facility	Operational test
g)	Shutter status lights	Operational test
h)	Radiation monitor alarm	Operational test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Operational test
j)	Intercoms	Operational test

In addition to the above, the medical therapy facility minor scram shall be tested prior to reactor startup if the reactor has been shut down for more than sixteen hours.

13. Manual operation of the medical therapy facility's shield door in which the door is opened fully shall be verified semi-annually.

14. Use of the medical therapy facility beam shall be subject to the following:

a) A calibration check of the beam and a functional check of the beam monitors that are described in provision 11 of this specification shall be made weekly for any week that the beam will be used for human therapy. These checks shall be made prior to any patient irradiation for a given week. In addition, a calibration check shall be performed prior to any patient irradiation in the event that any component of a given beam design has been replaced. Finally, a calibration and a functional check shall be performed prior to any patient irradiation in the event of a design modification.

b) A characterization of the beam shall be performed every six months for any six-month interval that the beam will be used for human therapy. This six-month characterization shall be made prior to any patient irradiation for a given six-month interval. A characterization shall also be performed prior to any patient irradiation in the event of a design modification. As part of the characterization process, the proper response of the beam monitors that are described in provision 11 of this specification shall be verified.

c) A calibration of the beam monitors that are described in provision 11 of this specification shall be performed at least once every two years for any two-year interval that the beam will be used for human therapy. The two-year calibration shall be made prior to any patient irradiation during any given two-year interval.

15. Maintenance, repair, and modification of the medical therapy facility shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the MIT Research Reactor. The 'medical therapy facility' includes the beam, beam shutters, beam monitoring equipment, medical therapy facility shielding, shield door, and patient viewing equipment. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59. The operating couch, patient positioning equipment, medical instruments, and other equipment used for the direct medical support of the patient are not considered part of the medical therapy facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

16. Personnel who are not licensed to operate the MIT Research Reactor but who are responsible for either the medical therapy or the beam's design including construction and/or modification may operate the controls for the medical therapy facility beam provided that:

(a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

(b) Instructions are posted at the medical therapy facility's local control panel that specify the procedure to be followed:

(i) to ensure that only the patient is in the treatment room before turning the primary beam of radiation on to begin a treatment;

(ii) if the operator is unable to turn the primary beam of radiation off with controls outside the medical therapy facility, or if any other abnormal

condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

(c) In the event that a shutter affects reactivity (e.g., the D_2O shutter), personnel who are not licensed on the MIT Research Reactor but who have been trained under this provision may operate that shutter provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency closures are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the MIT Research Reactor's training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

17. Events defined as 'recordable' under definition 8 of this specification shall be recorded and the record maintained for five years. Events defined as 'misadministrations' under definition 9 of this specification shall be reported to the U.S. Nuclear Regulatory Commission (24 hours verbal, 15 day written report). The 24 hour verbal reports will be made to the Regional Administrator, Region I, or his designate. The 15 day written reports will be sent to the NRC Document Control Desk with a copy to the Regional Administrator, Region I, or his designate.

18. The requirements of the Quality Management Program (QMP) for the Generation of Medical Therapy Facility Beam for Human Therapy at the Massachusetts Institute of Technology Research Reactor shall be observed for any human therapy. (Note: The presence of this commitment to observe the QMP in these specifications does not preclude modifying the QMP as provided in that document. Any such modifications are not considered to be a change to the MITR Technical Specifications.)

Definitions

1. The medical therapy facility is equipped with shutters that are used (i) to control beam delivery and (ii) to adjust the neutron energy spectrum of the beam. The former currently include lead, boral, and light water shutters as described in Reference 6.5-1. The heavy water blister tank, which is also described in Reference 6.5-1, is an example of the latter. It is conceivable that these designations may change should it be found desirable to alter the beam configuration. Accordingly, the phrase "shutters that control beam delivery" refers either to the aforementioned three existing shutters or to any future shutter or group thereof that provides an equivalent or greater reduction in beam intensity. Shutter-effect analyses shall be documented through the standard safety review process including, where appropriate, an SAR revision and submission to NRC under 10 CFR 50.59.

2. The term 'calibration check' refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.

3. The term 'functional check of the beam monitors' shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common reactor neutronic power level.

4. The term 'characterization' refers to the process of obtaining the dose-versus-depth profile in phantoms as described in Reference 6.5-2 or an equivalent process. The dose-versus-depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the body part to be treated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.

5. The term 'calibration of the beam monitors' refers to the process whereby the beam monitors that are described in provision 11 of this specification are calibrated against instruments that measure dose including a tissue-equivalent chamber and a graphite or magnesium wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.

6. The term 'design modification' as applied to the medical therapy facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and (b) to a change that has the potential to increase significantly the amount of activation products in the medical therapy facility when the beam is to be used for the treatment of human patients.

7. The term 'radiation fluence' means the total fluence of neutrons and gamma radiation that is emitted in the medical therapy facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron-sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

8. The term 'recordable event' means the administration of:

   (a) A radiation treatment without a written directive; or

   (b) A radiation treatment where a written directive is required without reporting to the medical use licensee in writing each fluence given within 24 hours of the treatment; or

   (c) A treatment delivery for which the administered radiation fluence for any given fraction is 15% greater than prescribed.

9. The term 'misadministration' means the administration of a radiation therapy:

   (a) Involving the wrong patient, wrong mode of treatment, or wrong treatment site; or

(b) When the treatment delivery is not in accordance with provision 11 of this specification.

10. The term 'written directive' means an order in writing for a specific patient, dated and signed by a physician authorized user prior to the administration of radiation and which specifies the treatment site, the total radiation fluence, radiation fluence per fraction, and overall treatment period.

11. The term 'human therapy' means radiation treatments that are of direct therapeutic benefit to the patient and/or part of investigatory studies that involve humans.

12. The term 'physician authorized user' means a medical physician approved for neutron capture therapy by an NRC-approved medical use licensee.

13. The term 'certified medical physicist' means a medical physicist certified in either radiological physics or therapeutic radiation physics by the American Board of Radiology, or in therapeutic radiation physics by the American Board of Medical Physics and who also has specific training in neutron dosimetry and neutron beam capture therapy.

Basis

The stipulation that patients only be accepted from NRC Medical Use Licensee No. 20-03857-06 or from any other medical use licensee that has been similarly authorized by NRC to utilize the MIT Research Reactor's Medical Therapy Facility beam for human therapy, ensures that medical criteria imposed by NRC on such licensees for the use of the MIT Research Reactor's medical therapy facility beam for human therapy will be fulfilled. The second provision delineates the division of responsibilities between the Massachusetts Institute of Technology and the medical licensee that refers the patient. Also, it establishes administrative authority and protocol for initiating and terminating a radiation therapy.

The requirement that it be possible to initiate a minor scram from a control panel located in the medical therapy facility area assures the attending physician and/or medical physicist of the capability to terminate the treatment immediately should the need arise. The provision that access to the medical therapy facility be limited to a single door ensures that

there will be no inadvertent entries. The various interlocks for the shutters that control beam delivery ensure that exposure levels in the medical therapy facility will be minimal prior to entry by personnel who are attending the patient. The shutter-indication lights serve to notify personnel of the beam's status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the medical therapy facility prior to entry. The purpose of this monitor's audible alarm is to alert personnel to the presence of elevated radiation levels, such as exist when the shutters that control beam delivery are open. This monitor and/or its alarm may be disabled once the medical therapy facility has been searched and secured so that it will (1) not disturb a patient and (2) not distract attending personnel. The monitor and/or its alarm are interlocked with the shield door so that they are made functional upon opening that door, and hence prior to any possible entry to the medical therapy facility. One intercom provides a means for the prompt exchange of information between medical personnel and the reactor operator(s). The second intercom is for monitoring the patient.

The provision for manual operation of the medical therapy facility's shield door ensures access to any patient in the event of a loss of electrical power. The presence of the viewing window and a closed-circuit TV camera provide the attending physician authorized user and/or medical physicist with the opportunity to monitor the patient visually as well as through the use of various instruments. The viewing window will function even during an electric power failure because of the provision for emergency lighting.

The specification that the total radiation fluence for a therapy (i.e., the radiation fluences for the sum of all fractions specified in a given treatment plan) not exceed that prescribed in the patient treatment plan by 20% establishes a trigger limit on the delivered fluence above which NRC has to be notified of a misadministration. The 20% criterion is based on the definition of misadministration (clause 4(iv)) as given in 10 CFR 35.2. The criterion that the difference between the administered and prescribed fluence for any seven consecutive days is set at 30%. This is also in accordance with the definition of misadministration (clause 4(iii)) as given in 10 CFR 35.2. Finally, if a treatment involves

three or fewer fractions, then a more stringent criterion, 10%, applies to the difference between the total radiation fluence for a therapy and that prescribed in the treatment plan (10 CFR 35.2(4ii)). The surveillance requirements for beam calibration checks and characterizations provide a mechanism for ensuring that the medical therapy facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the beam monitors ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence.

The specification on maintenance and repair of the medical therapy facility ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the medical therapy facility beam. (Note: Licensed reactor operators may, of course, operate the medical therapy facility beam.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

The specification on 'recordable events' and 'misadministrations' provides for the documentation and reporting to the U.S. Nuclear Regulatory Commission of improper events regarding the generation and use of the medical therapy facility beam. The requirement that the Quality Management Program (QMP) be observed ensures that radiation treatments provided by the medical therapy facility beam will be administered as directed by the physician authorized user.

References

6.5-1 MITR Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J., "Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor," Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.

(ii) Gaseous Waste (Summarized on a monthly basis)

(a) Radioactivity discharged during the reporting period (in curies) for:

(1) Gases

(2) Particulates, with half lives greater than eight days.

(b) The MPC used and the estimated activity (in curies) discharged during the reporting period, by nuclide, based on representative isotopic analysis.

(iii) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet).

(b) The total activity and type of activity involved (in curies).

(c) The dates of shipment and disposition (if shipped offsite).

i. A summary of the use of the medical therapy facility for human therapy.

(i) Investigative Studies

(a) Nature and status of the studies.

(b) Number of patients involved.

(ii) Human Therapy

(a) Number of patients treated.

(b) Type of cancer treated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 TO

FACILITY OPERATING LICENSE NO. R-37

DOCKET NO. 50-20

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

1.0 INTRODUCTION

By letter dated March 10, 1992, as supplemented on July 30, 1992, August 31, 1992, September 23, 1992, December 22, 1992, December 30, 1992, and January 22, 1993, the Massachusetts Institute of Technology (MIT) requested changes to the Technical Specifications (TS) of Facility Operating License No. R-37 for the MIT Research Reactor (MITR-II). The requested changes would add requirements to the TS concerning the use of the MITR-II Medical Therapy beam (beam) for human therapy.

2.0 BACKGROUND

The Atomic Energy Act of 1954, as amended (the Act) authorizes the issuance of class 104 licenses for medical therapy (Section 104(a) of the Act) and for research and development facilities (Section 104(c) of the Act). All non-power reactors (NPRs) currently licensed by the U. S. Nuclear Regulatory Commission (NRC or the Commission) hold class 104c licenses in accordance with 10 CFR 50.21(c). MIT also holds a 104a license in accordance with 10 CFR 50.21(a). MIT is the only NPR licensee to hold a 104a license. This license was issued in 1958 when the original MIT reactor was licensed by the Atomic Energy Commission.

A potential medical therapy use for NPRs that was identified in the 1950s is the use of neutron beams to treat the cancers glioblastoma multiform and metastasized melanoma. Called Boron Neutron Capture Therapy (BNCT), the treatment consists of a patient ingesting a boron compound that concentrates boron in the tumors. The patient is then exposed to a neutron beam from the NPR which causes the boron to fission into lithium and an alpha particle, which are heavy charged particles. These particles cause secondary ionization that kills the tumor cells.

MIT conducted patient trials during 1960-1962 with no success. Apparently, the boron drugs available at that time did not concentrate enough boron in the tumor and the use of thermal neutron beams did not allow neutrons to reach deep enough into the brain for successful treatment.

Since those early trials, improved boron compounds have been developed that are thought to increase concentrations of boron in the cancer cells. Also, MIT has focused on the use of an epithermal neutron beam that will allow neutrons to reach any part of the brain. These two developments have led researchers to believe that another attempt should be made to treat cancers using BNCT.

In letters to the NRC staff dated April 3, April 16, and June 6, 1991, MIT outlined the current proposed practice and procedures for using neutron beams in patient therapy, the history of the 1960-1962 patient trials, and historical information on the procedures for the medical beam therapy facility at MIT. MIT also expressed the opinion that they satisfy all NRC regulations concerning the use of neutron beams for the treatment of human subjects. The existing regulations authorize the issuance of licenses for human therapy but places no requirements or restrictions on the irradiation of humans with neutron beams from NPRs.

During the thirty years since the MIT patient trials, NRC has issued no other 104a license for NPR medical therapy. Thus, the NRC has not developed regulations for the use of special nuclear material for medical therapy at NPRs. However, regulations for the medical use of byproduct material have been developed in 10 CFR Part 35, "Medical Use of Byproduct Material." Although 10 CFR Part 35 may be used as a guide to regulate BNCT, it is not directly applicable to BNCT and the use of special nuclear material for human therapy.

The use of neutron beams for medical therapy has many similarities to teletherapy. Because of this, the NRC staff decided to use the criteria of 10 CFR Part 35, Subpart I, Teletherapy, as a model for the regulation of neutron beams for the treatment of human patients. The requirements of Part 35 were modified to account for the differences between teletherapy and BNCT. For regulatory purposes, BNCT is divided into two components: beam generation and beam use. MIT is responsible for the regulatory aspects of beam generation and the hospital providing treatment to the patient is responsible for the regulatory aspects of beam use. A February 19, 1992, letter to MIT from NRC outlined the commitments and information that MIT was requested to submit to NRC. Because no 10 CFR Part 50 regulations that parallel 10 CFR Part 35 exist, the commitments made by MIT for the conduct of human therapy would be captured in the facility Technical Specifications. The license amendment request from MIT evaluated by the staff in this safety evaluation report (SER) is in response to the February 19, 1992, NRC letter.

3.0 EVALUATION

The commitments and information requested by NRC of MIT to be included in the request for license amendment are as follows:

(1) A commitment to limit the delivery of neutrons only to human subjects pursuant to a written directive from a physician authorized user who is specifically authorized to perform boron neutron capture therapy by NRC medical use licensee No. 20-03857-06 (New England Medical Center).

(2) A commitment to record events equivalent to "recordable events" in 10 CFR 35.2, report events equivalent to "misadministrations" in 10 CFR 35.2, and establish a written quality management program using the criteria specified in 10 CFR Part 35 for teletherapy (the neutron beam).

(3) The methodology to ensure that the neutron flux, fluence, and spectrum delivered to the patient are as requested by the physician authorized user.

(4) The design aspects of the neutron beam delivery system that are important to patient or user safety such that these aspects cannot be changed without license amendment.

(5) The reactor operator and physician communication system and the method for terminating the treatment exposures.

(6) A list of the anticipated activities that may alter beam characteristics and may require spot-checks before the beam use and the spot-checks that will be performed in these situations.

(7) The interlock systems and safety precautions used to prevent personnel from being accidentally exposed to the beam in the treatment room and the safety precautions to be followed before, during, and after treatment exposures to limit occupational exposure to ionizing radiation. Include information on surveillances, if required.

Each of these areas have been addressed by the licensee in their amendment request.

A new section to the TS, Section 6.5, has been proposed by the licensee to place limits and conditions on the generation of a neutron beam for human therapy. These TS will ensure that patients are treated in accordance with the treatment plan established by the physician authorized user, that the as low as reasonably achievable (ALARA) principle is observed for all non-therapeutic radiation exposures, and that the medical therapy treatment room design features operate properly.

The licensee is not seeking approval of the details of beam design. Except for the initial facility design, and the review of changes to the facility, experiments, and procedures conducted under 10 CFR 50.59 for the beam and submitted to NRC, the NRC staff has not reviewed the specific design of the beam filter system or the gamma and neutron fluxes and energies of the beam. The ability to place filters in the beam to control beam characteristics is allowed in the existing license and is not unique to medical therapy. The filter construction and materials must be in accordance with the license and safety analysis report or changes approved under 10 CFR 50.59. The characteristics of the beam necessary to conduct treatment are decisions that are medical in nature and are the responsibility of the physician authorized user. The NRC staff is also reviewing an amendment to the medical use of byproduct material license for New England Medical Center to regulate beam use.

TS 6.5.1 requires that patients be referred by a written directive of the physician authorized user from a medical use licensee authorized by NRC to utilize the MITR-II. TS 6.5.2 clearly defines that treatment is the responsibility of the physician authorized user in charge of the therapy and that delivery of the requested radiation fluence is the responsibility of MIT. These TS are responsive to item (1) above.

TS 6.5.2 also assigns responsibility to medical and MIT personnel for the initiation, monitoring, and termination of treatment. TS 6.5.10 requires that the patient be observed through both a viewing port in the medical therapy facility and by a closed-circuit TV camera. Therapy may continue at the discretion of the physician authorized user if one of the viewing methods fails during the treatment. These TS are responsive to item (5) above. The staff concludes that patients will be properly referred for treatment and that the responsibility for the treatment will be properly controlled. This is acceptable to the staff.

TS 6.5.3 requires that it be possible to initiate a minor scram of the reactor from the medical therapy room control panel and provides a temporary provision of communicating with the control room operator to scram the reactor if the minor scram button on the medical therapy room control panel is out of service. This temporary measure is limited to seven consecutive working days. TS 6.5.8 requires that an intercom or other means of two-way communication exist between the medical therapy facility control panel and the control room and between the medical therapy facility control panel and the inside of the therapy room. This will allow the reactor, which is the source of neutrons, to be rapidly shut down in an emergency. These TS are responsive to items (5) and (7) above. The staff concludes that therapy can be rapidly terminated if the need arises. This is acceptable to the staff.

Item (7) above focuses on interlocks and safety precautions to prevent accidental personnel exposure and surveillances on these interlocks. TS 6.5.4, 6.5.5, 6.5.6, and 6.5.9 describe the operation and interlocks on the medical therapy facility doors that allow access into the facility and on the shutters that control beam delivery and, when closed, shield the inside of the medical therapy facility room from the reactor. TS 6.5.4 requires that access to the facility be controlled by a shield door. TS 6.5.5 requires that an interlock prevent the shutters from opening unless the facility shield door is closed. If the shutters are open and the shield door is opened, the shutters are interlocked to automatically close. The shutters will also close automatically if electric power fails. If the shutters are pneumatically powered and air pressure becomes low, the shutters will automatically close. The shutters are also designed to be manually closed. TS 6.5.5 also requires that the shutters are able to be closed manually from within the medical therapy facility. TS 6.5.6 requires that the status of the shutters be indicated by lights. A temporary alternate means of indicating shutter position may be used for up to seven consecutive working days if the light system fails. TS 6.5.9 requires that the facility shield door be able to be opened manually.

