ORG-NRR

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

September 26, 1994

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

SUBJECT: DRAFT APPLICATION FORMAT AND CONTENT GUIDANCE AND REVIEW PLAN AND ACCEPTANCE CRITERIA FOR NON-POWER REACTORS

The U.S. Nuclear Regulatory Commission (NRC) is in the process of developing for Non-Power Reactors (NPRs) a "Format and Content for Applications for the Licensing of Non-Power Reactors" (F&C) and a "Standard Review Plan and Acceptance Criteria for Applications for the Licensing of Non-Power Reactors" (SRP). The purpose of these documents will be similar to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" for nuclear power reactors. This letter forwards drafts of the document Introduction, Chapter 1, "General Description of the Facility," Chapter 6, "Engineered Safety Features," Chapter 8, "Electrical Power Systems," and Chapter 9, "Auxiliary Systems," of the F&C and SRP documents to NPR licensees for comment. Other draft chapters will be provided for comment as they are completed. Drafts of Chapter 5, "Reactor Coolant and Associated Systems," and Chapter 14, "Technical Specifications," were previously transmitted by letter dated July 12, 1993.

The purpose of the F&C is to establish a uniform format for presenting information in applications, to help ensure completeness of information provided, to assist the Commission staff and others in locating information, and it will aid in increasing the efficiency of the review process. The F&C represents a format for applications that is acceptable to the NRC staff. Conformance with the F&C, however, is not required. The principal purpose of the SRP is to assure the quality and uniformity of the staff reviews, to make information about regulatory matters concerning NPRs widely available and to improve the understanding of the staff review process by interested members of the NPR community and the public.

Your comments on these documents are encouraged. They should be sent to:

Director, Non-Power Reactors and Decommissioning
  Project Directorate
United States Nuclear Regulatory Commission
MS: O-11-B20
Washington, D.C. 20555

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Comments should be received by February 28, 1995. Comments will be considered in revising the documents for final issuance as NUREG reports. Questions concerning this project should be directed to the Project Manager for this effort, Alexander Adams, Jr. at (301) 504-1127.

Brian Grimes

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Enclosures:
As stated

I BACKGROUND

This document describes the format and content of the safety analysis report (SAR) to be submitted to the U.S. Nuclear Regulatory Commission (NRC) by an applicant (licensee) of a non-power reactor for a new license, license renewal, or license amendments. A companion document, the "Standard Review Plan and Acceptance Criteria for Applications for the Licensing of Non-Power Reactors"[1] (standard review plan) gives criteria to assist the NRC staff in effecting comparable, complete, and consistent reviews of licensing applications for non-power reactors. Applicants also may review the Standard Review Plan to gain further insight into the review process for non-power reactor applications.

The NRC published several documents that give guidance for commercial power reactors. In 1972, to assist commercial power applicants in preparing SARs for operating licenses for power reactors, the 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),"[2] with revisions in 1973, 1975, and 1978. In 1975 the 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 Standard Review Plan and published the revision as NUREG-0800. In 1987, the staff published an updated revision of NUREG-0800.[3]

The Standard Format and Content of SARs, RG 1.70, and the Standard Review Plan, NUREG-0800, were developed specifically for LWR nuclear power plants. Therefore, the use of these documents by applicants to prepare, or by the NRC staff to review, SARs for non-power reactor facilities may be very cumbersome because of the great differences in the complexity and hazards of non-power reactors as compared to 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.

Both power reactors and non-power reactors are licensed to operate as utilization facilities under Title 10 in accordance with the *Atomic Energy Act* (AEA or Act) of 1954, as amended. 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 two 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, 50.41, and 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 less than current power reactors. Therefore, the accumulated inventory of radioactive fission products in the fuel (in core) of non-power reactors is proportionately 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, those 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: (a) Class 104 for medical therapy and research and development facilities, and (b) Class 103 for commercial and industrial facilities. These classes come from definitions in the *Atomic Energy Act* of 1954, as amended. 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.

All current non-power reactors are licensed as Class 104 facilities. However, the 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 a Class 104a and 104c facility. All other non-power reactors are licensed as Class 104c facilities.

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

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 staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents since the majority of research reactors to date have been able to meet the conservative 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 the 10 CFR Part 20 (10 CFR 20.1 through 20.602 and Appendixes for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and Appendixes 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 30 rem thyroid for occupational exposed people 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. There have been several instances where very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above have been accepted by the staff.

If the facility meets 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. Any further references to 10 CFR Part 100 in this document refer 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 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 the 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 submission. If certain information identified herein is not yet available because the design has not progressed sufficiently, the SAR should include (a) the criteria and design bases used to develop the required information, (b) the concepts and alternatives under consideration, and (c) 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 evaluate definitively whether or not the facility can be constructed and operated in accordance with applicable regulations.

The licensing process follows the legislative requirement for minimum regulation stated in Section 104 of the AEA. A license for facility operation is the legal agreement between the licensee and the NRC and must be adhered to rigorously by both parties. 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 true safety limits may vary by at least an order of magnitude. However, each facility is obligated to adhere to its own license conditions.