TS 6.5.12 and 6.5.13 requires surveillance tests of the above interlocks and the medical therapy facility minor reactor scram. When the facility is in use for human therapy, the surveillances are conducted monthly with the exception of the full opening manual operation of the shield door which is semi-annual. These surveillances will also be conducted if repairs have occurred or the system has been deenergized. In addition, if the reactor has been shut down for more than sixteen hours, the medical therapy facility minor scram shall be tested.

The staff has determined that these TS requirements ensure that adequate safety precautions exist to prevent accidental exposure from the medical therapy facility beam and that surveillances exist to test the interlocks and scram. This is acceptable to the staff.

TS 6.5.7 concerns the requirements for radiation safety and monitoring of the medical therapy facility. This TS is also responsive to item (7) above. The TS requires a radiation monitor in the medical therapy facility with visual readout and local alarm. The primary purpose of this monitor is to warn the MIT staff if the shutters are not properly closed. The TS also encompasses requirements for backup power, checks for proper operation and quarterly calibration. The TS places the monitor setpoint at or below 50 mr/hr. This radiation setpoint is above the radiation levels in the room immediately after termination of treatment, but is low enough to provide warning if the shutters are not properly closed. The licensee has the ability, in accordance with the TS, to disable the monitor during therapy. This will prevent alarming of the monitor during therapy sessions which would be a distraction to the patient and the treatment staff. If the monitor is disabled, it shall be interlocked to automatically become functional again upon opening of the shield door. This TS also allows for temporary means of meeting the monitor requirements for up to seven consecutive working days with portable survey meters or audible dosimeters if the monitor is inoperable. Surveillance requirements for quarterly calibration of the radiation monitor are included in the TS.

The staff concludes that the medical therapy facility will be adequately monitored for radiation levels. This is acceptable to the staff.

TS 6.5.11 concerns the accuracy of the radiation fluence delivered by the medical therapy facility beam. This is responsive to item (2) above. The total fluence delivered shall not exceed the prescribed amount by more than 20 percent. There are also criteria for the administered and prescribed fluence not to exceed 30 percent for any given week (seven consecutive days) and a 10 percent criteria if the treatment consists of three or fewer fractions. Recordable events and misadministrations are defined in the definitions section of TS 6.5. TS 6.5.17 has the requirements for maintaining records of recordable events and for reporting misadministrations to the NRC. Recordable events records shall be maintained for five years. Misadministrations shall be verbally reported to NRC within 24 hours and in writing within 15 days. The definitions of recordable events and misadministrations agree with the definitions given in 10 CFR Part 35 for teletherapy. This section of the TS is acceptable to the staff.

TS 6.5.14 has the requirements for calibration checks, characterization, and calibration of the beam monitors for the medical therapy facility beam. The beam is calibration checked and the beam monitors are functional checked weekly when the beam is in use for treatment prior to treatment for that week. The beam will be characterized every six months for any six month interval that the beam will be used for human therapy prior to patient irradiation for the six month interval. If beam design modification occurs, the beam shall be calibration checked, functional checked, and characterized before patients are irradiated. In addition, a calibration check will be performed in the event of a beam component replacement.

The beam shall be calibrated against dose measuring instruments that have been calibrated by a secondary laboratory at least once every two years for any two year interval that the beam will be used for human therapy prior to patient irradiation for the two year interval. Calibration checks, characterization, functional checks, and calibration are defined in the definition section of TS 6.5. These provisions of the TS are responsive to items (3) and (6) above and will ensure that the beam characteristics remain stable over time.

The staff has determined that the calibrations, beam characterizations, functional checks, and calibration checks required by the TS are acceptable.

TS 5.6.15 controls maintenance, repair, and modification of the medical therapy facility. These changes to the facility will be made under the direction of an NRC licensed senior reactor operator. The TS confirms that changes to the facility will be reviewed pursuant to 10 CFR 50.59. The TS also states that certain items such as the operating couch, patient positioning equipment, medical instruments, and other equipment used for the direct medical support of the patient are not considered part of the medical therapy facility for the purposes of maintenance, repair or modification. These are the responsibility of the medical provider. However, MIT is responsible for the radiation safety aspects (activation and/or contamination control) of this equipment.

This TS responds to item (4) above. The staff concludes that maintenance, repair, and modification of the medical therapy facility will be adequately controlled, the NRC will be involved at the appropriate times through compliance with 10 CFR 50.59, and that MIT will be responsible for radiation safety of equipment in the medical therapy facility. The staff finds this TS to be acceptable.

TS 5.6.16 has the requirements for the training of personnel who are not reactor operators but who are responsible for medical therapy, beam design, or construction and/or modification of the beam. Instructions shall be provided to ensure that only the patient is in the therapy facility prior to the initialization of treatment, and for the steps to be taken if abnormal conditions occur or if the medical therapy facility controls fail to turn off the beam. Training records will be maintained for three years or in accordance with the reactor training program, and a list of qualified personnel will be maintained in the reactor control room. There is a

requirement that, except in emergences, manipulating shutters that could affect the reactivity of the reactor must be authorized in advance by the reactor console operator.

The staff has determined that personnel that operate the medical therapy facility will be properly trained in the use of the facility and that except in emergencies, the reactivity of the reactor will only be manipulated with the permission of the control room operator.

TS 5.6.18 requires that the Quality Management Program (QMP) for the Generation of Medical Therapy Facility Beam for Human Therapy at the Massachusetts Institute of Technology Research Reactor (Attachment 1) be observed for human therapy. The program parallels the requirements of 10 CFR 35.32, Quality Management Program. The program meets objectives similar to those of 10 CFR 35.32 for having a written directive prior to administration of treatment, for verification of the patient's identity by more than one method prior to treatment, that the treatment plan and related calculations are in accordance with the written directive, that the treatment is in accordance with the written directive, that unintended deviations from the written directive are identified and evaluated, and that appropriate action is taken.

Although the QMP is a requirement of the TS, the QMP is not part of the TS and may be changed by the licensee without license amendment. The requirements for QMP modifications are given in the QMP.

The licensee will develop written procedures for the conduct of therapy and will provide them to the NRC at least thirty days prior to the start of human irradiations.

The staff has reviewed the QMP and finds that there is assurance that the objectives discussed above will be met by MIT. This section of the TS is acceptable to the staff.

TS 7.13.5 on reporting requirements is amended to add a requirement that a summary of the use of the medical therapy facility for human therapy be included in the annual report of the MITR-II. The staff finds this addition to the annual report to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative

occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded that there is assurance that the use of the MIT medical therapy facility and beam for human therapy will be in accordance with the treatment plan established by the physician authorized user, that non-therapeutic radiation exposures will be ALARA, and that the medical therapy facility will function as designed.

The staff has also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of reactor accidents previously evaluated, or create the possibility of a new or different kind of reactor accident from any reactor accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant reactor hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

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Attachment:
As stated

Date: February 16, 1993

Quality Management Program

for

Generation of MITR-II Medical Therapy Facility Beam

for Human Therapy

ATTACHMENT 1

Quality Management Program: Generation of MITR-II Medical Therapy Facility Beam for Human Therapy

1. Purpose: The objective of this quality management program is to ensure that radiation treatments provided by the MIT Research Reactor's (MITR-II) Medical Therapy Facility beam will be administered as directed by a physician authorized user.

2. Authorized Medical Use Licensees: Use of the MIT Research Reactor's Medical Therapy Facility beam, for the treatment of human subjects, is limited to the physician authorized users authorized under:

   (a) NRC Medical Use Licensee No. 20-03857-06.

   (b) Any other medical use licensee that has been similarly authorized by NRC to utilize the MIT Research Reactor's Medical Therapy Facility beam for human therapy.

3. Program Requirements: The following requirements are established as part of this quality management program:

   (a) A written directive will, except as noted in subparagraph (iv) below, be prepared by a physician authorized user of the NRC-approved medical use licensee prior to the administration of any radiation therapy. This directive shall be written, signed, and dated by the physician authorized user and it shall include the following information:

      (i) Name and other means of identifying the patient.

      (ii) Name of the physician authorized user and certified medical physicist in charge of the therapy.

      (iii) The total radiation fluence to be administered, the radiation fluence per fraction, the treatment site, and the overall treatment period.

      (iv) If, because of the patient's condition, a delay in order to provide a written revision to an existing written directive would jeopardize the patient's health, an oral revision to an existing written directive will be acceptable, provided the oral revision is documented immediately in the patient's record and a revised written directive is signed by a physician authorized user within 48 hours of the oral revision.

      Also, a written revision to an existing written directive may be made for any therapeutic procedure provided that the revision is dated and signed by a physician authorized user prior to the administration of the next fraction.

      If, because of the emergency nature of the patient's condition, a delay in order to provide a written directive would jeopardize the patient's health, an oral directive will be acceptable, provided that the information contained in the oral directive is documented immediately in the patient's record and a written directive is prepared within 24 hours of the oral directive.

      (v) In order to ensure that the Staff of the MIT Research Reactor has the most recent written directive from the medical use licensee and the

correct directive for the patient in question, a copy of that directive shall be hand-delivered to the MITR Staff by the Staff of the medical use licensee who accompany the patient to MIT. This copy shall then be checked against the most recent previous transmission. Any discrepancy shall be resolved by the medical use licensee prior to the initiation of patient irradiation.

(vi) The Director of the MIT Nuclear Reactor Laboratory, or his designate, will date and sign the written directive to verify that current and accurate beam characteristic parameters were provided to the NRC-approved medical use licensee and that the radiation fluence desired in the written directive was delivered. A copy of this signed directive shall be provided to the medical use licensee within twenty-four hours of a treatment.

(b) Prior to each administration of any radiation, the patient's identity will be verified by more than one method as the individual named in the written directive. The MIT Nuclear Reactor Laboratory will use any two or more of the following acceptable methods of identification:

(i) Self-identification by patients who are conscious upon arrival at the MIT Research Reactor. Information provided by the patient shall include any two of the following: name, address, date of birth, or social security number. The information provided by the patient is to be compared to the corresponding information in the patient's record.

(ii) Hospital wrist band identification with the wrist band information to be compared to the corresponding information in the patient's record.

(iii) Visual identification against photographs provided with the written directive.

(iv) Other methods as specified in U.S. Nuclear Regulatory Commission Regulatory Guide 8.33, "Quality Management Program."

(c) The plan of treatment is certified by the certified medical physicist to be in accordance with the written directive. In this regard, the Massachusetts Institute of Technology is responsible for calibrating the output of the beam monitoring instrumentation versus dose in phantom and for providing a central axis dose versus depth profile. This information will then be used by personnel at the NRC-approved medical use licensee to generate a plan of treatment. Conformance of the beam to its design characteristics is confirmed through the measurements specified in MITR Technical Specification #6.5, "Generation of Medical Therapy Facility Beam for Human Therapy." The beam is characterized dosimetrically every six months (provision 14(b)), the beam monitors are calibrated every two years by a secondary calibration laboratory and their proper operation is verified semi-annually (provision 14(c)), and calibration checks are made of the beam at least weekly for any week that the beam will be used for human therapy (provision 14(a)).

(d) Each administration of radiation is in accordance with the written directive subject to the tolerances established in provision 11 of MITR Technical Specification #6.5, "Generation of Medical Therapy Beam for Human Therapy."

(e) Any unintended deviations from the written directive shall be identified and evaluated, and appropriate action taken. Such action shall include informing the medical use licensee of the deviation. These reviews shall be performed monthly for any month in which human therapy was conducted. For each patient case reviewed, it shall be determined whether the administered total fluence, fluence per fraction, treatment site, and overall treatment period were as specified in the written directive. In the event of any deviation from the written directive, the licensee (MIT) shall identify its cause and the action required to prevent recurrence. These actions may include new or revised policies, new or revised procedures, additional training, increased supervisory review of work, or other measures as deemed appropriate. Corrective actions shall be implemented as soon as practicable.

4. Program Implementation: The following practices shall be observed in order to ensure proper implementation of the quality management program:

(a) A review shall be conducted of the quality management program. This review shall include, since the last review, an evaluation of:

(i) A representative sample of patient administrations,

(ii) All recordable events, and

(iii) All misadministrations.

The objective of this review is to verify compliance with all aspects of the quality management program. For purposes of this review, the term 'representative' in statement (i) above is defined as 100% sampling up to twenty patients; a sample of twenty for twenty-one to one hundred patients, and 20% sampling for more than one hundred patients. In order to eliminate any bias in the sample, the patient cases to be reviewed should be selected randomly.

(b) The procedure for conducting the above review is as follows:

(i) The review shall be performed by the Director of the MIT Radiation Protection Program or his designate.

(ii) The review shall be performed annually.

(iii) Patient administrations selected for review shall be audited to determine compliance with each of the requirements listed in paragraph (3) above.

(iv) The review shall be written and any items that require further action shall be so designated. Copies of the review shall be provided to the NRL Director and to the MIT Reactor Safeguards Committee who will evaluate each review and, if required, recommend modifications in this quality management program to meet the requirements of paragraph (3) above. A copy of these reviews will also be provided to each medical use licensee.

(c) Records of each review, including the evaluations and findings of the review, shall be retained in an auditable form for three years.

(d) The licensee (MIT) shall reevaluate the Quality Management Program's policies and procedures after each annual review to determine whether the program is still effective or to identify actions required to make the program more effective.

5. Response to Recordable Event: Within thirty days after the discovery of a recordable event, the event shall be evaluated and a response made that includes:

(a) Assembling the relevant facts, including the cause;

(b) Identifying what, if any, corrective action is required to prevent recurrence; and

(c) Retaining a record, in an auditable form, for three years, of the relevant facts and what corrective action, if any, was taken.

A copy of any recordable event shall be provided to the affected medical use licensee.

6. Records Retention: The following records shall be retained:

(a) Each written directive for three years; and

(b) A record of each administered radiation therapy where a written directive is required in paragraph (3(a)) above, in an auditable form, for three years after the date of administration.

7. Program Modification: Modifications may be made to this quality management program to increase the program's efficiency provided that the program's effectiveness is not decreased. All medical use licensees shall be notified of any modifications and provided with a copy of the revised program. The licensee (MIT) shall furnish the modification to the NRC (Region I) within 30 days after the modification has been made.

8. Report and Surveillance Frequency: Any report or other function that is required to be performed in this Quality Management Program at a specified frequency shall be performed within the specified time interval with:

(a) a maximum allowable extension not to exceed 25% of the specified surveillance interval, unless otherwise stated in this Quality Management Program;

(b) a total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

9. Definitions:

(a) The term 'physician authorized user' means a medical physician approved for neutron capture therapy by an NRC-approved medical use licensee.

(b) The term 'certified medical physicist' means a medical physicist certified in either radiological physics or therapeutic radiation physics by the American Board of Radiology, or in therapeutic radiation physics by the American Board of Medical Physics and who also has specific training in neutron dosimetry and neutron beam capture therapy.

10. Applicability: This Quality Management Program applies solely to the generation of the medical therapy facility beam for the treatment of human subjects. It does not apply to any other use of the medical therapy facility and/or its beam. Reports and surveillances listed in this specification are only required if human therapy was conducted during the referenced interval.

17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

Since the first nuclear reactor was assembled in 1942, many non-power reactors, critical facilities, and special-purpose reactors have been dismantled, decontaminated, and decommissioned. More than 65 non-power reactors regulated by NRC have been dismantled and their licenses terminated. Approximately the same number of non-power reactors owned by the Department of Energy (DOE) or the Department of Defense have been decommissioned. In the absence of detailed regulations governing decommissioning, NRC has developed a systematic approach for licensee and NRC actions to terminate facility licenses. The approach includes issuance of possession-only license amendments and orders authorizing facility dismantlement.

On June 27, 1988, NRC published a notice of rulemaking (53 FR 24018) amending its existing regulations for terminating licenses. These amendments affected the decommissioning of licensed reactor facilities. The amended 10 CFR 50.82 requires that an application for a termination of a license be accompanied, or preceded, by a proposed decommissioning plan (DP). Licensees should be aware that additional changes have been proposed to the decommissioning regulations (see 60 FR 37374). Therefore, the following guidance is offered in the interim to facilitate the review of decommissioning activities during this period.

17.1 Decommissioning

17.1.1 Preliminary Decommissioning Plan

In addition to the DP required by 10 CFR 50.82, 10 CFR 50.75(f) requires each licensee to submit a preliminary DP. The preliminary DP shall be submitted at or about 5 years before the projected end of operation. The plan shall contain an estimate of the cost of decommissioning and an up-to-date assessment of the major technical factors that could affect planning for decommissioning. The factors to be considered in submitting this information include the following:

(1) the decommissioning alternative anticipated to be used, including consideration of the requirements of 10 CFR 50.82(b)(1)

(2) major technical activities necessary to carry out decommissioning safely

(3) the current situation for disposal of high- and low-level radioactive waste

(4) residual radioactivity criteria

(5) other site-specific factors that could affect decommissioning planning and cost

If necessary, the cost estimate should also include plans for adjusting levels of funds assured for decommissioning to demonstrate a reasonable level of assurance that funds will be available when needed to cover the cost of decommissioning. The licensee may use the format shown in NRC draft Regulatory Guide DG-1005 in preparing the preliminary DP. The preliminary DP need only address the five factors listed above and may be substantially less detailed than the final DP. Additional information on the financial aspects of decommissioning can be found in Chapter 15, "Financial Qualifications," of this format and content guide.

17.1.2 Decommissioning Plan

Applications for authorization to begin decommissioning and for NRC to terminate the license shall be filed as required by 10 CFR 50.82. The NRC staff recommends that the fuel be removed from the core or from an operational configuration under the existing facility operating license as soon as possible after reactor operations permanently cease. The fuel should be shipped off site in accordance with DOE, NRC, and Department of Transportation (DOT) regulations.

The schedule for decommissioning is discussed in 10 CFR 50.82(b)(1)(ii), which requires that a non-power reactor be decommissioned without significant delay, except when concern for public health and safety makes delay necessary. The factors to be considered in evaluating an alternative that delays the completion of decommissioning are discussed in 10 CFR 50.82(b)(1)(iii). Among the factors that may warrant a delay in decommissioning are unavailability of waste disposal facilities; other nuclear facilities, such as another non-power reactor, on the same site; and other site-specific factors.

Under some circumstances, the licensee can apply for a possession-only license amendment according to 10 CFR 50.90 after operations have ended and before decommissioning starts. This amendment normally does not extend the term of the facility license. Generally, the amendment should be based on the stated intent of the licensee to develop and submit a DP and an application to dismantle and decommission the facility and terminate the license. The possession-only license amendment grants the licensee authority to possess but not operate the facility. If requested by the licensee and justified by a safety analysis, regulatory relief from the license and technical specification requirements for an operating non-power reactor may be authorized with a possession-only license amendment. Further, this amendment permits the licensee to retain the reactor facility, related radioactive byproduct material, and, in some cases, special nuclear material, pending approval of the DP. If a non-power reactor is subject to annual licensing fees, the granting of a possession-only license amendment also removes the basis for annual fees.

In 10 CFR 51.53(b), NRC requires that each test reactor licensee submit, in addition to the DP, a supplement to its environmental report with the application to decommission. This ER supplement should reflect any information about significant environmental impacts associated with the proposed decommissioning activities.

A DP should show that the facility can be dismantled and decontaminated safely by describing the plans to decontaminate the facility and site to comply with the release criteria. The DP should describe an organized method for removing radioactive components and material, reducing contamination and radiation levels in the facility, and permitting release of the facility; the organization that will decommission the facility and the process this organization will follow; and the radiological status of the facility and the method of performing tasks to prevent undue radiation exposures to the facility staff and the public and the release of radioactivity to the environment. NRC approves a DP by issuing of an order authorizing implementation.

Some information necessary to develop the DP may not be available when the DP is required to be written, or changes in the DP may be necessary after decommissioning begins. Therefore, the DP should provide for accommodation of any necessary changes in the DP and procedures through a process similar to the one set forth in 10 CFR 50.59. Such a process could ensure that changes to the plan that are reviewed and determined to be not safety significant could be made with approval of the decommissioning safety committee (or other responsible licensee official or organization) and then reported to NRC. Safety-significant changes would still require prior review and approval by the NRC staff. Because the DP is put into effect with an order, in the absence of a process such as that described above, the NRC staff would have to review even minor changes that have no safety significance. This could result in the use of licensee and NRC resources with no concomitant safety benefit.

The NRC staff will not terminate the license until the decommissioned facility meets release criteria. After the facility has been decontaminated and the radioactive materials from decommissioning have been disposed of, the licensee shall measure the residual radiation and document the measurements in a final survey report. The licensee shall submit this report to NRC to be evaluated and to support the request that the license be terminated. NRC regional staff or an NRC contractor will conduct an onsite survey to verify the radiation exposure and contamination levels documented in the final survey report. When the requirements for the release criteria have been satisfied, NRC will terminate the license.

In some cases, the information requested in this format and content guide may be the same as or similar to information previously submitted to NRC. Information

contained in previous submittals, statements, or reports may be incorporated by clear and specific references, and only changes need be submitted.

The administrative format of the application and the procedure for submitting the application should conform to the guidance in Chapter 1 of this format and content guide.

17.1.3 Decommissioning Alternatives

According to 10 CFR 50.82, two general approaches to decommissioning are acceptable. Which approach is chosen depends on the type of residual radioactivity present, the design and number of facilities at a site, the availability of a waste disposal site, and other factors. One approach is to start decommissioning activities soon after operations cease and proceed immediately toward terminating the license and releasing the facility. This is the approach the regulations generally require for non-power reactors. The other approach is to perform limited decommissioning activities soon after operations cease to prepare the facility for safe storage or entombment. The facility will be maintained in a safe condition for a time, after which the licensee will either decontaminate the facility or allow the radioactivity to decay to radiation levels that conform with release criteria.

Three decommissioning alternatives are defined in NUREG-0586.

- In decontamination (DECON), the equipment, structures, and portions of a facility and site containing radioactive materials are removed or decontaminated to a level permitting release of the property by NRC shortly after operations cease.

- In safe storage (SAFSTOR), the nuclear facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to a level permitting release of the property by NRC.

- In entombment (ENTOMB), radioactive materials are encased in a structurally long-lived material such as concrete. The entombed structure is appropriately maintained and surveillance is continued until the radioactivity decays to a level permitting release of the property by NRC.

In general, only DECON is acceptable for non-power reactors, although for the cases discussed in 10 CFR 50.82(b)(1)(iii), such as unavailability of waste disposal capacity, SAFSTOR and ENTOMB are possible options. If either the SAFSTOR or the ENTOMB decommissioning method is selected, the preliminary DP should contain (1) the details for preparing the facility for safe storage or for entombment, (2) plans for monitoring and surveillance during the storage period, (3) plans to ensure that funds and other resources will be available for maintaining the facility

and completing decommissioning, including the means of adjusting cost estimates and associated funding levels over the safe storage or entombment period, and (4) a commitment to submit an updated final DP to obtain authorization before final decommissioning activities are started.

17.1.4 Release Criteria and Final Survey

To terminate a license, NRC shall determine that the health and safety of the public will continue to be protected after the facility and site are released. To make such a determination, NRC should have evidence that radiation levels at the facility permit release.

The criteria to release non-power reactor facilities for unrestricted use are as follows:

(1) (a) no more than 5 microrem (μR) per hour above background at 1 meter from the surface measured for indoor gamma radiation fields from concrete, components, and structures, or

(b) no more than 10 millirem (mrem) per year for gamma emitters above background absorbed dose to any person, considering reasonable occupancy and proximity (NRC letters dated March 17, 1981, and April 21, 1982)

(2) residual surface contamination consistent with Regulatory Guide 1.86

Criterion 1b, which considers reasonable occupancy and proximity, may allow release of concrete, components, and structures that have radiation fields above 5 μR per hour. However, the staff will consider this release criterion only if the licensee can justify its use — for example, if the licensee does not want to completely demolish a structure and if meeting the 5-μR-per-hour criterion would require removing concrete from the structure to the point of jeopardizing its integrity. If the area can be secured from occupancy so that criterion 1b can be met, the NRC staff would consider allowing the use of the 10-mrem-per-year criterion.

If there is a question of soil, ground water, or surface water contamination at the site, or if conditions warrant the development of a pathways analysis, the licensee should contact the NRC staff to discuss release limits (see "NRC Policy and Guidance Directive FC 83-23, and 40 CFR Part 141, "National Primary Drinking Water Standards").

The Commission is considering changes to the release criteria for decommissioning (see 59 FR 43200). Licensees should ensure that the release criteria used in their DPs are current.

The decommissioning rule requires the licensee to submit a final radiation survey plan that shall be described in the DP. Plans for the final radiation survey should show with a high degree of assurance (e.g., 95-percent confidence) that residual radioactive contamination levels will meet the release criteria. The procedures, results, and interpretations described in the final radiation survey plan should be verifiable by the NRC staff.

17.1.5 Format and Content of Decommissioning Plan

A proposed format for the DP is shown in Appendix 17.1, which lays out step by step the information required. The format and content are based on two conditions: (1) the DECON alternatives being used for decommissioning (this format and content may also apply to the decontamination and dismantling portion of SAFSTOR) and (2) all fuel has been removed from the reactor and the site. If a particular project does not meet these conditions, the licensee should contact the NRC project manager for additional guidance.

The licensee should number the DP sections to follow the format in Appendix 17.1. Each section gives the key items that should be addressed in any DP to allow timely approval by NRC. Additional guidance for DPs is presented in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.10–1981; International Atomic Energy Agency, 1986; NRC draft Regulatory Guide DG-1005; and NRC "Guidance and Discussion of Requirements for an Application To Terminate a Non-Power Reactor Facility Operating License."

17.2 Possession-Only License Amendment

Possession-only license amendments are issued to remove the authorization to operate the reactor from the facility license. NRC generally issues a possession-only license amendment to a non-power reactor facility after the fuel is permanently removed from the core and shipped off site. No regulations require the issuance of a possession-only license amendment. Licensees of non-power reactors can go directly from an operating license to an approved DP.

Before the new decommissioning regulations were promulgated in 1988 (53 FR 24018), possession-only license amendments were issued for the remaining term of the license. Without a regulatory requirement to complete decommissioning immediately, some licenses with possession-only amendments were renewed at least once. The new decommissioning regulations did not specifically address

possession-only license amendments, but 10 CFR 50.82(a), as amended, requires that an application for authorization to decommission be submitted within 2 years after reactor operations permanently cease. Possession-only license amendments issued by the NRC staff before July 27, 1988, have been allowed to remain in effect for the term of the license. If, because of factors similar to those in 10 CFR 50.82(b)(1)(iii), the licensee can show good cause for delaying the start of decommissioning, the staff will consider renewing these possession-only licenses issued before July 27, 1988.

The purposes of the possession-only license amendment include the following:

- to remove authority from the license to operate the facility
- to continue the licensee's authorization to possess byproduct and special nuclear material (SNM)
- to provide limitations and controls over the radioactive material in order to protect the health and safety of the public and the facility staff
- to continue the licensee's authorization to possess the reactor systems
- to allow time to develop, submit, and approve detailed plans for final dismantlement and decontamination of the facility

At the same time that a possession-only license amendment is requested, or at any time after it is issued, the licensee may request changes to reduce the technical specifications and other license requirements from those applicable to an operating reactor. These changes are not required as part of a possession-only license amendment application. However, the technical specifications, as written, remain in force during the possession-only license amendment period.

17.2.1 An Application for a Possession-Only License Amendment

The application for a possession-only license amendment should be made in sufficient time for NRC review and final action before the projected termination of reactor operation. If the reactor has been shut down unexpectedly because of factors beyond the licensee's control, the application should describe the circumstances. Factors beyond the licensee's control could include, for example, abrupt budgetary constraints or equipment failures.

The application for a possession-only license amendment should contain an explanation of why an amendment is being requested A specific duration for the possession-only license amendment should be requested, and the activities to be accomplished and their schedule while the amendment is in effect should be

discussed. The licensee should commit to a date to submit the applications for authorization to decommission and for license termination. Any factors beyond the licensee's control that affect the schedules should be discussed in detail.

The application for the possession-only license amendment may consist of a request to delete the authority to operate the facility from the license and a safety analysis of the permanently shutdown reactor. The licensee may also discuss proposed license conditions, proposed technical specifications, and proposed changes to physical security and emergency plans and related safety analyses if it chooses to change these aspects of the license. In the safety analysis the licensee should describe and analyze the shutdown facility and the provisions to control reactor-related radioactivity to protect the health and safety of the public.

17.2.1.1 Facility License

Proposed changes to the facility license should be consistent with the possession-only status. The authority to operate the facility should be removed from the facility license, and either the authorized maximum thermal operating power of the facility and the allowable pulse insertion, if applicable, should be reduced to zero or the paragraph concerning power level should be removed from the license. If fuel has been removed from the site, authorization to possess the reactor fuel should be removed from the paragraph authorizing the possession and use of SNM. The authority to possess any other SNM or byproduct material that is specifically authorized by the facility license and that has been removed from the site or transferred to another license of the decommissioning licensee or to another licensee should be removed from the facility license. The expiration date of the license should not be changed.

17.2.1.2 Technical Specifications

The licensee may choose to amend the technical specifications at this time. If so, the proposed technical specifications should be based on conditions analyzed in the possession-only safety analysis. It is desirable to remove the fuel from the site under the operating license, but it is not always possible. In the latter case, the technical specifications may be more complex, reflecting the need to prevent criticality and to ensure the integrity of fuel cladding.

Because the reactor will no longer operate, the safety limits and limiting safety system settings may be removed from the technical specification. The limiting conditions for operation may be modified to reflect the fact that the reactor will not operate again.

Although each situation is unique, technical specification concerning reactor core parameters, reactor control and safety systems, coolant systems, and experiments

can usually be relaxed. If fuel remains on site, techincal specifications concerning reactor control and safety systems and coolant system may remain. Corrosion or degradation of equipment that confines radioactivity should be monitored. To limit corrosion, potentially radioactive liquids should be monitored for acceptable pH and electrical conductivity. The licensee should consider the radiological condition of the facility and protection of the health and safety of the facility staff and the public when modifying technical specifications for the containment or the confinement, ventilation systems, emergency power, and radiation monitoring systems and effluents. The licensee should consider physical barriers or other constraints necessary to limit personnel access to radioactive and contaminated areas and should list required radiation monitoring instrumentation.

The technical specifications should contain a surveillance section that should include surveillance provisions for each limitation on possession. The surveillance specifications should state what is monitored, how, with what instruments, and on what schedule, and what limits are acceptable.

The technical specifications should contain a section for describing the site and facility; it should contain a concise description of the boundaries of the restricted area to which the possession-only license amendment will apply. If fuel remains in the facility, the reactor coolant system, fuel, and fissionable-material storage should be described.

The technical specifications should contain a section describing the facility administration. Sections on reactor operators may be removed from the technical specifications if all fuel is off site. The section on experiment review and approval may be removed (unless some characterization activities will be considered experiments). This section should give the information necessary to ensure continued management of the facility and should describe personnel and programs for specified surveillance and maintenance activities. This section will probably be the least changed of those in the operating license technical specification.

17.2.1.3 Emergency, Physical Security, and Operator Requalification Plans

Amending an operating license to a possession-only license may allow major changes in emergency and physical security plans if the fuel is removed from the site. These plans are not automatically eliminated. As discussed in Sections 6 and 7 of Appendix 17.1, the licensee may either apply for a specific exemption from the requirement to have a physical security and emergency plan or else submit amended plans in accordance with 10 CFR 50.54(p), (q), and (r). These plans should be modified to reflect the facility status and ensure that the health and safety of the public continue to be protected.

If the fuel has been removed from the site, licensed reactor operators are no longer needed. Therefore, the requirements for the operator requalification program are no longer needed and the licensee may ask to eliminate the plan. A licensed senior operator or licensed fuel handler whose requalification is current is required if fuel is to be moved in the facility.

17.2.1.4 Possession-Only License Amendment Safety Analysis

The purpose of the possession-only license amendment safety analysis is to show that the facility can be possessed in a way that protects the health and safety of workers, the public, and the environment. Even though the potential radiological hazards to the workers and the public are expected to be fewer for a shutdown non-power reactor, the licensee should consider them carefully. These hazards should be discussed and analyzed in a possession-only license amendment safety analysis. The extent of detail in the analysis should depend on how much the licensee proposes to modify the license and technical specifications. The safety analysis for a request to remove only permission to operate the facility may be very brief because all other requirements will remain.

When the license and technical specifications are to be further modified, the safety analysis will be more complex. Requirements that are proposed to be removed and relaxed in the license and technical specifications should be described and justified. The safety analysis should also form the basis for specifying controlling devices and administrative procedures in the technical specifications.

Dismantling a reactor is not permitted by a possession-only license amendment. However, to develop the DP, the licensee may do some limited work to characterize the facility. Characterization activities may be described in the possession-only safety analysis and limited by the technical specification. The reactor facility described in the possession-only safety analysis should be similar to the operating reactor facility, and the SAR should be referred to as much as possible. However, some components, instruments, and systems need not remain operable in the facility. The possession-only safety analysis should state which of the systems covered by the SAR will remain operable and which will be modified or deactivated.

The spectrum of accident scenarios for a possession-only safety analysis should be limited to ones credible for a shutdown reactor and for the residual radioactive material that are not described in the SAR for the operating facility. Postulated accidents should not subject the public or the workers to undue radiological exposure and shall not exceed the applicable regulatory requirements (see Chapter 13 of this format and content guide).

Any proposed changes in staff characteristics should be described and justified in the possession-only safety analysis. Other proposed administrative changes should also be described and justified.

17.2.1.5 Changes to Facility Without License Amendment

Because 10 CFR 50.59 applies only to changes under a license authorizing operation, the licensee should propose a method for making operational or procedural changes under the possession-only license amendment. This should be similar to the method discussed in Section 9 of Appendix 17.1).

Appendix 17.1

Format and Content of Decommissioning Plan for Non-Power Reactors

Format and Content of Decommissioning Plan for Non-Power Reactors

1 SUMMARY OF PLAN

1.1 Introduction

In this section of the decommissioning plan (DP), the licensee should state the type of reactor, its geographical location, its licensed power level, the term of licensed operation, the reason for and objectives of the decommissioning, the entity (facility staff or a contractor) writing the DP, and the organization that will implement the DP, including the final radiological survey. A synopsis of the DP should follow.

1.2 Background

In this section, the licensee should describe the reactor facility, including the licensed restricted area, the facility site, and the type, size, and other uses of the facility buildings. This section should also contain background information about the owner-operator and a brief operating history.

1.2.1 Reactor Decommissioning Overview

The licensee should give an overview of the reactor decommissioning process, including the method selected to decommission, the overall radiological status of the facility, the major dismantlement tasks to be accomplished, and schedules, including the estimated dates for starting and completing the decommissioning. This overview should briefly discuss the final radiation survey plan and the predicted collective dose equivalent to complete the decommissioning. Details about these topics should be given in the appropriate sections of the DP.

1.2.2 Estimated Cost

The licensee should present a cost estimate by principal task for the decommissioning. This estimate should be based on conditions at the facility at the time the DP is submitted. The estimate should include the costs of shipping and disposing of waste and of the final radiation survey. In accordance with 10 CFR 50.75, the licensee shall already have submitted a decommissioning report that contains a cost estimate with means of adjusting the cost estimate over the life of the facility. The cost estimate in this section of the DP should contain much greater detail than the earlier estimates.

1.2.3 Availability of Funds

The licensee should show how sufficient funds will be made available to accomplish decommissioning on the proposed schedule. In accordance with 10 CFR 50.75, the licensee shall have already submitted a decommissioning report that proposed a method or methods acceptable to the Commission to provide funds for decommissioning.

1.2.4 Program Quality Assurance

The licensee should describe the organizational structure established to ensure that quality assurance (QA) measures are applied to the planning, dismantlement, radiological surveys, and material shipments. The relationships of the QA function to the dismantlement organization and to facility management should be made clear.

2 DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Alternative

In accordance with 10 CFR 50.82(b)(1), the licensee shall discuss the approach chosen for decommissioning the reactor facility (DECON, SAFSTOR, or ENTOMB) and a description of the activities involved.

The regulations in 10 CFR 50.82(b)(1)(ii) and (b)(1)(iii) generally require that non-power reactor decommissioning activities be completed without significant delay and that the health and safety of the public be protected. If the licensee believes that significant factors such as those specified in the regulations, that are beyond its control prevent DECON, the licensee should contact NRC before developing a DP using another method.

In this section of the DP, the licensee should address the elements of the DECON alternative, summarizing the objectives, the possible future use of the site, and any special factors that relate to the alternative, such as availability of both licensed and unrestricted waste disposal facilities.

2.2 Facility Radiological Status

2.2.1 Facility Operating History

The licensee should give a detailed and quantitative discussion of the operational history of the reactor facility that could affect decommissioning planning and safety

and from which radiological information can be derived. The licensee should discuss both routine events and such accidents as radioactivity spills or releases that significantly contributed to facility radioactivity and contamination levels.

Experience in decommissioning non-power reactors has shown that most radioactivity to be removed has resulted from routine neutron irradiation of components. Locations of systems and components that may cause high levels of radiation and areas of the site that may contain radioactive hot spots should be identified. This information should be obtained from facility records and personnel familiar with the facility. As requested in Section 2.2.2 of this appendix, the licensee should use the operational history along with surveys and sampling to estimate the facility radionuclide status.