II DOCUMENT STRUCTURE

This document describes the standard format and content of non-power reactor SARs. The staff also prepared a companion to this document for the standard review and acceptance criteria for non-power reactors. These documents are complementary, with section titles and numbers that correspond to the SAR sections.

The Introduction discusses the background of this document and describes the general licensing approach to non-power reactors. The structure of the document is summarized in the introduction. The general requirements of the safety analysis are presented along with information on the purpose, applicability and use of this document. The introduction also contains specifications for applications to NRC.

Chapter 1 is a summary of 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 later 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. This chapter includes the safety considerations and features of the reactor.

Chapter 5 lists the design bases and the functional descriptions of the reactor coolant and associated systems, including the primary and secondary (if applicable) systems, and coolant makeup and purification systems. The chapter also describes provisions for adequate heat removal either while the reactor is operating or shut down.

Chapter 6 lists the design bases and functional descriptions 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 assumes an incredible failure that can lead to fuel cladding or fueled experiment containment breech. This accident is used to bound credible accidents in the accident analysis.

Chapter 7 lists the design bases and functional descriptions of the instrumentation and control systems and subsystems with emphasis on safety-related systems and safe reactor shutdown.

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

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

Chapter 10 lists the design bases and functional descriptions of the 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, based on the proposed experimental programs.

Chapter 11 lists the design bases and functional descriptions of the facility radiation protection program and the radioactive waste management program. This chapter 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 facility program to maintain radiation exposures and releases as low as reasonably achievable (ALARA) is described in this chapter. These programs include the control and disposal of radiological waste from both reactor operations and experimental programs.

Chapter 12 lists the bases and functional descriptions 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 reactor startup planning.

Chapter 13 lists the bases, scenarios, and analyses of accidents at the reactor facility. This description includes 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 any ESFs is included in the accident analysis.

Chapter 14 presents the technical specifications (TS), 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. Other issues not discussed in the other chapters of 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 convert an operating license to a possession-only license.

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

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

III GENERAL REQUIREMENTS

Introduction

Section 50.34 of Title 10 requires each applicant for a license to include 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 several 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 such 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 TS based on the SAR. (The TS express an agreement between NRC and the licensee on the way in which the facility will be managed and operated to ensure the protection of the health and safety of facility personnel and the public and protection of the environment.)

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

The SAR is the basic document that justifies licensing of the facility to the NRC and the public. The SAR includes information to facilitate understanding the design bases for the 10 CFR 50.59 change process, for training reactor operators, preparing reactor operator licensing examinations, and preparing for NRC inspections. The SAR should remain an accurate, current description of the facility. The regulations do not require licensees for non-power reactors periodically to update the SAR similar to 10 CFR 50.71(e) for power reactors. However, non-power reactor licensees are encouraged to maintain current SARs on file at NRC by submitting replacement pages with applications for license amendment and 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 topics in this document by submitting an updated, complete SAR to account for any facility changes and any new regulatory requirements. While 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 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 the NRC for additional guidance, if needed.

Purpose of the Format and Content Guidance Document

This document 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 Guidance Document

The NRC staff recommends this document 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. These guidelines will help licensees to prepare complete packages and thus reduce delays from possible NRC requests for additional information. Applications for license amendments should be written in accordance with applicable sections of the guidance document; 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 as part of the amendment application. A complete revision of the SAR is strongly encouraged for license renewal. This format applies 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.

Use of the Standard Format and Content Document

Although the use of this document is not required, 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 to determine if the SAR includes 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

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 document. Such information includes summaries of the responses to NRC regulatory guides or proposed regulations and supplementary information regarding 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 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 for design. The measurements should be given in both SI (System International or metric) and "English" units, with the design unit first, followed by the numerical equivalent and unit in the other system in parentheses.

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.[4] 10 CFR Part 170 presents information on licensing and amendment fees.

- Graphic Presentations

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

- References

  List documents referenced under a heading or topic at the end of the chapter or place them as footnotes on the page where 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 (see 10 CFR 2.790).

- Printing Specifications

  Paper size should be 8-1/2 x 11 inches for text pages and most drawings and graphics. If a larger size paper 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 heading followed by a hyphen and a sequential number (e.g., the third page of the discussion under Section 4, "Reactor," should be 4-3).

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

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

  The document may be mechanically or photographically reproduced. All pages of text may be printed on both sides of the paper. However, each chapter 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 document may be updated or revised by replacing pages. 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. The line should be the same length as the portion changed. Replacement pages may be added in response to the NRC staff requests for additional information.

  All change 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 licensee should contact the Information and Records Management Branch, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555, to obtain specifications and copy requirements before submitting materials other than paper.

The format and content guidance will be revised and updated periodically as needed to clarify the content or correct errors and to incorporate modifications. The revision number and publication date will be printed at the lower right corner of each updated page. The revision numbers and dates need not be the same for all sections because individual sections will be revised as needed. A list of effective pages will indicate the revision numbers for the current sections. As necessary, corresponding changes to the Standard Review Plan will also be made using these methods.