2.2.2 Current Radiological Status of the Facility

The licensee should characterize the reactor-related radioactivity at the facility. The information should be based on reactor operating history, calculations, surveys, and sampling. The radiological characterization is very important because it will be used in planning tasks, estimating doses, characterizing the radioactive waste, and estimating its volume. Suggested information includes the following:

- lists of principal radioactive components
- lists of isotopes with gamma-ray energy groups, curie strength, and geometry
- locations of radioactive sources on reactor and plant layout drawings
- discussion of methods and models used to assess the radioactivity
- cleanup or decontamination tasks already completed

The models and parameters used to calculate the source magnitudes should be given for all sources, including neutron activation byproduct sources. Predicted radiation exposure levels from systems, structures, and components should be evaluated at the time the DP is submitted. Complete information may not be available for detailed planning. However, to the extent practicable, the plan should describe the sources of radiation on which radiation protection and control are based. Information in this section should be updated as additional radiation surveys are made. These updates need not be transmitted to NRC unless required by Section 9 of the DP.

2.2.3 Release Criteria

The licensee should state which release criteria will be applied to the decommissioning project. Considering the radiological status of the facility, along with the release criteria, will allow the licensee to plan the decommissioning activities and tasks.

2.3 Decommissioning Tasks

2.3.1 Activities and Tasks

The licensee should describe in detail the activities and tasks necessary for the two phases: (1) preparing the facility for decommissioning and (2) dismantling and decontaminating the facility. Work should be planned with accurate determinations of the locations and types of radioactive material to be removed.

On the basis of information in Section 2.2 of the DP, the licensee should describe the methods, techniques, and equipment necessary to segment or otherwise dismantle components and systems. All tasks associated with decontaminating the facility and packaging materials for shipment should be discussed.

The licensee should describe procedures for accomplishing major activities. Health and safety considerations should be addressed for each task, as appropriate.

2.3.2 Schedule

The licensee should give the sequential schedules for the activities necessary to plan and complete the facility decommissioning, including the final facility radiation survey.

For major activities, the relationship between activities and tasks should be shown. An activity may comprise several specific tasks. Where pertinent, the schedules for accomplishing interrelated activities and tasks should be delineated. Schedules and critical path diagrams should indicate the estimated time and resources required by major activities, including health physics or radiation protection activities, for the proposed dismantlement and decontamination of the facility. The proposed schedule should identify realistic locations for disposing of radioactive waste. The projected date for submitting the final termination radiation survey report to NRC should be given. The licensee is strongly advised to compare initial estimates with schedules documented for decommissioning similar non-power reactors. The licensee should show that the decommissioning can be completed without significant delay. However, the plan should be flexible enough to allow for changes in schedules.

2.4 Decommissioning Organization and Responsibilities

The licensee should describe the organization that will plan and implement the overall decommissioning, especially management positions. The lines of authority and the management roles of the personnel protection manager and the facility safety committee should be shown.

Key positions in the decommissioning organization should be listed and their functions described. The licensee should diagram the lines of authority from upper management to the workers. The qualifications and experience of the director of decommissioning should be addressed. The staff position with the licensee's onsite management authority should be stated and the qualifications, duties, and responsibilities for that position should be described. In addition, the minimum education, training, and experience requirements should be described for positions that are important to safety.

The licensee should discuss who manages the contractors and what jobs contractors do. Contractor qualification requirements should be defined (see Section 2.6 of this appendix). The discussion should show that the licensee continues to be responsible for overall supervision, compliance with applicable regulations, and protection of the health and safety of the public.

The management policy and organizational structure related to ensuring that occupational radiation exposures are as low as is reasonably achievable (ALARA) should be described. The responsibilities and the activities of management and health physics personnel responsible for radiation protection and the ALARA program should be discussed.

The role and composition of the facility safety committee should be described, especially its review and auditing authority. Attention should be given to its experience with radioactive waste and radiological exposure.

2.5 Training Program

Dismantlement and decontamination may be a new experience for the reactor staff; therefore, special training may be required. Also, contractors should receive training on the DP and the site. The training plans should demonstrate that the licensee is aware of the differences between normal operation and decommissioning.

This section should describe the proposed training program. The training should cover the applicable requirements of the DP, principles and techniques of decontamination and dismantlement activities, industrial hygiene, health physics,

and use and maintenance of monitoring and safety equipment. Compliance with 10 CFR Part 19 should be discussed as required to accomplish DP activities.

The qualifications, experience, and responsibilities of persons responsible for training should be discussed. The following should be discussed for all training activities: training status for all personnel (including contractor personnel), training required for new employees, frequency of refresher training, and personnel certifications.

2.6 Contractor Assistance

The licensee may use contractors to perform some or all of the decommissioning activities and tasks. However, the responsibility for health and safety and compliance with the regulations during all aspects of decommissioning rests with the licensee. The licensee should ensure that contractors comply with all applicable license conditions and regulations.

It is important to discuss the administrative control system used by the licensee to ensure adequate health and safety protection. It is also important to discuss how quality assurance will be achieved. For each contractor, the plan should describe the type and scope of work to be accomplished and the relationship of the contracted work to the schedule for the other activities. The level of qualifications and experience required of contractors and their employees should be discussed. The prior history and performance of the contractor on non-power reactor decommissioning projects should be discussed.

2.7 Decontamination and Decommissioning Documents and Guides

The licensee should discuss the health physics and industrial health criteria and standards that will guide the activities described in the DP. Relevant documents include Regulatory Guide 1.86, ANSI/ANS 15.10-1981, and Occupational Safety and Health Administration (OSHA) requirements.

Much information has been published about decommissioning. Some documents are the following: Manion and LaGuardia, 1980; NRC draft Regulatory Guide DG-1006; NRC IE Circular 81-07; NRC Information Notices 83-05 and 85-92; NRC letters dated March 17, 1981, and April 21, 1982; and NUREG/CR-1756 and addendum. The licensee should review the literature on the decommissioning of similar non-power reactors, including DPs that have been submitted to and approved by NRC. The licensee is advised to use all possible information in preparing the DP.

The licensee shall also show compliance with the regulations, including 10 CFR Parts 20, 30, 50, 70, 71, and 73 as they apply to decommissioning and with 49 CFR Parts 170 through 189.

3 PROTECTION OF THE HEALTH AND SAFETY OF RADIATION WORKERS AND THE PUBLIC

3.1 Radiation Protection

3.1.1 Ensuring As Low As Reasonably Achievable Radiation Exposures

The licensee should state its policy for maintaining radiation exposures and releases ALARA during decommissioning and describe how the policy will be implemented throughout the decommissioning process. The DP should describe either a modification of the existing facility ALARA program or the creation of a complete ALARA program specifically for decommissioning.

Management positions responsible for ensuring radiation protection and implementation of the ALARA program during decommissioning should be described. The licensee should show where the manager of the ALARA program and the radiation protection program fit into the overall management structure and should explain how authority to implement the program is derived and applied. Information from previous decommissioning reports and available literature may be used to assist in developing the program.

3.1.2 Health Physics Program

The licensee should describe in detail the health physics program for decommissioning activities. The overall management and the authority, responsibility, and duties of each position should be identified. This section should include a diagram showing the relationship of the health physics manager to the decommissioning program manager and should discuss the authority to stop potentially unsafe practices.

The licensee should describe radiation protection methods for the public, the decommissioning staff, and radiation workers. These methods should include protection from radiation fields, contamination, and airborne radiation hazards created during the decommissioning process. This section should contain information about survey and personnel monitoring equipment, including the criteria for selecting the equipment and requirements for maintaining, storing, and

calibrating it. The policy, frequency, and procedures for conducting radiation surveys, monitoring effluents and personnel, and assessing and documenting radiation exposures should be described in detail.

The radiation protection practices used by the licensee to meet 10 CFR Part 20 requirements and the procedures for implementing the ALARA program should be described in detail. The licensee should clarify whether the existing health physics program will continue during decommissioning or will be modified.

3.1.3 Dose Estimates

For each major task in which radiation is a factor, the licensee should estimate the total cumulative and individual maximum dose equivalents for radiation workers to complete the task. Both external and internal exposures should be estimated. In addition, this estimate should include credible potential exposure pathways and doses to the public. Methods to ensure compliance with 10 CFR Part 20 and the ALARA program should be explained The licensee may gain insight into dose estimates by reviewing final reports from non-power reactors that have been decommissioned.

3.2 Radioactive Waste Management

3.2.1 Fuel Removal

New and irradiated fuel at non-power reactors should be shipped off site before reactor dismantlement or decommissioning begins. Fuel may be removed from the reactor and shipped under the existing operating license. The DP need only give the date of shipment and disposition of the fuel. The licensee should state that fuel removal and shipment are not included in the DP. The Department of Energy (DOE) has agreed to accept all fuel from licensed non-power reactors under the Nuclear Waste Policy Act of 1982. Fuel must be shipped under applicable NRC and Department of Transportation regulations. The DOE will help the licensee plan for and implement fuel shipments.

The regulations do not require that fuel be completely removed before decommissioning is authorized but the presence of fuel on site during decommissioning may constrain planning and work flexibility and limit the amount of work that can be done. For example, in planning the decommissioning, the licensee must analyze any possible accidents that could occur if demolition took place in the same building as the fuel. The emergency plan and security plan must continue to be in effect if the fuel is on site. A licensee considering the inclusion of fuel disposal in the DP should contact the NRC staff

3.2.2 Radioactive Waste Processing

The licensee should describe the systems and procedures used to detect, control, and process radioactive wastes before disposal. These may be gaseous, liquid, or solid radioactive wastes generated during decontamination and other decommissioning activities. The discussion should address radioactivity levels, volumes of radioactive waste, existing treatment and processing systems that will be retained, process control program, disposal schedules, and radioactive waste control systems to comply with the regulatory requirements and the guidelines of the ALARA program.

3.2.3 Radioactive Waste Disposal

The licensee should give the quantities and types of the radioactive materials for disposal during decommissioning based on the radiological status of the facility as discussed in Section 2.2.2 of the DP that result from the decommissioning activities. It should explain where and how all materials will be disposed of and how the materials will be transported to disposal sites. It should cite applicable regulations and license conditions and explain how compliance will be achieved.

The licensee should discuss radiation measurements, survey methods, and the criteria for determining which materials should be disposed of in licensed locations and which materials may be disposed of in unrestricted sites. The licensee should discuss regulations and guidance for determining what is radioactive waste (NRC IE Circular 81-07 and NRC IE Information Notices 83-05 and 85-92). It should also discuss the experience and qualifications of any persons who will handle or control radioactive materials in unrestricted areas, as during tran

3.2.4 General Industrial Safety Program

The licensee should discuss the protection of personnel from potentially hazardous nonradiation exposures and situations resulting from the decommissioning. These hazards might include conditions associated with controlled demolition; open holes; nonradioactive airborne debris; nonradioactive airborne hazards generated by burning or cutting; including displacement of oxygen; and solvents used to remove radioactive contamination. Many of these hazards may be new to the reactor operating organization. The licensee should describe the industrial safety program and show that all applicable OSHA and industrial safety requirements will be met. The authority and responsibility of each nonradiological safety position in the decommissioning organization should be stated. In addition, this section should specify the criteria for selecting equipment and methods to control nonradioactive exposures and hazards. Nonradiological accident prevention and response should be discussed.

3.3 Radiological Accident Analyses

The licensee should discuss potential radiological accidents that could affect the public or occupational health and safety and present its conclusion as to the acceptability of the results of the accident analysis. Accidents that are directly related to decommissioning and that differ from accidents related to normal reactor operation or maintenance should be emphasized. Sufficient detail should be included to clearly define and analyze the consequences of any significant potential accidents. If NRC has approved fuel removal as a part of the DP, radiological accidents related to fuel storage during decommissioning should be discussed.

All potential releases of radioactivity to or from controlled areas should be discussed.

4 PROPOSED FINAL RADIATION SURVEY PLAN

The final radiation survey report is very important because it forms the basis for verifying that the facility, site, and environs meet radioactivity levels that permit NRC release of the property.

In this section of the DP, the licensee should propose a plan for conducting final radiation surveys. The plan should describe how the facility and site will be surveyed to demonstrate compliance with the release criteria. The description should include the following information:

- methods to ensure that sufficient data about pertinent structures, systems, components, equipment, the site, and the environs are included in the survey (including maps, diagrams, and plant layout drawings)
- types, calibrations, and operating conditions of instruments to be used
- methods to obtain and analyze data, including the methodology selected to translate instrument readings or sample analysis results into appropriate units for the report (e.g., dpm/100 cm^2, μCi/g of soil, curies, μR/h)
- comparisons with preoperational radiation survey results and other data on background radiation
- methods for auditing and verifying data
- quantitative error analyses of the results

- basis and methods for making statistical inferences from the data selected to ensure that all significant residual sources of radiation are found and quantified

The discussion of the radiation survey plan should include an outline of the final report which will contain the results of the final survey and cite relevant references. NUREG/CR-2082 contains significant information about final survey monitoring.

The final survey report should be submitted when the licensee requests the NRC confirmatory termination survey. The report should summarize the decommissioning activities conducted at the facility to prepare for license termination. It should show how actual quantities of radioactive material, waste removed from the site, and cumulative and individual exposures compare with the estimates of the DP. The report should also discuss any unexpected results or findings that differ from those predicted by the DP, the resolution of these issues, and any significant insights to assist other licensees that will decommission their facilities one day. The report should present the results of the final radiation and environmental surveys, and their respective analyses, and should make direct comparisons to radiological guidelines and limits for release of the facility. The report should discuss what was done with radioactive material that was not disposed of as waste, such as fuel, sources, and components that were transferred for use or possession under other licenses.

The radioactivity at the time of decommissioning will consist of contamination (spilled, deposited, and adherent) and radioactivity formed within components by neutron irradiation. All sources of radiation should be considered and either removed or reduced sufficiently for release of the facility. The licensee should discuss how compliance with the release criteria was achieved. These criteria are discussed in Section 2.2.3 (above).

If there is a question of soil, ground water, or surface water contamination at the site or if conditions warrant the development of a pathways analysis, the licensee should contact the NRC staff to discuss release limits.

Currently, no radioactive material, as determined by acceptable industry standards, may be disposed of at unlicensed locations, such as public landfills, without prior NRC approval (see NRC IE Circular 81-07 and NRC Information Notice 83-05 and 85-92).

5 TECHNICAL SPECIFICATIONS

After the nuclear fuel is removed from the reactor and shipped off site, most of the technical specifications for the operating license will not apply if the license has been amended to possession-only or modified by an order to decommission.

During the decommissioning phase, technical specifications should be derived from an analysis of the radiological health and safety of personnel and from an environmental assessment of the decommissioning of the facility. The format of the technical specification should follow the appropriate sections of ANSI/ANS 15.1 as discussed in Chapter 14, "Technical Specifications," of this format and content guide. They should include required equipment, radiation protection, and administrative requirements such as those pertaining to organization, review and audit, procedures, reporting, and records. The analysis should lead to the conclusion that projected radiation exposures of decommissioning personnel and the public are ALARA and are small fractions of the respective regulatory limits (10 CFR Part 20). The decommissioning technical specifications and procedural controls should contain the safety precautions necessary during the decommissioning phases. The technical specifications should contain the following:

- a section imposing limiting decommissioning conditions at the facility comparable to the limiting conditions for operation for required equipment and operational conditions

- a section providing for surveillance of the required equipment and conditions for decommissioning

- a section describing the residual defueled facility and site to which the decommissioning order will apply

- an administrative section that outlines the management structure, provides for review and audit functions, provides for development and use of necessary procedures, and contains reporting and record-retention requirements.

These controls should allow margins sufficient for onsite reaction to changing or unanticipated events.

6 PHYSICAL SECURITY PLAN

This section should contain a description and schedule of proposed changes to the NRC-approved physical security and material control and accountability plans, when applicable Information in this section that must be protected from public disclosure under 10 CFR 73 21 or 10 CFR 2.790 shall be submitted under separate cover and identified appropriately.

If the fuel is shipped off site before or at the time decommissioning activities are authorized, the licensee may request relief from the physical security plan under 10 CFR 50.54(p). The requirements for this plan depend on the amount and

enrichment of special nuclear material (SNM) on site. Removal of fuel may allow the level of site security required by the regulations to be reduced. The licensee will be required to take measures to control public and worker access to radiation areas. Such measures could be included in the revised technical specifications.

7 EMERGENCY PLAN

The regulations require a facility to have an emergency plan as long as it has a license. However, if the fuel, which is the major source of radioactive material on site, is removed, accidents that involve enough radioactive material to reach the emergency action levels spelled out in the plan may be impossible. Fuel removal may allow a major reduction in the level of emergency preparedness required at the facility. Under 10 CFR 50.54(q), the licensee, without prior NRC approval, may make changes to the emergency plan that do not reduce its effectiveness. Changes that reduce the effectiveness of the plan cannot be made under 10 CFR 50.54(q) and shall be approved by NRC before the changes are put into effect. To completely eliminate the requirement for the emergency plan, the licensee should apply for a specific exemption and address all the factors in 10 CFR 50.12.

8 ENVIRONMENTAL REPORT

In accordance with 10 CFR 51.21, the NRC staff shall prepare an environmental assessment for the proposed decommissioning of a non-power reactor and the termination of the facility license. The assessment should be written in compliance with 10 CFR 51.30 and should be based on information given by the licensee in an environmental report (ER). The licensee may include the ER as a section of the DP or submit it as a separate report with the DP. To the extent possible, the content of the ER should follow 10 CFR 51.45. The ER should also address the following:

- the collective dose equivalent to workers for the entire dismantling and decommissioning project
- potential exposures of the public, including the most exposed person, from radioactive effluents released during the decommissioning activities and radioactive waste shipments
- anticipated potential exposures of the public after license termination

9 CHANGES TO THE DECOMMISSIONING PLAN

Because not all events and discoveries during decommissioning are predictable, the DP may need to be changed. Unless the DP contains a method to make changes

without prior NRC approval, NRC will have to approve all changes to the plan, no matter how small or insignificant with regard to safety. Because the DP takes the place of the facility safety analysis report, the licensee should describe provisions for changing the approved DP and DP procedures and for conducting tests. The provisions for changes should be similar to those described in 10 CFR 50.59 for operating reactors. The DP or decommissioning technical specifications should describe the following:

- the kinds of changes the licensee can make without prior NRC approval [similar to the requirements of 10 CFR 50.59(a)(1)]
- the definition of an unreviewed safety question [similar to the requirements of 10 CFR 50.59(a)(2)]
- the way records of the changes made will be held by the licensee and the contents of the records [similar to the requirements of 10 CFR 50.59(b)(1)]
- a requirement to submit a report of these changes to NRC [similar to the requirements of 50.59(b)(2)]
- the length of time records will be held by the licensee [similar to the requirements of 10 CFR 50.59(b)(3)]
- the procedure for making changes to the DP technical specifications or changes that involve an unreviewed safety question [similar to the requirements of 10 CFR 50.59(c)]

Further information about changes to DPs can be found in license Amendment 85 for the Fort St. Vrain facility (Docket No. 50-267), dated November 23, 1992.

10 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.10, "Decommissioning of Research Reactors," ANS, LaGrange Park, Illinois, 1981.

Code of Federal Regulations, Title 10, "Energy," and Title 49, "Transportation," U.S. Government Printing Office, Washington, D.C., revised periodically.