This document was prepared by A. Adams, Jr., Senior Project Manager, Non-Power Reactors and Decommissioning Project Directorate, Division of Operating Reactor Support, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Major contributors to the document include the project manager, S. Weiss, M. Mendonca, and T. Michaels of NRC; R. Carter, W. Carpenter, P. Napper, P. Wheatley, S. Bryan, D. Ebert, and R. Garner 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. Any 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. Notices of errors or omissions should also be sent to the above address.

IV REFERENCES

1. U.S. Nuclear Regulatory Commission, "Standard Review Plan and Acceptance Criteria for Applications for the Licensing of Non-Power Reactors."

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

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

4. Generic Letter 84-18, "Filing of Applications for Licenses and Amendments," July 6, 1984.

I BACKGROUND

This document gives guidance to staff reviewers in the Office of Nuclear Reactor Regulation (NRR) and reviewers under contract to the U.S. Nuclear Regulatory Commission (NRC) for performing safety reviews of applications to construct, modify, or operate a nuclear non-power reactor. The principal purpose of this document is to ensure the quality and uniformity of reviews by presenting a definitive base from which to evaluate applications for license or license renewal. This document also makes information about regulatory matters widely available and helps interested members of the public and the non-power reactor community better understand the review process.

The NRC has published several documents that give guidance for commercial power reactors. In 1972, the NRC issued Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)"[1] to assist commercial power plants in applying for light-water reactor (LWR) licenses. The staff revised RG 1.70 in 1972, 1975, and 1978. In 1975, the NRC issued the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (NUREG-75-087) to ensure the quality, completeness, and uniformity of staff reviews of power reactor safety analysis reports (SARs), and to assist the staff in performing the reviews. In 1981, the staff completely revised the Standard Review Plan and published the revision as NUREG-0800. In 1987, the staff published an updated revision of NUREG-0800.[2]

The staff issued RG 1.70 and NUREG-0800 for LWR nuclear power plants, which are much larger and more complex than non-power reactor facilities. Recognizing that non-power reactor licensees need not be required to comply with the SAR guidelines for power reactors, NRC issued a format and content guide[3] for non-power reactor license applicants and issued this document for the NRC staff to use in reviewing and evaluating SARs submitted for non-power reactors.

Reactors designed and operated for research, development, education, and medical therapy are called non-power reactors (defined in the *Code of Federal Regulations*, Title 10, Section 50.2 (10 CFR 50.2)). This class of reactors includes research reactors (defined in 10 CFR 170.3) and testing facilities (also referred to as test reactors in some regulations), which are defined in 10 CFR 50.2 and 10 CFR 100.3. The format and content document contains additional information on the classification of non-power reactors.

Both power reactors and non-power reactors are licensed to operate as utilization facilities under Title 10 in accordance with the *Atomic Energy Act* (AEA or Act) of 1954, as amended. 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, 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, 50.41, and 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 less than current power reactors. Therefore, the accumulated inventory of radioactive fission products in the fuel (in core) of non-power reactors is proportionately less and requires less stringent and less prescriptive measures to give equivalent protection for 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 will be implemented in a different way for each category of reactor consistent with protecting the health and safety of the public, workers, and the environment. 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.

Section 50.34 of Title 10 requires that each application for a construction permit for a nuclear reactor facility include a preliminary safety analysis report (PSAR) and that each application for a license to operate such a facility include a final safety analysis report (FSAR). A single SAR document may be acceptable for non-power reactors but must be sufficiently detailed to permit the NRC staff to determine whether or not the facility can be built and operated consistent with applicable regulations.

Most of the design, operating, and safety considerations for non-power reactors apply to both test and research reactors. The guidance herein for reviewing submittals and the criteria for acceptability should be followed for all non-power reactors. Differences for test reactors will be discussed in the applicable chapters.

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 staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents since the majority of research reactors to date have been able to meet the conservative 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 the 10 CFR Part 20 (10 CFR 20.1 through 20.602 and Appendixes for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and Appendixes 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 30 rem thyroid for occupational exposed people 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. There have been several instances where very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above have been accepted by the staff.

If the facility meets 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. Any further references to 10 CFR Part 100 in this document refer to test reactors only.

The SAR for a new facility should describe the design of the facility in sufficient detail to enable the reviewer to evaluate definitively whether the facility can be constructed and operated in accordance with applicable regulations.

The regulations (see 10 CFR 2.105(c)) do not preclude, and the NRC prefers, a joint application for a construction permit and operating license for the initial licensing of a research reactor facility. If well planned, the final facility design and the final SAR descriptions, analyses, and conclusions will not be significantly changed from those in the initial application, and a one-step licensing procedure can be undertaken. To initiate this process, the application should request both a construction permit and an operating license to be issued when construction and operating readiness are acceptable to NRC. The submitted SAR should be complete, appropriate, and acceptable for both permits. This allows a joint notice of intent to be published in the *Federal Register* at the construction permit stage that includes issuance of the operating license without further prior notice when appropriate. The joint application and joint notice procedure streamlines the licensing process. If a final SAR is submitted which documents changes made during construction, it must demonstrate that the facility design and the safety conclusions of the previous SAR documents are unchanged.