Manion, W. J., and T. S. LaGuardia, "Decommissioning Handbook," DOE/EV/10128/1, prepared for the U.S. Department of Energy, November 1980.

U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1006, "Records Important for Decommissioning of Nuclear Reactors," September 1989.

U.S. Nuclear Regulatory Commission, IE Circular. 81-07, "Control of Radioactively Contaminated Material," May 14, 1981.

U.S. Nuclear Regulatory Commission, IE Information Notice 83-05, "Obtaining Approval for Disposing of Very Low-Level Radioactive Waste," February 24, 1983.

U.S. Nuclear Regulatory Commission, IE Information Notice 85-92, "Survey of Wastes Before Disposal From Nuclear Reactor Facilities," December 2, 1985.

U.S. Nuclear Regulatory Commission, letter from John F. Stolz to Dr. Roland A. Finston, Stanford University, "Use of the 10 mrem per Year Release Criteria," March 17, 1981.

U.S. Nuclear Regulatory Commission, letter from James R. Miller to Dr. Roland A. Finston, Stanford University, "Use of the 10 mrem per year Release Criteria," April 21, 1982.

U.S. Nuclear Regulatory Commission, NUREG/CR-1756, "Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors," March 1982, and Addendum 1, July 1983.

U.S. Nuclear Regulatory Commission, NUREG/CR-2082 "Monitoring for Compliance With Decommissioning Termination Survey Criteria," July 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors." 1974.

18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

The licensee should use the guidance in this chapter of the SAR in preparing the safety analyses to support the conversion from highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel in an existing non-power reactor.

In 1986, NRC promulgated the final rule "Limiting the Use of Highly Enriched Uranium in Domestically Licensed Research and Test Reactors" in Section 50.64 of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR 50.64, 51 FR 6514, February 25, 1986). This rule requires that NRC licensees of non-power reactors, who are authorized to possess and use HEU fuel, convert from HEU to LEU fuel if Federal funding is available unless a facility is specifically exempted because of a unique purpose as defined in 10 CFR 50.2. Each affected licensee is required to submit a conversion proposal to NRC for implementing the regulation.

The conversion proposal and the supporting safety analyses should consider all changes to the facility, license, and procedures necessary to convert from HEU to LEU fuel. NRC will determine if the proposed changes and supporting safety analyses ensure that the public health and safety will be protected. If NRC finds that the proposal acceptably implements the rule, it will issue an enforcement order directing the conversion and any necessary changes in the license, facility, and procedures. In most cases, these orders will modify the license under 10 CFR 2.202.

NRC has selected to use enforcement orders to require the conversion from HEU to LEU fuel, rather than to amend the operating license. This process was selected so that NRC will be responsible for defending the conversion order. This process ensures that any intervention will be made against the NRC order and not against the licensee's proposal for conversion. (Intervention allows a person or persons whose interest may be affected by the order to request a hearing or to participate as a party in hearings.)

The licensee should limit its conversion proposal to changes required to convert from HEU to LEU fuel because the regulations and the associated NRC order are solely for that purpose. The licensee should show that the proposed changes are required to make a safe and effective fuel conversion. If other changes are desired, they should be proposed separately in accordance with the appropriate regulatory requirements (e.g., 10 CFR 50.90).

Factors beyond the control of the licensee in the conversion process may require modifications to the facility or its operating characteristics. For example, the NRC, the U.S. Department of Energy, and the non-power reactor community agreed that a standardized LEU fuel plate would be designed for plate-type

reactors. The characteristics of the approved LEU fuel plate could be sufficiently different from the HEU fuel plate in use at the facility to require changes to the reactor. The licensee should note the physical reason for the required changes and should prepare a safety analysis of the effects of the changes.

The Commission has determined that the conversion to LEU should not result in either a significant reduction in the nuclear capabilities of the reactor or in an increase in capability with a significant decrease in safety margin. Therefore, conversion to LEU fuel should allow non-power reactors to increase or generally maintain the nuclear capabilities similar to the current HEU cores if there is no significant change in the safety analysis results. That is, the safety analyses should demonstrate that acceptance criteria to ensure public health and safety will be met and that the change is needed to accommodate the design differences for the LEU core and fuel.

The information provided in the conversion proposal and associated safety analyses should reflect the current state of the technology, applicable regulatory guides, and industry codes and standards, as well as developments in nuclear safety and experience gained in the construction and operation of non-power reactors. The proposal and safety analyses must account for changes in applicable NRC regulations which were promulgated after the facility was licensed. The design information should reflect the most advanced state of design at the time.

Conformance with this standard format is not required, but the content of the proposal and associated safety analyses should be sufficient for NRC to arrive at all necessary conclusions to approve the proposal and issue the conversion order. NRC will compare the content of the proposal and associated safety analyses against this document and will review it using Chapter 18 of NUREG-1537, Part 2. If the proposal and associated safety analyses are not comprehensive, the staff will request additional information before continuing its review.

Because of the variety of non-power reactors with differing operating characteristics, this chapter attempts to include all possible areas that may be affected by the conversion. Therefore, some of the content may not be applicable for a particular facility. In such cases, the licensee should briefly explain why some topics are omitted.

Portions of the SAR (e.g., site characteristics) that will not change because of the conversion need not be repeated. However, although not required for the HEU conversion to LEU fuel, the licensee may consider submitting a complete revision of the SAR separate from the conversion proposal to have a single complete reference that describes the current licensing bases for the LEU facility.

The numbering system in Appendix 18.1, "Format and Content for HEU to LEU Conversions at Non-Power Reactors," corresponds to the numbering system of the conversion SAR.

It is organized to be consistent with recent licensee SARs and NRC safety evaluation reports for non-power reactors with changes related to specific HEU to LEU conversions. If there is no change to the content of a section for conversion, the licensee should state that fact and should briefly discuss the reason.

Reference made to a chapter other than this one (Chapter 18) alludes to another chapter in the format and content guide. Reference to a section alludes to a section in the appendix.

Additional guidance may also be found in Part 2 of this document as well as in other chapters of Part 1. Also, the licensee can consult such documents as the International Atomic Energy Agency (IAEA) documents IAEA-TECDOC-223, -324, and -643 for additional guidance.

Appendix 18.1

Format and Content for HEU to LEU Conversions at Non-Power Reactors

Format and Content for HEU to LEU Conversions at Non-Power Reactors

1 GENERAL DESCRIPTION OF THE FACILITY*

1.1 Introduction

In this section the licensee should give a brief overview of the conversion from HEU to LEU fuel. The overview should summarize the physical, nuclear, and operational characteristics of the facility for the LEU fuel and should refer to the changes and associated sections of this appendix needed to convert from HEU fuel. It should also have comparisons and references to the existing reactor described in the HEU SAR.

1.2 Summary and Conclusions of Principal Safety Considerations

All changes to the facility SAR should be briefly discussed. Emphasis should be placed on the changes that significantly affect facility safety. Reference should be made to the section(s) in which these changes are analyzed in detail. The referenced SAR section(s) should explicitly mention and compare the changes to the current HEU SAR.

1.3 Summary of Reactor Facility Changes

A summary description of the changes to the reactor and the site associated with the conversion to LEU fuel should be provided. The types of HEU fuel currently used in the reactor and the proposed LEU fuel should be briefly described. This summary description should refer to subsequent discussions on detailed fuel design and should reference documents that describe the proposed type of LEU fuel. Both the HEU and the LEU fuel should be briefly described and compared. If NRC has previously reviewed and accepted the proposed fuel, reference should be made to the acceptance document (e.g., NUREG-1281, -1282, or -1313). Additionally, this portion of the HEU to LEU conversion SAR should discuss any other proposed changes to the facility and give reasons why these changes are required by the conversion.

*The numbering system in this appendix (Sections 1 through 15 8) corresponds to the numbering system in the conversion SAR

1.4 Summary of Operating License, Technical Specifications, and Procedural Changes

In this section, the licensee should summarize any required changes to the facility, operating license, technical specifications, and operating procedures that are proposed to accompany the conversion and should reference the sections in which they are discussed. The following changes should be considered:

- possession limits on special nuclear material
- required provisions to possess HEU and LEU fuel simultaneously
- technical specification changes to specify such LEU fuel and reactor parameters as
  - dimensions
  - materials
  - uranium loading
  - enrichments
  - core sizes and number of fuel elements
  - reactor physics parameters
  - process variable values
  - safety limits, limiting safety system settings, and limiting conditions for operation.

1.5 Comparison With Similar Facilities Already Converted

If possible, the facility should be compared briefly with similar facilities that have already been converted from HEU to LEU fuel. The information should include the following:

- Similarities and differences between the facilities in design, construction, fuels, uses, and operational characteristics.
- Problems related to conversion that were encountered and resolved at the other facilities.
- Discussion of how problems are addressed in the plans for conversion of the subject reactor, if applicable Reference appropriate sections of the SAR where details are provided.

2 SITE CHARACTERISTICS

Among the site characteristics to be described are geography; demography; nearby transportation, industrial, and military facilities; meteorology; geology; seismology; and hydrology. Generally, there should be no change to these topics with a conversion from HEU to LEU fuel. However, it is possible that new information on site characteristics has become available since the last revision of the SAR. For example, it is possible that demography and nearby transportation, industrial, and military facilities around the reactor facility have changed since the last SAR revision. If these changes are significant to the safety analysis of the reactor or if the licensee wants the SAR to be current, these new factors should be justified and submitted to NRC as an update to the HEU SAR preceding the submittal of the LEU SAR. Changes to the site characteristics caused by the passage of time or the discovery of new information unrelated to the conversion cannot be approved by NRC using the conversion order. After separate review and approval by NRC, these changes can be described in this section of the LEU SAR. The existing site characteristics should be analyzed to determine what changes, if any, have occurred in the safety analysis of the LEU-fueled reactor as compared to the same analyses for the HEU-fueled reactor.

If the changes in site characteristics are not significantly different for the LEU-fueled reactor when compared to the HEU-fueled reactor, the bases for this conclusion should be simply stated.

If the conversion from HEU to LEU changes the effect that nearby industrial or transportation uses have on the LEU facility, such as ensuring that personnel can continue to perform their functions in the event of an incident at the industrial or transportation facilities, the licensee should describe the changes in the facility design and operating criteria to ensure safe facility conditions. Such changes should be listed in this section, and described and analyzed where the design or operating criteria of the specific systems, structures, or components are discussed

Most non-power reactors release small quantities of radioactive argon-41 to the unrestricted area during normal operations. Any changes in this and other radioactive effluents resulting from the proposed conversion and the impact on the topics of this section should be mentioned. However, comparisons of potential doses to the public considering current population distributions should be made in Section 11, as well as comparisons with regulatory requirements.

The site should be shown to remain suitable for the safe operation of the reactor. Additional general guidance may be found in American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 15.7–1977

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

The changes to principal architectural and engineering design bases for the structures, systems, and components of the facility should be discussed here. Generally, there should be no change to this section as a result of converting a non-power reactor to LEU fuel, and extensive reference can be made to the current HEU SAR.

In this section, the licensee should reference all significant standards and guides applicable to the conversion, including rules and regulations, NRC staff reports (NUREGs), regulatory guides, and the ANSI/ANS standards used to design all proposed modifications to the reactor facility. In this section, the licensee should also describe how the facility is designed to accommodate potential damage from wind, water, and earthquakes, as well as any changes to the design bases for structures, systems, and components that are a result of the HEU to LEU fuel conversion. The discussions should note the HEU SAR design bases and details for the proposed conversion to LEU fuel to offer reasonable assurance that the associated structures, systems, and components will perform their design function. The licensee may be able to verify that the conclusions of the HEU SAR remain valid by comparing key HEU and LEU characteristics.

The applicable design standards, codes, and criteria that may be affected by the conversion to LEU fuel should be described. In the case of structures, for example, building codes and allowable leak rates for the containment or confinement should ensure that any possible potential increase in fission product inventory associated with the LEU fuel element inventory in accident scenarios is acceptably controlled.

If the effective conversion of the reactor from HEU to LEU fuel requires changes to the systems, structures, or components that are designed to limit the effect of environmental or other factors on the facility, the changes should be listed here, and described and analyzed where those systems, structures, or components are discussed. The information should provide the design bases and functional descriptions, concluding that the changes will not significantly increase potential radiological exposures to the facility staff, the environment, or the public.

4 REACTOR DESCRIPTION

4.1 Reactor Facility

A summary description of the changes to the reactor facility for the conversion to LEU fuel should be presented in this section. It should include replacement of the

HEU fuel and necessary changes to the balance of the reactor facility to accommodate the conversion. The design functions of the reactor with LEU fuel should be demonstrated to perform acceptably over the range of potential conditions. The similarities and differences between the currently licensed HEU-fueled reactor and the proposed LEU-fueled reactor should be tabulated.

In addition to fuel enrichment, other parameters directly affected by the conversion may be the density of the uranium, the cladding material and dimensions, the chemical composition and dimensions of the fuel, irradiation behavior, and configuration of fuel elements. Although not generally anticipated, any necessary changes to in-core components, such as the core grid plate and neutron moderator or reflector, should be described. The licensee should discuss whether any significant operational characteristics or procedures will be changed for the proposed LEU-fueled reactor and should provide reference to associated analyses.

4.2 Reactor Core

A detailed discussion of the components and structures in the reactor core should be provided, including a summary description of the core changes for the conversion to LEU fuel. The licensee should submit figures showing the locations and geometries of the components in the two cores and lists comparing important design parameters and operational characteristics.

4.2.1 Fuel Elements

The LEU fuel elements should be compared to the HEU fuel elements. Any changes resulting from the lower enrichment and possible higher uranium concentration in the LEU elements should be included. The licensee should discuss in detail the mechanical design of the fuel element, volume ratios of fuel to moderator and fuel to coolant, uranium burnup, fission product barrier (cladding) and retention capabilities, and thermal capabilities and characteristics of fueled components. Dimensions such as water gap thickness and fuel element spacing also should be given. If applicable, control and dummy elements should be described.

NRC issued safety evaluations for the types of fuel previously found acceptable in conversions of non-power reactors from HEU to LEU fuel (i.e., NUREG-1281, -1282, and -1313). If one of these fuels is used for the conversion, the SAR should show that the characteristics of the proposed fuel in the converted reactor will be consistent with the characteristics reviewed and evaluated in the corresponding staff safety evaluation. Conformance of fuel fabrication with applicable guidance in ANSI/ANS 15 2–1990 should also be confirmed for plate-type fuels. For a fuel design that differs from those previously found acceptable in NUREG-1281,

-1282, and -1283, analyses and bases similar in content to those reviewed and accepted in the NRC documents should be given.

The HEU and the proposed LEU thermal capabilities and characteristics of fueled components should be compared, and power density and peaking and all allowed core configurations should be considered. These comparative analyses also should discuss such factors as changes in neutron flux densities or shifts in neutron flux spectra.

Fission-product inventory, barriers, and retention for the fuel should be analyzed. These characteristics may differ significantly between the current HEU and the proposed LEU fuel designs. For example, the licensee should demonstrate that the fuel design will acceptably accommodate internal pressures from fission product gases associated with the burnup of LEU fuel.

Safety limits and their bases for the HEU and LEU fuels should be compared, and the LEU values should be shown to be applicable to the fuel design. These parameters should be correlated to the dynamic design and the accident analyses.

Analyses for this section may simply involve reference to the appropriate section of the SAR and report (i.e., NUREG-1281, -1282, and -1313) for the accepted fuel design

4.2.2 Control Rods

Any changes, although not generally anticipated, to the characteristics of the control rods (e g., the mechanical design, material, or configuration) for the proposed LEU core should be described. An example of change in the control rod design would be a change in neutron absorber material or distribution within the control rod, or location in the core Such a change may be due to changes in the neutron spectrum for the LEU core. The changes should ensure that control rod worth for the proposed LEU-fueled core remains within acceptable limits for control or shutdown functions or for potential reactivity change accidents. Another possible change would be the addition of fuel follower control rods to maintain nuclear capability

4.2.3 Neutron Reflector

Any changes in the neutron reflector required by the conversion, including mechanical and functional designs, materials of construction, thermal capabilities, design life, and replacement procedures, should be discussed and analyzed in this section of the SAR. The reflector design may need to be changed to maintain thermal neutron flux at experimental facilities at HEU levels for the LEU cores.

4.2.4 Neutron Source and Holder

Any differences in the neutron source design or location between the HEU and the proposed LEU core should be analyzed to ensure continued safe function. Although not generally expected, changes in fuel geometry for the LEU core may make it necessary to change the source, source holder, or source location.

4.2.5 In-Core Experimental Facilities

Any proposed changes to the in-core experimental facilities required by conversion to the LEU fuel should be described. The design bases, design parameters, and effects on reactivity during allowed operations should be compared for the HEU- and LEU-fueled cores. This comparison should include the thermal-hydraulic characteristics of the in-core experiment facilities and analysis of the thermal-hydraulic effects on nearby fuel elements. Accident scenarios related to in-core experimental facilities should be presented in Section 13 on accident analyses. The use of the experimental facilities should be addressed in the experimental programs portion of the HEU-to-LEU fuel-conversion SAR.

4.2.6 Reactor Materials

In this section of the SAR, the licensee should give the composition of the LEU fuel (e g., cladding and matrix material). Any changes in material and environmental conditions that are needed for the conversion from HEU to LEU fuel also should be discussed. Such changes would include radiation intensity changes as a result of neutron spectrum hardening, erosion, corrosion, and changes as a result of fuel element dimension and associated flow rate changes.

4.3 Reactor Tank and Biological Shield

Any proposed modifications to the tank/pool or the biological shield required by the conversion from HEU to LEU fuel should be analyzed in the SAR. Although not generally expected, these components could change because of modified geometry or neutron flux for the LEU-fueled core. The licensee should show that the change in neutron spectrum should not cause unacceptable irradiation effects on the tank/pool or the biological shield The licensee should demonstrate that modifications to the biological shield should not significantly decrease its physical integrity against cracking or other structural malfunctions for design conditions, including natural phenomena and accident conditions. The licensee should show that the radiation shielding protection provided for the proposed LEU core and reactor should not be significantly decreased below currently acceptable values

4.4 Core Support Structure

The design features, materials of construction, criteria for rigidity and weight support capability to maintain fuel, and other core components in position as designed should be analyzed. The LEU core support structure should be compared with that of the HEU core support structure in such areas as design capability and material composition The licensee should verify that the design of the core support structure remains valid for the LEU-fueled core or should propose changes so that the design function is acceptably maintained. Specifically, the changes in LEU fuel configuration, weight, and neutron spectrum or flux level should be considered to ensure that the current core support structure or proposed changes would perform the intended function.