The Standard Review Plan covers a variety of site conditions and plant designs. Each section includes the necessary procedures and acceptance criteria for all areas of review pertinent to that section. However, the reviewer may select and emphasize particular aspects of each Standard Review Plan section, as is appropriate for the application. In some cases, the major portion of the review of a facility feature may be done generically with the designer of that feature rather than during reviews of each particular application. In other cases a facility feature may be sufficiently similar to that of a previously reviewed facility so that an additional review of the feature is not needed. For these and other similar reasons, the reviewer may choose not to carry out in detail all of the review steps listed in each Standard Review Plan section for every application. Rational for each decision should be documented in the appropriate section of the SAR.

II DOCUMENT STRUCTURE

The standard format and content document and this review document for non-power reactors are complementary documents with similar section titles and numbers that correspond to the SAR sections. This document consists of subsections for areas of review, acceptance criteria, review procedures, and evaluation findings for each section of the SAR to be reviewed and evaluated. The subsections are defined as follows:

- Areas of Review. This subsection describes the scope of review, including a description of the systems, components, analyses, data, or other information that is part of the particular safety analysis section in question.

- Acceptance Criteria. This subsection states the purpose of the review, the applicable NRC requirements, and the technical bases for determining the acceptability of the design or the programs within the scope of the review. The technical bases consist of specific criteria, such as NRC regulatory guides, codes and standards, branch technical positions, and other criteria that apply to non-power reactors.

  NRR technical positions or practices describe the technical bases for sections of the Standard Review Plan. These positions typically explain the solutions and approaches determined to be acceptable in the past by reviewers dealing with a safety-related design area or analyses. These solutions and approaches are presented in this form so that reviewers can take uniform positions for these issues in future cases.

  The technical positions in these documents represent solutions and approaches that are acceptable but not required as the only solutions and approaches. However, applicants should recognize that, as in the case of regulatory guides, the NRC spent substantial time and effort preparing the technical positions, and a corresponding amount of effort would probably be required to review and accept new or different solutions and approaches. Thus, applicants proposing solutions and approaches differing from those described in the technical positions may expect longer review times and more extensive questioning in these areas.

- Review Procedures. This subsection discusses how the review is accomplished and is generally a description that the reviewer follows to verify that the applicable safety criteria have been met.

- Evaluation Findings. This subsection presents the type of conclusions needed to accept the particular review area. The safety evaluation report should include a conclusion for each section to document the results of the review.

Appendix A of this document is an example of an NRC Safety Evaluation Report.

Although not specifically discussed in each section of the Standard Review Plan, each safety evaluation report section should describe the review, including the aspects of the review that were selected or emphasized, matters that were modified by the applicant or required additional information, the design of the plant or the programs of the applicant that deviated from the criteria stated in the Standard Review Plan, and the bases for any deviations from the Standard Review plan.

Selected chapters end with a reference section citing the documents, regulations, and articles referenced in the Standard Review Plan.

The Standard Review Plan and the format and content guidance documents were developed for all designs and generally apply to non-power reactors of all power levels. However, license applicants for 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 Standard Review Plan and format and content guidance incorporate many years of experience in applying regulatory requirements to evaluate the safety of non-power reactors and to review SARs. These documents are part of the NRC's continuing effort to improve regulatory standards by documenting current methods of review and establishing a baseline for orderly modifications of the review process in the future.

In 1991-1994, the NRC wrote these documents with three major objectives: (1) to discuss the NRC requirements germane to each review topic, (2) to describe how the review determines that if the requirements have been satisfied, and (3) to document the practices developed by NRR in previous regulatory efforts for non-power reactors.

The staff will periodically revise this document to clarify the content, correct errors, and incorporate modifications. The revision number and publication date will be printed at the lower right corner of each revised page. The revision numbers and dates need not be the same for all sections because each section will be reissued only when it is revised. A list of affected pages will indicate the revision numbers for the current sections. The staff will also follow these methods in making corresponding changes to the format and content guidance document.

This document was prepared by A. Adams, Jr., Senior Project Manager, Non-Power Reactors and Decommissioning Project Directorate, Division of Operating Reactor Support, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Major contributors to the document include the project manager, S. Weiss, M. Mendonca, and T. Michaels of NRC; R. Carter, W. Carpenter, P. Napper, P. Wheatley, S. Bryan, D. Ebert, and R. Garner 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. Any 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. Notices of errors or omissions should also be sent to the same address.

III GENERAL REQUIREMENTS

Most non-power reactor operating licenses are issued for 20 years. These licenses permit the non-power reactor to operate within the constraints of the technical specifications derived from the SAR. Each non-power reactor facility applying for an initial license or for license renewal should submit an SAR that follows the standard format and content guidance.

The SAR incorporates the formal documentation for a facility, presenting basic information about the design bases, and the considerations and reasoning used to support the conclusion by the applicant that the facility can be operated safely. The descriptions and discussions also support the assumptions and methods of analysis of postulated accidents, including the maximum hypothetical accident (MHA), and the design of engineered safety features (ESF), if any, used to mitigate accident consequences. The MHA assumes an incredible failure that can lead to fuel cladding or fueled experiment containment breech. This accident is used to bound credible accidents in the accident analysis.

The SAR gives the NRC and the public justification for licensing of the facility and gives information for understanding the design bases for the 10 CFR 50.59 change process, for training reactor operators, preparing reactor operator licensing examinations, and preparing for NRC inspections. For these reasons, and 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 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 with applications for license amendment and with the annual report that summarizes changes made without prior NRC approval under 10 CFR 50.59.

While these procedures will not completely eliminate the need to revise sections of the SAR at license renewal, they can reduce the amount of resources needed to revise the SAR. The NRC plans to remind licensees by letter to review the license renewal requirements of 10 CFR 2.109 at least a year before the expiration date of a facility operating license and to contact NRC for additional guidance if needed. A letter with a standard format has been developed for this purpose and is included as Appendix B to this document.

As noted above, 10 CFR 50.34 requires each applicant for a license to include an SAR as part of the application. Although no regulations apply specifically to SARs for non-power reactor license renewal, the NRC staff determined that it cannot effectively arrive at the findings necessary to renew a facility license without reviewing and evaluating a current SAR.

This document and the associated format and content guidance document for licensing also apply for non-power reactor license amendments, such as those for license renewal, power increases, excess reactivity increases, major core configuration changes, and other significant changes to a non-power reactor facility. License renewal applications should address all topics covered in this document to account for facility changes and any new regulatory requirements issued since initial licensing of the facility. Each submittal should specify all safety issues and address them adequately in revised sections of the SAR. The reviewer must confirm that all safety issues have been addressed.

IV REFERENCES

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

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

3. U.S. Nuclear Regulatory Commission, "Format and Content for Applications for the Licensing of Non-Power Reactors."

1 GENERAL DESCRIPTION OF THE FACILITY

Chapter 1 presents an introduction to the safety analysis report (SAR) and the facility. The introduction should give 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 facility description
- 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, research institute, or company name) and briefly state the purpose and intended use of the facility, the 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 of 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 assure 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 discussed briefly.

- 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 (MHA), 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 where detailed analyses are included.

1.3 General Facility Description

The applicant should briefly describe the reactor facility as follows: (a) geographical location; (b) principal characteristics of the site; (c) the principal design criteria, operating characteristics, and safety systems; (d) engineered safety features, if any; (e) instrumentation, control, and electrical systems; (f) reactor coolant and other auxiliary systems; (g) radioactive waste management provisions or system and radiation protection; and (h) 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, 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 might be water purification systems, electrical supplies, heating, ventilation, and air conditioning systems, and the building that houses the reactor room.

- Any other reactor, subcritical, 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 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 instrument 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 can be referenced and used to support analysis in appropriate chapters of the SAR.

1.6 Summary of Operations

The applicant should briefly discuss reactor operations, experimental programs, and mission of the reactor. The actual or proposed operations are important for estimating such parameters 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 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 the DOE for return of the material. A copy of the cover letter for the contract between the applicant or licensee 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 staff has experience with other research reactors, a brief description of this information should be presented. The licensee should state if the facility has not undergone significant or safety-related physical or operational modifications since it was initially licensed, or the last renewal 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 10 CFR 50.59 that effect the SAR descriptions should be provided. If applicable, technical specifications changes should also be given.

1 GENERAL DESCRIPTION OF THE FACILITY

Chapter 1 of the safety analysis report (SAR) is an overview or executive summary of topics covered in detail in other chapters. The applicant should include a general introduction to the SAR and the non-power reactor facility. The applicant should state the purpose of the SAR and briefly describe the application.

1.1 Introduction

1.1.1 Areas of Review

This section is a very brief introduction to the applicant and the facility.

The NRC will review

- Identification and description of the applicant,
- Purpose and intended use of the reactor facility,
- Geographical location,
- Type and power level of the reactor,
- Inherent or passive safety features, and
- Unique design features, such as a pressurized primary coolant system or unique fuel design, which would be notable for an NRC-licensed non-power reactor.

1.1.2 Acceptance Criteria

The following acceptance criteria apply for the introduction:

- The purpose of the SAR is clearly stated.
- The applicant is identified.
- The location, purpose, and use of the facility are briefly described.
- The basic characteristics of the facility that affect licensing considerations are briefly discussed.
- The design or location features included to address basic safety concerns are outlined.
- Any unique safety design features of the facility different from previously licensed non-power reactor facilities are highlighted.

1.1.3 Review Procedures

The reviewer will systematically evaluate the content of Section 1.1 to confirm that the applicant submitted all information requested in the format and content document.