4.5 Dynamic Design

The licensee should confirm that the integrity of the proposed LEU fuel will not be compromised The control systems and reactor physics parameters should be analyzed and compared with the existing HEU-fueled reactor. Computed HEU characteristics should be compared with HEU core measurements to assist in confirming the validity of calculational methods and codes, and LEU characteristics should be predicted using the same methods and codes These analyses and comparisons should demonstrate that the reactor design, including temperature and power coefficients, will prevent fuel failure from any allowable power, coolant flow, and addition of reactivity conditions (accident conditions should be assessed in the accident analyses, Section 13). The pulsing characteristics of the reactor also should be analyzed if the reactor is authorized to operate in that mode. Additional guidance may be found in selected portions of NUREG-1281, -1282, and -1313, Bullock, 1962, Oak Ridge National Laboratory reports ORNL- 2892 and TM-627; and Shibata, 1984.

4.5.1 Control Rod Worth and Excess Reactivity

The HEU-fueled core should be compared with the proposed LEU-fueled core for control rod worth and excess reactivity. This comparison should include such factors as the effects of xenon-135, samarium-149, and other fission products; void coefficient, temperature coefficients of reactivity for fuel and moderator; burnup and creation of fissile material; movable and secured experiments; and burnable poisons. The neutron spectral hardening associated with conversion to LEU should also be considered. Any changes in control rod worth or excess reactivity should be analyzed, and required changes to associated technical specification limitations should be established and justified Changes should be analyzed to demonstrate that reactor control and function should be acceptably ensured The control rod worth and excess reactivity should be verified in the startup test

program by measurement. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program.

4.5.2 Shutdown Margin

Any changes in the shutdown reactivity for the cold, clean reference core or any other authorized core arrangements and operating conditions that may be limiting with regard to shutdown margin should be analyzed. The safe shutdown of the reactor should be demonstrated, including any reactivity changes caused by expected burnup and by failure of any allowed movable experiment. The shutdown margin required by the technical specifications for the HEU-fueled core should be verified to be applicable for the LEU-fueled core or a change in the associated technical specifications and bases should be proposed and justified. Any change should be demonstrated to continue to ensure acceptable reactor shutdown characteristics, for example, those specified in Chapter 14 as revised from ANSI/ANS 15 1.

The shutdown reactivity should be verified to satisfy shutdown margin requirements during the startup test program. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program

4.5.3 Other Core Physics Parameters

The anticipated inherent reactivity feedback coefficients, including the reactivity coefficients of the fuel temperature, moderator temperature, moderator density and voids, and the power defect should be provided. The HEU-fueled core physics parameters should be compared with the LEU-fueled cores The characteristics and mechanisms of a prompt fuel temperature coefficient of reactivity and its effect on stability and safety of reactor operation, including any applicable pulsing, should be considered Changes in delayed neutron fraction and prompt neutron lifetime resulting from the core conversion should also be considered. These analyses should show that design parameters will compensate for any allowable addition of excess reactivity so that reactor control will be ensured. For example, a rapid reactivity change equivalent to the maximum allowable worth movable experiment will not result in an uncontrollable reactor configuration

The plutonium produced during burnup of the proposed LEU fuel should be analyzed. The significant effects of plutonium-239 on LEU reactor operating characteristics, such as changes in reactivity and delayed neutron fraction, should be included in these analyses. The plutonium production should be compared with existing HEU fuel, and methods to assess plutonium inventory, including end-of-life conditions, should be established and discussed The changes in plutonium inventory for the LEU-fueled core should be considered and a brief description of

the methods to ensure compliance with the requirements of 10 CFR 70 51 and 74.13 for material reporting and status reports should be provided (increased reporting may be required for the LEU fuel under these regulations).

This discussion should compare the HEU and LEU core physics parameters in tabular form. Any related license requirements or proposed changes in technical specifications required by conversion from HEU to LEU fuel should be discussed and justified, and reference should be made to the discussion on license or technical specifications changes.

Core physics parameters that verify proper core configuration and behavior and can be accurately measured, should be verified in the startup test program. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program for comparison with startup test results.

4.5.4 Operating Conditions

The dynamic operating conditions for the LEU fuel designs should be compared to those for the HEU fuel. Core thermal power and neutron flux spatial distributions, both radial and axial, power peaking within individual fuel plates or rods, and related temperature distributions for the range of allowable operating conditions should be analyzed. This analysis should include calculations of both the HEU and LEU cores, giving a quantitative description of the main changes in operating characteristics. The values and characteristics calculated as operating conditions (e.g., maximum temperatures, power peaking factors) should be used or referenced in other areas such as under accident analyses in Section 13.

The dynamic response of the reactor to anticipated and postulated disturbances in the process variables and to changes in reactivity, including inherent feedback mechanisms and protective actions of the reactivity control elements, should be provided. For example, analyses for maximum allowable power, temperature, and reactivity addition conditions should not result in exceeding LEU fuel design temperature limits. Further, any relevant changes in the technical specifications should be discussed and justified here and referenced to the technical specifications discussion.

Operating parameters that verify proper core configuration and behavior and that can be accurately measured should be verified by measurements in the startup test program. A range of expected values with acceptance criteria and bases should be provided in this discussion and referenced to the discussion of the startup test program.

4.6 Functional Design of the Reactivity Control System

Although not expected for LEU conversion, any proposed changes in the design features of the rod-drive systems should be analyzed. Potential changes include reactivity worth and rate of addition or withdrawal for allowable conditions of the LEU-fueled cores. The design of the reactivity control system should be analyzed to verify function for the LEU core under postulated scenarios and conditions of the accident analyses.

Selected parameters, such as those described in Section 12.6, "Reactor Reload and Startup Plan," should be established to verify proper core configuration and behavior through measurement in the startup test program A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program.

The analyses of any necessary changes in the technical specifications that address reactivity control systems should be provided and justified in this section and referenced in the specific discussion on technical specifications changes.

Additional guidance is provided in selected portions of ANSI/ANS 15.15–1978 and ANSI/ANS 15.20 (draft)

4.7 Thermal-Hydraulic Characteristics

Any changes in the thermal-hydraulic parameters and characteristics during routine and transient conditions that may arise from the fuel conversion should be analyzed. Possible changes include the number or dimensions of coolant channels and fuel plates or rods, core dimensions, power density, fuel and cladding temperatures, surface heat flux, radiation heating, thermal conductivity of the fuel, and coolant flow rate. Changes to power peaking factors for the conversion to the LEU reactor core should be included for expected and limiting control rod operating conditions in this analyses This analysis should show that the LEU-fueled reactor thermal-hydraulic design will function under postulated accident scenarios and conditions.

The HEU and LEU thermal-hydraulic characteristics should be calculated. The analytical methods may be confirmed with measurements on the licensed HEU core Any change in the thermal-hydraulic parameters between the HEU core and the proposed LEU core should be discussed. Changes in the thermal-hydraulic characteristics when converting from HEU to LEU fuel should not result in exceeding design limits, such as departure from nucleate boiling ratio (DNBR), flow instability, or fuel safety limits. If there would be significant changes in fuel element flow rates or heat transfer characteristics, verification should be provided that the changes result from factors in the conversion process beyond the control

of the licensee. Any proposed changes to the technical specifications, including safety limits, should ensure that the reactor will operate safely during routine and transient conditions under all analyzed combinations of system parameters.

The initial startup test program should include comparisons, confirmations, and acceptance criteria for thermal-hydraulic characteristics that verify proper core configuration and behavior and that can be accurately measured. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program.

5 REACTOR COOLANT SYSTEMS

Any changes in the reactor coolant system or associated procedures that are required by the conversion to the proposed LEU fuel should be analyzed. This analysis should contain a summary functional description of the individual components and a statement of relevant characteristics (e.g., pressure, temperature, and flow rate). The mechanisms and system characteristics that provide for the automatic transition of forced-flow to natural-flow convection cooling as required to maintain fuel integrity should be analyzed. Changes in the reactor coolant and associated systems needed to convert from HEU to LEU fuel should not result in exceeding any design or safety limit. Changes in this system may affect the reactor, engineered safety features, instrumentation and controls, and accident analyses. For example, changing fuel geometry may result in changes in reactor coolant system flow rates and pressure drops, which may require changes to reactor coolant system configuration or components.

The bases and justification for any changes that may be needed in the related technical specifications should be analyzed in this section and referenced to Section 14 on technical specifications. Selected parameters that verify proper core configuration and behavior and that can be accurately measured should be verified by measurement in the startup test program. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program.

6 ENGINEERED SAFETY FEATURES

Any changes in the engineered safety features (ESFs) required to accommodate new or changed accident progressions brought about by the proposed conversion from HEU fuel should be analyzed. This analysis should consider any accident analyses assumptions and conditions for the ESFs. ESFs may include the reactor room ventilation system, the stack exhaust system, and an emergency core cooling system (ECCS). Any modifications or additions of ESF systems or functions should ensure that design and safety limits continue to be satisfied. Such changes may use guidance from Chapter 6 of the format and content guide.

The bases for any changes that may be needed in the related technical specifications should be analyzed and discussed or referenced to that topic. Selected parameters that verify proper core configuration and behavior and that can be accurately measured should be verified in the startup test program. A range of expected values with acceptance criteria and bases should be provided in this section and referenced to the startup test program.

7 INSTRUMENTATION AND CONTROL SYSTEMS

Although changes are not expected for the conversion to LEU fuel, any changes in instrumentation and control systems, including instrument readings, interlocks, inhibits, and trip setpoints, should be analyzed. This analysis should demonstrate that there is no significant decrease in functional safety capabilities, including speeds of response, redundancy and diversity, or reactor shutdown capability. Additional guidance may be found in ANSI/ANS 15.15–1978 and ANSI/ANS 15.20 (draft).

8 ELECTRICAL POWER SYSTEMS

Although changes are not expected for the conversion to LEU fuel, any changes in the electrical power systems required to implement the core conversion should be analyzed This includes normal power systems and emergency or backup power systems.

9 AUXILIARY SYSTEMS

Any changes in operational characteristics or in components of auxiliary systems required by the HEU to LEU conversion should be discussed in this section. For example, specific consideration should be given to fuel handling and storage systems for conversion to LEU fuel Although not expected, changes to the ventilation and exhaust systems or the heating and air conditioning systems for the reactor room and auxiliary rooms should be explained

Any changes in the LEU fuel configuration or the radioactive source terms from irradiated fuel elements and the effect on fuel storage, handling, and shipping should be analyzed The storage provisions for both fresh and spent fuel elements, their locations, and their capacity should be presented. Criticality safety considerations should be analyzed. For example, maximum k_{eff} under optimum moderator conditions should be determined to be within technical specifications limits and SAR assumptions. If fuel movement and receipt are not regular occurrences, updated procedures may be required Any changes to the fuel handling and storage systems or procedures should be shown to maintain safe conditions, including shielding, subcritical configuration, and cooling.

If the LEU core will be received before the disposal of the HEU core, storage of both cores should be analyzed. The requirements and capability to maintain self-protecting radiation levels of the HEU fuel should be considered.

Additional guidance may be found in ANSI/ANS 15.19–1991.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

Any changes in the experimental irradiation facilities required by the core conversion, such as changes to thermal columns, neutron beam facilities or pool irradiations, as well as changes to in-core experiment tubes and pneumatic transfer systems, that were not analyzed elsewhere in the SAR, should be analyzed. If changes were analyzed elsewhere in the SAR, this discussion should make reference to those analyses. See the discussion in Chapter 10 of the format and content guide for additional guidance on this subject

11 RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT PROGRAMS

Radiation protection and radioactive waste management programs overlap each other on certain issues for non-power reactors. Guidance may also be found in Chapter 11 of this document, Regulatory Guides (RGs) 1.25 and 1.109, NUREG-0851, and ANSI/ANS 15.7–1977 and 15.11–1993. The licensee should also provide its "as low as is reasonably achievable" (ALARA) program and methods for compliance with other applicable regulations, such as 10 CFR Part 71.

11.1 Radiation Protection Program

The following effects on the radiation protection program from modifications to any components or systems, or from changes in operational limits or procedures required by the conversion to LEU fuel should be analyzed:

- potential radiation sources and magnitudes
- radiation detection and measurement methods and procedures
- provisions made to assess potential exposures and doses to the staff and public
- methods used to limit doses to within applicable regulations

- commitment to an ALARA program for potential exposures
- assessment of the effectiveness of the dose control programs
- methods to detect potential inadvertent criticality

11.2 Radioactive Waste Management

The following items should be addressed if affected by the conversion to LEU fuel:

- the capabilities of the reactor facility to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials
- the instrumentation and methods used to measure the quantities of radioactivity and to monitor the release of radioactive wastes outside of the restricted area

All types of radioactive wastes that are generated (solid, liquid, and gaseous) in the operation and maintenance of the reactor facilities and in the experimental programs should be considered. How the conversion of the fuel would change the formation, control, release, or disposition of radioactive waste materials associated with the reactor should be analyzed. Any changes to the facility or procedures should be analyzed and their effects evaluated. Any changes in methods used to store and dispose of radioactive wastes should be described. Any changes in the methods used to assess the source strengths and predict potential radiation exposures to personnel due to radioactive waste in both restricted and unrestricted areas should also be described

For many non-power reactors, argon-41 is a significant radioactive waste that is produced by operations. All changes in the sources of argon-41 and all potential exposure pathways should be addressed. Changes in the sources of argon-41 may be related to changes in fuel geometry and neutron spectra.

The handling or release of all radioactive wastes should be verified to remain consistent with the facility program, applicable regulations, and the facility guidelines with regard to ALARA.

12 CONDUCT OF OPERATIONS

This section provides information on planning and implementing the activities for reactor facility operation during conversion from HEU to LEU fuel.

12.1 Organization and Staff Qualifications

It is not expected that reactor conversion from HEU fuel will require changes in the licensee organization. Note that significant organization changes would have already been reviewed and accepted by NRC in technical specifications and in the license revisions.

12.2 Procedures

Any procedural changes necessary for the conversion from HEU fuel should be described. Procedural changes to account for LEU-fueled core and fuel configuration may include such issues as location of fuel elements and control rods, conduct of critical experiments, storage and shipment of fuel elements as well as procedural changes to account for needed technical specifications changes

The licensee should consider specific changes in shipping requirements and procedures (e.g , type B material, and associated quality assurance program and cask user registration requirements). ANSI/ANS 15.19–1991 provides guidance.

Any new procedures that are needed for the conversion should similarly be addressed. For example, changes to the reactor coolant system to ensure adequate flow may require modified operational procedures

12.3 Operator Training and Requalification

Certain operating characteristics, license conditions, operating procedures, and technical specifications requirements may change as a result of the fuel conversion. The implications of these changes should be incorporated into the operator training and requalification program The operator training and requalification program should clearly establish the procedures, techniques, and on-the-job training which ensures that the licensed operators are instructed on all changes to the reactor, its operating characteristics, written procedures, license conditions, and technical specifications. Additional general guidance can be found in ANSI/ANS 15 4–1988

12.4 Emergency Plan

The conversion to LEU fuel should not significantly affect the emergency action level guidelines or potential reactor emergency event consequences. Although changes are not expected, any changes to the emergency plan that might occur should be submitted as part of the conversion application in accordance with applicable regulatory requirements Additional general guidance may be found in ANSI/ANS 15 16–1982 and RG 2 6

12.5 Physical Security Plan

A less demanding physical security plan could result from the conversion to LEU fuel if the conversion affects the category for possession of special nuclear material as discussed contain 10 CFR 73.2, 73.6, and 73.67. Any proposed changes in the physical security plan and implementing procedures should demonstrate that the applicable requirements of the regulations are still met when either irradiated or unirradiated fuel is at the facility and in transit. These changes should allow possession of LEU and HEU fuel for as long as necessary. Additionally, if the licensee has relied on the exemption of 10 CFR 73.6(b) for self-protection of the HEU fuel, the licensee should demonstrate that the exemption will continue to be applicable while the HEU is still on site or should discuss the options to the self-protecting exemption. Applicable alternatives and changes to the security plan and related, license conditions should be proposed in accordance with regulatory requirements.

12.6 Reactor Reload and Startup Plan

Reloading with LEU fuel and implementing a startup plan are very similar to the initial loading and startup as part of the original licensing action. Many of the same concerns and precautions should be considered in the conversion reload. A reload and startup plan should be submitted as part of the proposal for authorization to convert the reactor from the use of HEU fuel to make sure that the operating characteristics are well understood and to validate the predicted behavior. Therefore, if practical, measurements of selected parameters of the HEU-fueled core should be compared to calculated values to verify analytical methods and ensure that meaningful acceptance criteria for the LEU-fueled core have been established from the calculational methods The plan should provide that the measurements for the LEU reactor are compared to acceptance criteria as established in this conversion proposal. The acceptance criteria should ensure that the LEU reactor is functioning as it was designed and analyzed and that the license and technical specifications will be satisfied

The plan should contain the following items:

- a well-planned systematic set of subcritical multiplication measurements or an inverse multiplication approach to critical measurement during new fuel loading, and confirmation that analysis subcritical multiplication or critical fuel loading are within preestablished acceptable limits

- an experimental measurement plan to determine the important operational reactor physics parameters (such as control rod worth, excess reactivity, reactor thermal power, coefficients of reactivity, and power peaking factors)

and thermal-hydraulic parameters (such as fuel, cladding, and coolant temperatures, reactor coolant system flow rates, and pressure drops, if appropriate), comparisons with predictions and acceptance criteria established in the applicable section of this conversion proposal, and discussions of any discrepancies that may have arisen

- measurements of magnitudes of area radiation fields and radioactive, effluents, and comparisons with the same parameters for operation of the HEU-fueled reactor and preestablished acceptance criteria from the applicable sections of this conversion proposal for the LEU-fueled reactor

A startup report should be submitted to the NRC, in accordance with the time frame required in the technical specifications or within 6 months after the LEU fuel is loaded. An example of an outline for a startup report is given below:

(1) critical mass and final criticality conditions for this first LEU core

- measurement with HEU cores
- measurement with LEU cores
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(2) measurements and comparison to prediction of subcritical multiplication for initial core fuel loading

(3) excess (operational) reactivity

- measurement with HEU cores
- measurement with LEU cores
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(4) control and regulating rod calibration

- measurement of differential and integral rod worth for HEU core
- measurement of differential and integral rod worth for LEU core
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(5) reactor power calibration

- methods and measurements that ensure operation within the license limit
- comparison between HEU and LEU nuclear instrumentation setpoints, detector positions, and detector output
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(6) shutdown margin

- measurement with HEU
- measurement with LEU
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(7) partial fuel element worth

- measurements of the worth for HEU-fueled core
- measurements of the worth for LEU-fueled core
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(8) thermal neutron flux distributions

- measurements of core and experimental facilities with HEU
- measurements of core and experimental facilities with LEU
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(9) enhanced radiation measurements of reactor coolant fission product inventory or release during startup test program to detect potential fuel fission product barrier degradation or contamination from other sources

- measurements of core and experimental facilities with HEU

- measurements of core and experimental facilities with LEU
- comparisons with acceptance criteria established earlier in conversion proposal calculations and analyses

(10) results of determination of LEU effective delayed neutron fraction, temperature coefficients, and void coefficients for LEU- and HEU-fueled cores and comparison with acceptance criteria established earlier in conversion proposal calculations and analyses

(11) comparison of the various results with the two fuel types, and discussion of the comparison, including an explanation of any significant differences that could affect operation and possible accidents with the reactor

(12) thermal-hydraulic characteristics such as fuel, cladding, and coolant temperatures, and reactor coolant system flow rates and pressure drops

13 ACCIDENT ANALYSES

In this section, the licensee should demonstrate that the accident analyses results for the LEU reactor do not significantly exceed acceptance criteria of the accidents previously postulated and analyzed in the safety analyses or evaluated in staff safety evaluation reports In the accident analyses, the licensee also should consider if any new accidents may be introduced by the conversion.

This discussion should summarize the comparison of the HEU and LEU accident analyses. This comparison should demonstrate that the conclusions previously reached in the safety analyses and staff safety evaluation reports remain valid for the LEU-fueled core. For example, a maximum hypothetical accident that assumes a fission product release scenario for an HEU fuel element may be used for the LEU fuel if the licensee demonstrates that the two fuels have sufficiently similar physical characteristics and fission product inventories. If these conditions can be verified, a largely qualitative analysis with check calculations should be sufficient. This process involves a comparison of the scenarios and conclusions for the HEU-fueled reactor accidents considering the physical characteristics of the LEU-fueled reactor

New or revised accident analyses for the LEU fuel should be performed if the techniques used for the current HEU analyses are no longer available or appropriate, or if the conversion creates the possibility of a new, previously unanalyzed accident scenario or significantly changes the potential consequences of a previously accepted HEU accident scenario (If this approach is adopted, the licensee may need to reanalyze some HEU data with the new methods to provide a

valid comparison.) New or revised safety analyses should demonstrate that the LEU core can safely withstand the accident. That is, these analyses should show that the consequences meet the acceptance criteria that were used and accepted for the HEU-fueled reactor. Alternatively, these analyses should show that the accidents meet the acceptance criteria that are applicable to current non-power reactor licensing activities (e.g., 10 CFR Parts 20 or 100). If new safety analyses are required for the conversion, all enveloping scenarios should be identified and analyzed for the LEU reactor.

13.1 Maximum Hypothetical Accident (MHA)

The MHA is the hypothetical accident in which the potential radiological consequences to the public health and safety are greater than those from any other postulated event at the facility The MHA has the greatest radiological consequences of analyzed accidents for a non-power reactor to the facility staff, the public, and the environment The MHA is usually, but not always, associated with assumed fuel cladding or fission product retention failure and fission product release. This accident usually assumes conditions that are not considered credible, but are bounding and demonstrate that under the most extreme conditions and assumptions, the radiological consequences at a non-power reactor could not exceed previously used acceptance criteria (e.g., 10 CFR Part 20 or 100).

If the HEU and the LEU reactor characteristics are sufficiently similar, principally in physics, thermal-hydraulics, and radioactive material inventories, the licensee may be able to demonstrate that the conversion from HEU to LEU fuel does not introduce a new and unreviewed type of MHA and does not invalidate the conclusions about HEU reached in the SAR and in the staff's safety evaluations on radiological consequences of the MHA. As previously discussed for MHA scenarios, the licensee may use comparative analyses with relatively simple calculational checks to demonstrate that conversion causes no significant increase in the radiological consequences of the MHA.

On the other hand, if the conversion does create the possibility of a previously unanalyzed MHA scenario or significantly increases the potential consequences of the previously accepted MHA, safety analyses should be submitted for the accident. In this case, the MHA analysis should demonstrate adequate assurance of public health and safety for the proposed LEU core and fuel. That is, the MHA analysis should show that the appropriate acceptance criteria for non-power reactor accident analyses continue to be met or that additional requirements or compensatory measures are proposed to ensure that the acceptance criteria will be met.

The probability or mechanistic details of the MHA scenario need not be described or evaluated Only the consequences need be described and analyzed. MHAs are

facility specific. Additional general guidance may be found in RGs 1.25 and 1.109, NUREG-0851 and -1313, ANSI/ANS 15.7–1977, an NRC memorandum dated June 17, 1981, U.S. Atomic Energy Commission report TID-14844, and in Chapter 13 of this format and content guide.

13.2 Rapid Addition of Reactivity Accident

This accident analysis assumes the addition of the maximum reactivity for a single credible event to the reactor. It also assumes that this maximum reactivity is added to the reactor at the maximum credible rate. Any feedback effects (e.g., the Doppler effect in uranium-238) changed by converting to LEU should be considered. As for other accident analyses, this accident analysis may be carried out by one of the following two methods:

(1) The parameters that are important to this accident should be compared to verify that the LEU fuel and core will not differ significantly from the same accident analyzed for HEU fuel and core. The limiting rapid addition of reactivity accident scenario postulated in the current SAR for HEU fuel should be described, as well as the initial core and reactor conditions that are pertinent to the analysis methods and assumptions for this accident. Using the appropriate analytical methods, the licensee should verify that the consequences arising from this accident scenario for the LEU-fueled core will not result in a significant change from that for the HEU fuel (i.e., there will be no significant reduction in the margin of safety to maintaining fuel integrity) Methods of analyses for licensee consideration are discussed in an article by William L. Woodruff that appeared in February 1984 in the journal *Nuclear Technology*

(2) If a meaningful comparison is not feasible because the calculational techniques have changed since the current SAR analyses for HEU fuel were performed or because the LEU fuel and core characteristics cannot be correlated and shown to be similar to that of the HEU, a new standalone analysis for LEU fuel assessing the rapid addition of reactivity accident safety should be done This new analysis should describe the limiting accident scenario, stating the postulated initial core and reactor conditions that are pertinent, and the analysis methods and assumptions to be used. The analysis of this accident scenario for the LEU-fueled core should demonstrate that the effects of this accident will not result in any localized fuel melting or any challenge to the cladding integrity or fuel fission product retention capability, or, alternatively, will not increase the potential radiation exposure over that acceptable for non-power reactors For validation of the method, analysis of HEU fuel and core conditions should be considered, if applicable, for comparison to the previously

reviewed and accepted analysis of the accident involving a rapid addition of reactivity

13.3 Reduction-in-Cooling Accidents

Two such accidents are the loss-of-coolant accident (LOCA) and the loss-of-flow accident (LOFA). The scenario(s) as postulated in the current SAR and any differences in the progression of the accidents with LEU fuel as compared to HEU fuel should be described. Any mitigating effects provided by an ESF, such as the ECCS, should be analyzed.

Comparison of the parameters of importance to these HEU and LEU accident analyses should verify that the temperature changes over time in the applicable LOCA and LOFA scenarios do not result in safety analysis findings greater than the acceptance criteria for HEU and LEU fuel and cores. The postulated initial core and reactor conditions that are pertinent to the comparison of the HEU and LEU cores and the thermal-hydraulic methods and assumptions should be described. This analysis should be conducted to the point at which LEU fuel cooling and shielding conditions have been stabilized The analysis should demonstrate that the consequences to fuel integrity and the intensity of the radiation field for the LEU-fueled core are not significantly increased from those acceptable for the licensed HEU-fueled core.

If, however, the conversion significantly increases the consequences of the LOCA or LOFA for the LEU-fueled core or introduces a new and previously unanalyzed accident scenario, new safety analyses should be submitted. The consequences arising from these scenarios that involve reduced cooling should not result in any localized fuel melting or breach of cladding integrity or fuel fission product retention capability, or, alternatively, should not significantly increase the potential radiation exposure over the exposure that is acceptable for the non-power reactor accident analyses. Additional guidance may be found in Hunt and DeBevee, 1959; Knexevich et al., 1965; Gulf General Atomics GA 6596, and Chapter 13 of this document.

13.4 Other Accidents

The accident caused by the failure of the fuel fission product barrier generally involves the failure of fuel plate or rod cladding under some mechanistic scenario. As previously discussed, this accident scenario may be the MHA for some non-power reactors It may also assume failures of large fractions of the fuel fission product barrier in order to demonstrate acceptable safety design for a broad variety of potential events and conditions. The potential radiological consequences to the facility staff, the public, and the environment should be evaluated using conservative dose calculation methods and assumptions

As discussed in Section 13.1 of this appendix, if the HEU and the LEU reactor characteristics are sufficiently similar, the licensee may be able to demonstrate that the conversion does not introduce a new and unreviewed type of such an accident and does not invalidate the HEU SAR and staff conclusions on the associated radiological consequences. The licensee may use comparative analyses with relatively simple calculational checks to demonstrate that conversion causes no significant increase in the radiological consequences of the fission product barrier failure. On the other hand, if the conversion does create the possibility of a previously unanalyzed scenario or significantly increases the potential consequences of the previously accepted accident analyses, a revised SAR should be submitted for the accident to provide reasonable assurance that the radiological consequences will not endanger the facility staff, the public, and the environment.

Other postulated accidents may be affected by or result from the conversion from HEU to LEU fuel. Such reactor-specific accidents as the following should be identified and analyzed:

- misplaced experiments or experiment malfunctions (considering different reactivity response for the LEU and HEU fuel)
- mishandling or malfunction of fuel (considering potential differences in radioactivity release for LEU and HEU fuel)
- loss of normal electrical power
- external events
- mishandling or malfunction of specific equipment
- leakage of coolant into the cladding with a resulting steam pressure failure of the cladding
- failure of a fueled or unfueled experiment
- physical damage to the core

14 TECHNICAL SPECIFICATIONS

Changes in the technical specifications that are required by the conversion should be proposed and explained. Technical specifications that may need to be changed are those specifications and their bases that relate to the fuel (e.g., core reactor physics and thermal-dynamic parameters dependent on fuel characteristics, and fuel storage requirements), including safety limits and limiting safety system settings.

In addition, the technical specifications design features that describe the core and fuel parameters may need to be modified.

The changes in the technical specifications should be the minimum required by the differences in the characteristics of the HEU and LEU fuels. These changes should be traceable to the HEU to LEU conversion safety analyses from which they are derived. Any new or modified structures, systems, or components required for the HEU to LEU conversion should be added or appropriately changed in the applicable sections of the technical specifications (e.g., new or modified ESFs).

The licensee should supply appropriate technical specifications consistent with the assumptions of the LEU safety assessment. Technical specifications should contain the necessary requirements specified in 10 CFR 50.36. The format and content of the revised technical specifications should be maintained the same as the current technical specifications. If a change in content of the technical specifications is appropriate for the LEU conversion, additional guidance is provided in ANSI/ANS 15.1–1990 and Chapter 14 of this document.

15 OTHER LICENSE CONSIDERATIONS

15.1 Prior Utilization of Reactor Components

If the HEU fuel will be replaced by new, unirradiated, recently fabricated LEU fuel that has been approved by the NRC (such as uranium silicide–aluminum dispersion fuel or high-uranium-density, 20-percent-enriched TRIGA fuel), this discussion may not be applicable.

If the replacement fuel is not new (e g., if it has been irradiated, such as previously exposed TRIGA fuel), or if it was not recently fabricated (such as SPERT fuel), histories and safety implications of using the LEU fuel should be analyzed. This analysis should consider custodianship; earlier operations; environmental conditions under which the fuel was used and stored, operational factors such as power level, neutron flux and fluence level, burnup, temperatures and temperature gradients across the fuel and the cladding, and coolant flow rates; pulsing; and chemical composition and temperature of the coolant in which the fuel resided over its lifetime

The effect of these conditions on fuel integrity should be analyzed, including consideration of irradiation and test experience for similar or identical fuel; erosion and corrosion resulting from previous use; internal pressure and composition of gas, including fission product gases; fuel inspection results and plans; and startup test plans and acceptance criteria established earlier in this conversion proposal.

If appropriate, prior use of other components required for the conversion to LEU (e.g., experimental facilities that are compatible with fuel design) or of existing reactor components should also be analyzed.

15.2 License Conditions

The need for revised license conditions for the possession of the LEU fuel should be considered since the LEU fuel and core will in all likelihood have a different amount of uranium-235 than the HEU fuel and core. To arrive at a "reasonable possession" limit, the amount of LEU fuel assemblies that are needed to operate the reactor plus an adequate number of LEU fuel assemblies to accommodate the time necessary to get additional fuel to compensate for burnup, fuel failure, or other reasons to replace fuel should be explained. An example of a license condition for possession is the following:

> ... pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to *x.x* kilograms of contained uranium-235 at enrichments less than 20 percent in the form of non-power reactor fuel in connection with operation of the reactor; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

The licensee should propose similar wording or the wording in the current license for possession of material. It should be noted that other special nuclear material (e.g., sources) must be specified in the license to allow possession.

A license change to possess but not use the HEU fuel until it can be shipped off site and is no longer under the jurisdiction of the reactor licensee also should be proposed if the plans are to possess both HEU and LEU fuel simultaneously. An example follows:

> ...pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to *x.x* kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment and other such special nuclear material produced by operation of the facility in the form of non-power reactor fuel until this fuel is removed from the facility.

15.3 Decommissioning

The effects of conversion to LEU should be evaluated. Changes may involve activation changes due to spectrum hardening and flux distribution changes. Additional guidance in this regard may be found in Chapter 17 of the format and content guide

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APPENDIX A

APPLICABILITY OF SELECTED REGULATIONS IN TITLE 10, CHAPTER I, OF THE CODE OF FEDERAL REGULATIONS TO NON-POWER REACTORS

Applicability of Selected Regulations in Title 10, Chapter I, of the *Code of Federal Regulations* to Non-Power Reactors

Licensees have occasionally asked the staff about the applicability of certain regulations to non-power reactors. In this appendix to the format and content guide, the staff gives the results of its assessment of the applicability of the regulations in selected parts of Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I, to non-power reactors. The assessment was based on the staff's technical background and past regulatory experience. This appendix is not an interpretation of the regulations and should not be used by licensees and other interested parties as such. The applicability of the regulations to non-power reactors as determined by the assessment will not be recognized to be binding on the Commission. Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding on the Commission.

The regulations are updated as the need arises by rulemaking. This appendix is based on the regulations that were in effect as of July 1, 1995. In assembling the information in this appendix, the staff assumed that there are no commercial (as defined by 10 CFR 50.22) non-power reactors, although for the purpose of fee collection, there are several commercial non-power reactors as defined in 10 CFR Part 170, which uses a different test than does 10 CFR Part 50. None of the non-power reactors licensed today meet the 10 CFR Part 50 definition of commercial. The applicability of the regulations for a commercial non-power reactor as defined in 10 CFR Part 50 would be different from that for non-commercial, non-power reactors.

The regulations are presented in two sections.

- Section I, "Selected 10 CFR Parts Applicable to Non-Power Reactors (Including Test Reactors) Licensed in Accordance With 10 CFR Part 50," contains a listing of all of those sections in 10 CFR Parts 19, 20, 30, 50, 70, 170, and 171 that are applicable to non-power reactors. Several regulations in this section are only applicable to research reactors and are clearly marked as such.

- Section II, "Selected 10 CFR Parts Applicable to Test Reactors Licensed in Accordance With 10 CFR Part 50," contains a listing of additional regulations in 10 CFR Parts 20 and 50 that are applicable only to test reactors (also called testing facilities in certain regulations).

Other parts of the regulations may also be applicable to non-power reactors if certain events occur. For example, 10 CFR Part 36, "Licenses and Radiation Safety Requirements for Irradiators," may be applicable to non-power reactors if they have an irradiator. Because certain events must occur for other parts of the regulations to be applicable, the staff cannot make a statement about the applicability of other parts to all non-power reactors.

SECTION I

SELECTED 10 CFR PARTS APPLICABLE TO NON-POWER REACTORS (INCLUDING TEST REACTORS) LICENSED IN ACCORDANCE WITH 10 CFR PART 50

Part 19 - Notices, Instructions, and Reports to Workers: Inspection and Investigations

19.1	Purpose
19.2	Scope
19.3	Definitions
19.4	Interpretations
19.5	Communications
19.8	Information collection requirements: OMB [Office of Management and Budget] approval
19.11	Posting of notices to workers
19.12	Instructions to workers
19.13	Notifications and reports to individuals
19.14	Presence of representatives of licensees and workers during inspections
19.15	Consultation with workers during inspections
19.16	Requests by workers for inspections
19.17	Inspections not warranted; informal review
19.18	Sequestration of witnesses and exclusion of counsel in interviews conducted under subpoena
19.20	Employee protection
19.30	Violations
19.31	Application for exemptions
19.32	Discrimination prohibited
19.40	Criminal penalties

Part 20 - Standards for Protection Against Radiation

Subpart A - General Provisions

20.1001	Purpose
20.1002	Scope
20.1003	Definitions
20.1004	Units of radiation dose
20.1005	Units of radioactivity
20.1006	Interpretations
20.1007	Communications
20.1008	Implementation
20.1009	Information collection requirements: OMB approval

Subpart B - Radiation Protection Programs

20.1101	Radiation protection programs

Subpart C - Occupational Dose Limits

20.1201	Occupational dose limits for adults
20.1202	Compliance with requirements for summation of external and internal doses
20.1203	Determination of external dose from airborne radioactive material
20.1204	Determination of internal exposure
20.1205	[Reserved]
20.1206	Planned special exposures
20.1207	Occupational dose limits for minors
20.1208	Dose to an embryo/fetus

Subpart D - Radiation Dose Limits for Individual Members of the Public

20.1301	Dose limits for individual members of the public
20.1302	Compliance with dose limits for individual members of the public

Subpart E - [Reserved]

Subpart F - Surveys and Monitoring

20.1501	General
20.1502	Conditions requiring individual monitoring of external and internal occupational dose

Subpart G - Control of Exposure From External Sources in Restricted Areas

20.1601	Control of access to high radiation areas
20.1602	Control of access to very high radiation areas

Subpart H - Respiratory Protection and Controls To Restrict Internal Exposure in Restricted Areas

20.1701	Use of process or other engineering controls
20.1702	Use of other controls
20.1703	Use of individual respiratory protection equipment
20.1704	Further restrictions on the use of respiratory protection equipment

Subpart I - Storage and Control of Licensed Material

20.1801	Security of stored material
20.1802	Control of material not in storage

Subpart J - Precautionary Procedures

20.1901	Caution signs
20.1902	Posting requirements
20.1903	Exceptions to posting requirements
20.1904	Labeling containers
20.1905	Exemptions to labeling requirements
20.1906	Procedures for receiving and opening packages

Subpart K - Waste Disposal

20.2001	General requirements
20.2002	Method for obtaining approval of proposed disposal procedures
20.2003	Disposal by release into sanitary sewerage
20.2004(a)	Treatment or disposal by incineration
20.2005	Disposal of specific wastes
20.2006	Transfer for disposal and manifests
20.2007	Compliance with environmental and health protection regulations

Subpart L - Records

20.2101	General provisions
20.2102	Records of radiation protection programs

20.2103	Records of surveys
20.2104	Determination of prior occupational dose
20.2105	Records of planned special exposures
20.2106	Records of individual monitoring results
20.2107	Records of dose to individual members of the public
20.2108	Records of waste disposal
20.2110	Form of records

Subpart M - Reports

20.2201(a)(1)	Reports of theft or loss of licensed material
20.2201(a)(2)(i)	
20.2201(a)(2)(ii)	
20.2201(b)(1)	
20.2201(b)(2)(ii)	
20.2201(c)	
20.2201(d)	
20.2201(e)	
20.2202(a)	Notification of incidents
20.2202(b)	
20.2202(c)	
20.2202(d)(1)	
20.2202(d)(2)	
20.2202(e)	
20.2203(a)	Reports of exposures, radiation levels, and concentrations of radioactive material exceeding the limits
20.2203(b)	
20.2203(d)	
20.2204	Reports of planned special exposures
20.2205	[Reserved]

Subpart N - Exemptions and Additional Requirements

20.2301	Applications for exemptions
20.2302	Additional requirements

Subpart O - Enforcement

20.2401	Violations
20.2402	Criminal penalties

Appendix A to Part 20 - Protection Factors for Respirators

Appendix B to Part 20 - Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

Appendix C to Part 20 - Quantities of Licensed Material Requiring Labeling

Appendix D to Part 20 - United States Nuclear Regulatory Commission Regional Offices

Appendix E to Part 20 - [Reserved]

Appendix F to Part 20 - Requirements for Low-Level-Waste Transfer for Disposal at Land Disposal Facilities and Manifests

Appendix G to Part 20 - Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests

Part 30 - Rules of General Applicability to Domestic Licensing of Byproduct Material

Comment - Non-power reactor licenses issued under 10 CFR Part 50 contain authorization to receive, possess, and use byproduct material pursuant to 10 CFR Part 30. The Part 50 license states that the receipt, possession, and use of byproduct materials as authorized by the license will be in accordance with the Commission's regulations in 10 CFR Part 30 including Section 30.33.