The reviewer should confirm that the introduction includes sufficient information to support conclusions that the applicant and the proposed facility fall within the scope of NRC licensing authority, and that the evaluations and conclusions of other sections of the SAR will address the relevant details of the facility.

1.1.4 Evaluation Findings

The NRC does not write specific evaluation findings on the introduction to the SAR submitted by the licensee because a detailed review is performed of the application in subsequent chapters and evaluation findings are made. Section 1.1 of the NRC safety evaluation report is an introduction to the NRC report and has a standard format. This format is shown in Appendix A to this document which is an example of a safety evaluation report for a research reactor renewal application. The introduction to the NRC safety evaluation report discusses the identity of the applicant, identifies the licensing action that is evaluated, lists the dates of the application and supplements, lists the documents submitted by the applicant, provides information on where the material is available for review by the public, states the purpose of the review, lists the requirements and standards used in the review, and states who performed the review for NRC. The reviewer should use Section 1 of this example safety evaluation report as guidance for the safety evaluation report statements for this section. The statements should be appropriately modified for an initial application for construction and operation.

1.2 Summary and Conclusions for Principal Safety Considerations

1.2.1 Areas of Review

The NRC performs licensing reviews of nuclear reactor facilities to establish the acceptability of the designs and associated considerations for ensuring the health and safety of the public. Therefore, it is essential that the SAR discuss all possibilities for radiological exposure of the public that could result from operation of the facility. This section should summarize the types of radiological exposure, the magnitude of potential radiation exposure, and the design features that control and limit the potential exposure to acceptable levels prescribed by regulations. These safety considerations include the range of normal operations and accident scenarios that influenced the location and design of the non-power reactor facility.

The NRC will review

- Safety criteria proposed by the applicant,
- Principle safety considerations of the facility design,
- Potential radiological consequences of operation and method of providing protection,
- Description of safety of unique design features, and
- Discussion of accidents.

1.2.2 Acceptance Criteria

Discussions of the safety considerations in the SAR should allow the reviewer to conclude that sufficient design features have been included to protect the health and safety of the public, and that no exposures from normal operation would exceed the requirements of 10 CFR Part 20 and the guidance of the facility program for keeping exposures as low as reasonably achievable (ALARA). Accidents should be briefly discussed. The SAR must include all modes of operation and events that could lead to significant radiological releases and exposure of the public.

1.2.3 Review Procedures

The reviewer will systematically evaluate this section to confirm that it includes all required information. The reviewer will consider the stated criteria to ensure safety, and the application to the reactor facility design. The summary discussions and descriptions in this section include such safety considerations as a conservative restricted area to exclude and protect the public, confinement or containment to control radioactive releases, operation with thermal hydraulics parameters that are conservative compared with the designed capabilities of the fuel and cladding, diversity and redundancy of instrumentation and control systems, and other defense-in-depth features. These discussions do not take the place of the detailed analysis in the SAR but briefly summarize some of the information in the SAR. The reviewer will review the detailed discussions as part of the review of other chapters the SAR.

1.2.4 Evaluation Findings

The NRC does not write specific evaluation findings on the summary and conclusions of principal safety considerations presented in the SAR submitted by the applicant because a detailed review is performed of the application in subsequent chapters and evaluation findings are made. This section of the NRC safety evaluation report contains the summary and conclusions of principal safety considerations as determined by the NRC staff. These conclusions are summarized from the analysis of the complete SAR by the reviewer and are not derived from the information in Chapter 1 of the SAR. These summary conclusions and the "findings" at the end of a typical safety evaluation report section are required by NRC in support of the issuance of a license for a non-power reactor. As an example, see 10 CFR 50.56 and 10 CFR 50.57. Statements in a renewal application will differ slightly from those in initial applications. The conclusions the NRC places in this section of the safety evaluation report are as follows:

(1) The design, testing, and performance of the reactor structure and the systems and components important to safety during normal operation were adequately planned, and safe operation of the facility can reasonably be expected.

(2) The management organization of the licensee is adequate to maintain and operate the reactor so that there is no significant radiological risk to the employees or the public.

(3) The expected consequences of several postulated accidents have been considered, emphasizing those likely to cause a loss of integrity of fuel-element cladding. The staff performed conservative analyses of the most serious, hypothetically credible accidents and determined that the calculated potential radiation doses outside the reactor site are not likely to exceed the guidelines of 10 CFR Part 20 for research reactors (10 CFR 20.1 through 20.602 and Appendixes for research reactors licensed before January 1, 1994, or 10 CFR 20.1001 through 20.2402 and Appendixes for research reactors licensed on or after January 1, 1994) or 10 CFR Part 100 for test reactors for doses in unrestricted areas.

(4) The management organization of the licensee, conduct of training and research activities, and security measures are adequate to ensure safe operation of the facility and protection of its special nuclear material.

(5) Releases of radioactive materials and wastes from the facility are not expected to result in concentrations outside the limits specified by regulations of the Commission and are ALARA.

(6) The technical specifications of the licensee, which state limits controlling operation of the facility, give a high degree of assurance that the facility will be operated in accordance with the assumptions and analyses in the SAR. The technical specifications ensure that there will be no significant degradation of equipment.

(7) The financial data submitted by the licensee demonstrate that the licensee has reasonable access to sufficient revenues to cover (construction) operating costs and eventually to decommission the reactor facility.

(8) The licensee program for physically protecting the facility and its special nuclear materials complies with the requirements of 10 CFR Part 73.

(9) The licensee procedures for training its reactor operators and the plan for operator requalification are adequate; they give reasonable assurance the reactor will be operated competently.

(10) The licensee emergency plan provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

1.3 <u>General Facility Description</u>

1.3.1 Areas of Review

This section of the SAR should provide an overview of the facility design, showing how design features implement the safety criteria and safety considerations of Section 1.2. The descriptions should be sufficiently quantitative to clearly summarize the facility to someone who understands non-power reactors. The licensee should include details in later chapters of the SAR. Drawings, tables, and photographs should be included as necessary.

The NRC will review

- The location of the facility and principal characteristics of the site.
- The basic design features, operating characteristics, and safety systems of the reactor and its instrumentation and control and electrical systems.
- The thermal power level of the reactor (and any pulsing capability) and the system that removes and disperses the power.
- The basic experimental features and capabilities in the design.
- Engineered safety features designed to control radiation releases, and
- The design features of the radioactive waste management system or provisions and radiation protection.

1.3.2 Acceptance Criteria

The descriptions in this section should give the information requested in this section of the format and content guidance. Full facility descriptions and analysis are found in and referenced to other sections of the SAR and will be evaluated by the reviewer in other sections of the safety evaluation report.

1.3.3 Review Procedures

The reviewer will systematically compare the descriptions in this section of the SAR to the information requested in the format and content document to confirm that the applicant acceptably covered issues outlined in the acceptance criteria.

1.3.4 Evaluation Findings

Because this section is a simple general description of the facility, this section of the safety evaluation report should contain a short (several paragraphs) description of the facility with no conclusions.

1.4 Shared Facilities and Equipment

1.4.1 Areas of Review

Many non-power reactor facilities will not be housed in a separate building, and many will not have facilities and equipment dedicated only to their use. Some non-power reactor facilities may contain more than one licensed reactor in the same building, and may contain radiation or subcritical nuclear facilities licensed under other NRC or State licenses. The areas of review for this section are brief descriptions and discussions of facilities and equipment shared between the facility described in this SAR and others. This section should summarize the safety implications and relationships among the subject facility and its shared systems or facilities. The shared equipment or functions could be heating and air conditioning, electrical power supplies, cooling and process water, sanitary waste disposal, compressed air, provisions for radiological waste storage and disposal, multipurpose rooms, and cooling towers. Other chapters of the SAR will include detailed descriptions and safety implications of these shared equipment or functions.

1.4.2 Acceptance Criteria

The principal criteria for acceptance should include the following considerations:

- The non-power reactor facility is designed to accommodate all uses or malfunctions of the shared facilities without degradation of the non-power reactor safety features.

- The non-power reactor is designed to avoid conditions in which contamination could be spread to the shared facilities or equipment.

- Where necessary, barriers should be described briefly to ensure that the requirements of the two items above are met.

1.4.3 Review Procedures

The reviewer will verify that all facilities or equipment shared by the non-power reactor have been discussed in the SAR. The applicant should discuss in the SAR how the normal operating use and malfunctions of the licensed facility could affect the other facilities. The reviewer should also assess the SAR discussion of the effect of the shared facilities on the safety of the subject facility. The reviewer may need to review discussions and analysis in other sections of the SAR.

1.4.4 Evaluation Findings

The SAR should contain sufficient information to support the following types of conclusions considering that most of the conclusions in this section summarize the analysis and findings of other parts of the safety evaluation report.

- The shared facilities are clearly and completely listed and the other users identified. The SAR shows that a malfunction or loss of function of these shared facilities would not affect the operation of the non-power reactor. A malfunction or loss of function would not damage the non-power reactor or its capability to be safely shut down.

- Either normal operation or loss of function of the shared facilities would not lead to uncontrolled release of radioactive material from the licensed facility to unrestricted areas, or in the event of release, the exposures are analyzed in Chapter 13, "Accident Analysis," and are found to be acceptable.

1.5 Comparison with Similar Facilities

1.5.1 Areas of Review

Since the early 1940s, several hundred non-power reactors have been built in the United States, and many more in other countries. Most have been based on the safety considerations and principles established by the first few.

Several non-power reactors not licensed by NRC were used as early prototypes or to develop fuels or other components. Examples of prototype or developmental test facilities, whose results were adopted by licensed facilities, include the following:

- Bulk Shielding Facility (BSF),
- Materials Testing Reactor (MTR),
- Special Pulsing Experimental Reactor Tests (SPERT), and
- Chicago Pile #5 or Argonne Research Reactor (CP-5).