General Provisions

30.1	Scope
30.2	Resolution of conflict
30.3	Activities requiring license
30.4	Definitions
30.5	Interpretations
30.6	Communications Comment - Also see 10 CFR 50.4.
30.7	Employee protection Comment - Also see 10 CFR 50.7.
30.8	Information collection requirements: OMB approval
30.9	Completeness and accuracy of information Comment - Also see 10 CFR 50.9.
30.10	Deliberate misconduct Comment - Also see 10 CFR 50.5.

Exemptions

30.11	Specific exemptions
30.12	Persons using byproduct material under certain Department of Energy and Nuclear Regulatory Commission contracts
30.13	Carriers
30.14	Exempt concentrations
30.15	Certain items containing byproduct material
30.16	Resins containing scandium-46 and designed for sand-consolidation in oil wells
30.18	Exempt quantities
30.19	Self-luminous products containing tritium, krypton-85, or promethium-147
30.20	Gas and aerosol detectors containing byproduct material

Licenses

30.31	Types of licenses

30.32	Application for specific licenses Comment - 10 CFR 50.31 allows an applicant to combine in one, several applications for different kinds of licenses. The application for a byproduct material license is normally part of the Part 50 license application for a non-power reactor.
30.33	General requirements for issuance of specific licenses
30.34	Terms and conditions of licenses
30.35	Financial assurance and recordkeeping for decommissioning Comment - Also see 10 CFR 50.33 and 50.75.
30.36	Expiration and termination of licenses Comment - Also see 10 CFR 50.82.
30.37	Application for renewal of licenses Comment - Also see 10 CFR 50.51.
30.38	Application for amendment of licenses Comment - Also see 10 CFR 50.90.
30.39	Commission action on applications to renew or amend
30.41	Transfer of byproduct material

Records, Inspections, Tests, and Reports

30.50	Reporting requirements
30.51	Records
30.52	Inspections
30.53	Tests
30.55	Tritium reports

Enforcement

30.61	Modification and revocation of licenses
30.62	Right to cause the withholding or recall of byproduct material
30.63	Violations
30.64	Criminal penalties

Schedules

30.70	Schedule A - Exempt concentrations
30.71	Schedule B
30.72	Schedule C - Quantities of radioactive materials requiring consideration of the need for an emergency plan for responding to a release Comment - Also see 10 CFR 50.47

Appendix A to Part 30 - Criteria Relating to Use of Financial Tests and Parent Company Guarantees for Providing Reasonable Assurance of Funds for Decommissioning

Appendix B to Part 30

Appendix C to Part 30 - Criteria Relating to Use of Financial Tests and Self Guarantees for Providing Reasonable Assurance of Funds for Decommissioning

Part 50 – Domestic Licensing of Production and Utilization Facilities

General Provisions

50.1	Basis, purpose, and procedures applicable
50.2	Definitions
50.3	Interpretations
50.4(a)	Written communications
50.4(b)	
50.4(b)(1)	
50.4(b)(2)	
50.4(b)(2)(i)	
50.4(b)(3)	
50.4(b)(4)	
50.4(b)(5)	
50.4(b)(5)(i)	
50.4(b)(5)(ii)	
50.4(b)(5)(iii)	
50.4(c)	
50.4(d)	
50.4(e)	
50.4(f)	
50.5	Deliberate misconduct
50.7	Employee protection
50.8	Information collection requirements: OMB approval
50.9	Completeness and accuracy of information

Requirement of License, Exceptions

50.10(a)	License required
50.10(b)(1)	
50.10(b)(2)	
50.10(b)(3)	
50.10(b)(4)	Comment - This regulation allows universities to construct without a construction permit multipurpose buildings that will be used for the research reactor at some time in the future. It applies to research reactors only. It does not apply to test reactors.
50.10(c)	
50.10(c)(1)	
50.10(c)(2)	
50.10(c)(3)	Comment - This regulation applies to research reactors only. It does not apply to test reactors.

50.10(d)	
50.11	Exceptions and exemptions from licensing requirements
50.12	Specific exemptions
50.13	Attacks and destructive acts by enemies of the United States; and defense activities

Classification and Description of Licenses

50.20	Two classes of licenses
50.21	Class 104 licenses; for medical therapy and research and development facilities
50.21(a)	
50.21(c)	
50.22	Class 103 licenses; for commercial and industrial facilities Comment - This regulation contains the test for determining if a non-power reactor is a commercial or industrial facility.
50.23	Construction permits

Applications for Licenses, Form, Contents, Ineligibility of Certain Applicants

50.30(a)(1)	Filing of applications for licenses; oath or affirmation
50.30(a)(2)	
50.30(a)(5)	
50.30(a)(6)	
50.30(b)	
50.30(c)	[Removed]
50.30(d)	
50.30(e)	
50.30(f)	
50.31	Combining applications
50.32	Elimination of repetition
50.33	Contents of applications; general information
50.33(a)	
50.33(b)	
50.33(c)	
50.33(d)	
50.33(e)	
50.33(f)	
50.33(h)	
50.33(j)	
50.33(k)	
50.34(a)	Contents of applications; technical information
50.34(a)(1)	
50.34(a)(2)	

50.34(a)(3)	
50.34(a)(4)	
50.34(a)(5)	
50.34(a)(6)	
50.34(a)(7)	
50.34(a)(8)	
50.34(a)(9)	
50.34(a)(10)	
50.34(b)	
50.34(b)(1)	
50.34(b)(2)	
50.34(b)(2)(i)	
50.34(b)(3)	
50.34(b)(4)	
50.34(b)(5)	
50.34(b)(6)	
50.34(b)(6)(i)	
50.34(b)(6)(ii)	
50.34(b)(6)(iii)	
50.34(b)(6)(iv)	
50.34(b)(6)(v)	
50.34(b)(6)(vi)	
50.34(b)(7)	
50.34(b)(8)	
50.34(c)	
50.34(d)	Comment - See also 10 CFR 73.60.
50.34(e)	
50.35	Issuance of construction permits
50.36(a)	Technical specifications
50.36(b)	
50.36(c)	
50.36(c)(1)	
50.36(c)(1)(i)(A)	
50.36(c)(1)(ii)(A)	
50.36(c)(2)	
50.36(c)(3)	
50.36(c)(4)	
50.36(c)(5)	
50.36(c)(6)	
50.36(c)(7)	
50.36(d)	
50.37	Agreement limiting access to restricted data
50.38	Ineligibility of certain applicants
50.39	Public inspection of applications

Standards for Licenses and Construction Permits

50.40	Common standards
50.41	Additional standards for Class 104 licenses
50.41(a)	
50.41(b)	
50.45	Standards for construction permits

Issuance, Limitations, and Conditions of Licenses and Construction Permits

50.50	Issuance of licenses and construction permits
50.51	Duration of licenses, renewal
50.52	Combining licenses
50.53	Jurisdictional limitations
50.54	Conditions of licenses
50.54(b)	
50.54(c)	
50.54(d)	
50.54(e)	
50.54(f)	
50.54(g)	
50.54(h)	
50.54(i)	
50.54(i-1)	
50.54(j)	
50.54(k)	
50.54(l)	
50.54(m)(1)	
50.54(n)	
50.54(p)	
50.54(q)	
50.54(r)	
50.54(v)	
50.54(x)	
50.54(y)	
50.54(aa)	
50.54(cc)	
50.54(dd)	
50.54(ee)	
50.55	Conditions of construction permits
50.55(a)	
50.55(b)	
50.55(c)	
50.55(d)	

50.55(e)	
50.55a	Codes and standards
50.55a(a)(1)	
50.55a(a)(3)	
50.55a(b)	
50.55a(b)(1)	
50.55a(b)(2)	
50.55a(b)(2)(i)	
50.55a(b)(2)(ii)	
50.55a(b)(2)(iv)	
50.55a(b)(2)(iv)(A)	
50.55a(b)(2)(v)	
50.55a(b)(2)(vi)	[Reserved]
50.55a(b)(2)(vii)	
50.55a(b)(2)(viii)	
50.55a(c)	
50.55a(c)(1)	
50.55a(c)(2)	
50.55a(c)(3)	
50.55b	[Revoked]
50.56	Conversion of construction permit to license; or amendment of license
50.57	Issuance of operating license
50.59	Changes, tests, and experiments
50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors

Inspections, Records, Reports, Notifications

50.70(a)	Inspections
50.70(b)(1)	
50.70(b)(3)	
50.71(a)	Maintenance of records, making of reports
50.71(c)	
50.71(d)	
50.74	Notification of change in operator or senior operator status
50.75(a)	Reporting and recordkeeping for decommissioning planning
50.75(d)	
50.75(e)(1)	
50.75(e)(2)	
50.75(f)	
50.75(g)	

US/IAEA [International Atomic Energy Agency] Safeguards Agreement

50.78	Installation information and verification

Transfers of License - Creditors' Rights - Surrender of Licenses

50.80	Transfer of licenses
50.81	Creditor regulations
50.82(a)	Application for termination of license
50.82(b)	
50.82(b)(1)	
50.82(b)(1)(ii)	
50.82(b)(1)(iii)	
50.82(b)(2)	
50.82(b)(3)	
50.82(b)(4)	
50.82(b)(5)	
50.82(c)	
50.82(d)	
50.82(e)	
50.82(f)	

Amendment of License or Construction Permit at Request of Holder

50.90	Application for amendment of license or construction permit
50.92(a)	Issuance of amendment
50.92(b)	

Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission

50.100	Revocation, suspension, modification of licenses and construction permits for cause
50.101	Retaking possession of special nuclear material
50.102	Commission order for operation after revocation
50.103	Suspension and operation in war or national emergency

Enforcement

50.110	Violations
50.111	Criminal penalties

Appendix E to Part 50 - Emergency Planning and Preparedness for Production and Utilization Facilities

I	Introduction
II	The preliminary safety analysis report
III	The final safety analysis report
IV	Content of emergency plans
V	Implementing procedures

Part 70 - Domestic Licensing of Special Nuclear Material

Comment - Non-power reactor licenses issued under 10 CFR Part 50 contain authorization to receive, possess, and use special nuclear material pursuant to 10 CFR Part 70. The Part 50 license states that the receipt, possession, and use of special nuclear material as authorized by the license will be in accordance with the Commission's regulations in 10 CFR Part 70 including 10 CFR 70.23 and 70.31.

General Provisions

70.1(a)	Purpose
70.2	Scope
70.3	License requirements
70.4	Definitions
70.5	Communications Comment - Also see 10 CFR 50.4.
70.6	Interpretations
70.7	Employee protection Comment - Also see 10 CFR 50.7.
70.8	Information collection requirements: OMB approval
70.9	Completeness and accuracy of information Comment - Also see 10 CFR 50.9.
70.10	Deliberate misconduct Comment - Also see 10 CFR 50.5.

Exemptions

70.11	Persons using special nuclear material under certain Department of Energy and Nuclear Regulatory Commission contracts
70.12	Carriers
70.14(a)	Specific exemptions
70.14(b)	[Reserved]

General Licenses

70.18	Types of licenses
70.19	General license for calibration or reference sources
70.20	General license to own special nuclear material
70.20a	General license to possess special nuclear material for transport

License Applications

Comment - 10 CFR 50.31 allows an applicant to combine in one, several applications for different kinds of licenses. The application for a special nuclear material license is normally part of the Part 50 license application for a non-power reactor.

70.21(a)(2)	Filing
70.21(a)(3)	
70.21(b)	
70.21(c)	
70.21(d)	
70.21(e)	
70.21(f)	
70.21(g)	
70.22(a)	Contents of applications
70.22(c)	[Reserved]
70.22(d)	
70.22(e)	
70.22(g)	
70.22(i)	
70.22(k)	
70.22(l)	
70.22(m)	
70.23(a)	Requirements for the approval of applications
70.23(a)(1)	
70.23(a)(2)	
70.23(a)(3)	
70.23(a)(4)	
70.23(a)(5)	
70.23(a)(6)	
70.23(a)(7)	
70.23(a)(9)	
70.24	Criticality accident requirements Comment - 10 CFR 70.24(c) refers to critical assembly reactors.
70.25	Financial assurance and recordkeeping for decommissioning Comment - Also see 10 CFR 50.33 and 50.75.

Licenses

70.31(a)	Issuance of licenses
70.31(b)	[Deleted]
70.31(c)	

70.31(d)	
70.32(a)	Conditions of licenses
70.32(b)	
70.32(c)(2)	
70.32(d)	
70.32(e)	
70.32(f)	[Reserved]
70.32(g)	
70.32(h)	[Reserved]
70.32(i)	
70.32(j)	
70.33	Renewal of licenses Comment - Also see 10 CFR 50.51.
70.34	Amendment of licenses Comment - Also see 10 CFR 50.90.
70.35	Commission action on applications to renew or amend
70.36	Inalienability of licenses
70.37	Disclaimer of warranties
70.38	Expiration and termination of licenses Comment - Also see 10 CFR 50.82.

Acquisition, Use, and Transfer of Special Nuclear Material, Creditors' Rights

70.41	Authorized use of special nuclear material
70.42	Transfer of special nuclear material
70.43	[Deleted]
70.44	Creditor regulations

Special Nuclear Material Control, Records, Reports and Inspections

70.50	Reporting requirements
70.51(a)	Material balance, inventory, and records requirements
70.51(b)	
70.51(c)	
70.51(d)	
70.51(i)	
70.52	Reports of accidental criticality or loss or theft or attempted theft of special nuclear material
70.53(a)	Material status reports
70.54	Nuclear material transfer reports
70.55(a)	Inspections Comment - Also see 10 CFR 50.70.
70.55(b)	
70.55(c)(3)	

70.56	Tests
70.60	[Removed]

Modification and Revocation of Licenses

70.61	Modification and revocation of licenses
70.62	Suspension and operation in war or national emergency

Enforcement

70.71	Violations
70.72	Criminal penalties

Part 170 - Fees for Facilities, Materials, Import and Export Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, as Amended

Comment - This part applies to those non-power reactor licensees that are subject to licensing fees (application fees and hourly fees).

General Provisions

170.1	Purpose
170.2	Scope
170.3	Definitions
170.4	Interpretations
170.5	Communications
170.8	Information collection requirements: OMB approval
170.11	Exemptions Comment - This section contains the exemption for nonprofit educational institutions, Government agencies, and State-owned research reactors used primarily for educational training and academic research purposes.
170.12(a)	Payment of fees
170.12(b)(2)	
170.12(c)(2)	
170.12(d)(2)	
170.12(e)(1)	
170.12(g)	
170.12(h)	
170.12(i)	
170.20	Average cost per professional staff-hour

Schedule of Fees

170.21	Schedule of fees for production and utilization facilities, review of standard referenced design approvals, special projects, inspections, and import and export licenses
170.31	Schedule of fees for materials licenses and other regulatory services, including inspections, and import and export licenses

Enforcement

170.41	Failure by applicant or licensee to pay prescribed fees
170.51	Right to review and appeal of prescribed fees

Part 171 - Annual Fees for Reactor Operating Licenses, and Including Holders of Materials Licenses, Compliance, Registrations, and Quality Assurance Program Approvals and Government Agencies Licensed by NRC

Comment - This part applies to those non-power reactor licensees that are subject to annual fees.

171.1	Purpose
171.3	Scope
171.5	Definitions
171.7	Interpretations
171.8	Information collection requirements: OMB approval
171.9	Communications
171.11(a)	Exemptions Comment - This section contains exemptions for nonprofit educational institutions under some circumstances and for federally and State-owned research reactors used primarily for educational training and academic research purposes.
171.11(b)	
171.11(c)	
171.13	Notice
171.15(a)	Annual fees: reactor operating licenses
171.15(e)	
171.15(f)	
171.16	Annual fees: material licensees, holders of certificates of compliance, holders of sealed source and device registrations, holders of quality assurance program approvals, and government agencies licensed by the NRC
171.17	Proration
171.17(a)	
171.19	Payment
171.23	Enforcement
171.25	Collection, interest, penalties, and administrative costs

SECTION II

SELECTED 10 CFR PARTS APPLICABLE TO TEST REACTORS LICENSED IN ACCORDANCE WITH 10 CFR PART 50

Part 20 - Standards for Protection Against Radiation

Subpart M - Reports

20.2206 Reports of individual monitoring

Part 50 - Domestic Licensing of Production and Utilization Facilities

Requirement of License, Exceptions

50.10(e)	License required

Applications for Licenses, Form, Contents, Ineligibility of Certain Applicants

50.30(a)(3)	Filing of applications for licenses; oath or affirmation
50.30(a)(4)	
50.36b	Environmental conditions

Issuance, Limitations, and Conditions of Licenses and Construction Permits

50.58	Hearings and report of the Advisory Committee on Reactor Safeguards

Inspections, Records, Reports, Notifications

50.71(b)	Maintenance of records, making of reports

Amendment of License or Construction Permit at Request of Holder

50.91	Notice for public comment; State consultation
50.91(a)	
50.91(b)	
50.91(c)(1)	
50.92(c)	Issuance of amendment

Appendix C to Part 50 - A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Facility Construction Permits

Appendix Q to Part 50 - Pre-Application Early Review of Site Suitability Issues

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NUREG-1537, Part 1 gives guidance to non-power reactor licensees and applicants on the format and content of applications to the Nuclear Regulatory Commission for licensing actions. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

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