Applicants for licenses are expected to use pertinent information from these and other reactors in their design, and reviewers should compare the submitted information with the referenced facility designs. Areas of the SAR that may be reviewed by comparison or reference to similar facilities may include

- Chapters 4, "Reactor Description" and 13 for the bases of NRC's acceptance of fuel performance [e.g., SPERT, Oak Ridge Research Reactor (ORR), System for Nuclear Auxiliary Power (SNAP), General Atomics TRIGA reactor, MTR, and the Advanced Test Reactor (ATR)].

- Chapters 4 and 13 for the bases for reactor core critical size and geometry [e.g., BSF, TRIGA, CP-5, Argonne Nuclear Assembly for CP-11 (Argonaut), and SPERT].

- Chapter 6, "Engineered Safety Features" for the bases of accident mitigation systems (most reactor types).

- Chapter 7, "Instrumentation and Control Systems" for the bases of redundancy and diversity in instruments and controls, including SCRAM systems [e.g., BSF, TRIGA, Omega West Reactor (OWR), MTR, and CP-5], and

- Other specific license conditions acceptable to NRC of other facilities which demonstrate acceptable technical performance (previously licensed facilities with similar thermal power level, similar fuel type, and similar siting considerations).

1.5.2 Acceptance Criteria

The applicant should compare the design bases and operating characteristics of the proposed facility with developmental and similar licensed facilities. If the applicant takes credit for the design and operation of similar facilities in various chapters of the SAR, the parameters compared will be acceptable to the reviewer if the comparisons show that the proposed facility would not exceed the safety envelope of the similar facilities, or if there is reasonable assurance that radiological exposures of the public would not exceed the regulations and the guidelines of the proposed facility ALARA program.

1.5.3 Review Procedures

The reviewer will verify that the characteristics of any facilities compared with the proposed facility are similar and relevant. The reviewer will verify that the operating history of licensed facilities cited by the applicant demonstrate consistently safe operation, use, and protection of the public.

1.5.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The applicant compared the design bases and safety considerations with facilities of similar fuel type, thermal power level, and siting considerations. The history of these facilities demonstrates consistently safe operation that is acceptable to NRC.

- The applicant's design does not differ in any substantive way from similar facilities and should be expected to perform in a similar manner when constructed to that design.

- The applicant used test data from similar reactor facilities in designing components. The actual facilities were cited with the components. These data provide assurance that the facility can operate safely as designed.

The safety evaluation report should contain a summary of the similar facilities discussed by the applicant.

1.6 Summary of Operations

1.6.1 Areas of Review

Many non-power reactors do not operate frequently at the maximum licensed power level, and many operate on demand. Some operate daily at the licensed power level, and some operate continuously with periodic shutdowns for maintenance, fuel shuffling, and experiment changes. Unless there is a safety reason to limit operation of the reactor, the reviewer will assume that the reactor will operate continuously. If there is a safety reason to limit operation of the reactor, then the reactor operating time should be limited by license condition as discussed in the appropriate chapter of the SAR.

The NRC will review the proposed operating plans for a new facility to evaluate the following:

- Possible effect on the power and heat removal capabilities discussed in Chapters 4 and 5 of the SAR,

- Assumed inventory of fission products and source of decay heat, and

- Assumed releases of radioactive effluents to the unrestricted environment.

The reviewer should evaluate the operating characteristics and schedules in an application for license renewal for significant changes and for consistency with the proposed technical specifications.

1.6.2 Acceptance Criteria

The applicant should demonstrate the consistency of proposed operations with the assumptions in later chapters of the SAR including the effect on reactor integrity and potential radiological exposures. The later analyses should demonstrate that the proposed reactor operation was conservatively considered in the design and safety analyses. The proposed operations for license renewal would be acceptable if they are consistent with the assumptions in later chapters of the SAR.

1.6.3 Review Procedures

While the NRC has not issued criteria for evaluating proposed operations, the reviewer should compare proposed operations with the current operations of any similar facilities. The reviewer should verify that proposed operations are summarized and compare them with similar facilities for initial licensing, or previous operations if the application is for license renewal. For license renewal, the reviewer should solicit evaluations by NRC inspectors from the appropriate regional office. Evaluations by NRC regional inspectors and evaluations based on the annual report from the facility should provide additional verification that the licensee can operate the facility as specified in the SAR. If there are limitations on operation of the reactor, verify that they are a license condition.

1.6.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following type of conclusion, which will be included in the safety evaluation report:

> The proposed operating conditions and schedules are consistent with those of similar facilities, and with the design features of the facility. The proposed operations are consistent with relevant assumptions in later chapters of the SAR, where any safety implications of the proposed operations are evaluated. The proposed operating power levels and schedules are in accordance with the proposed license conditions.

1.7 <u>Nuclear Waste Policy Act of 1982</u>

1.7.1 Areas of Review

This section of the SAR should confirm that the licensee or applicant contracted with the U.S. Department of Energy (DOE) to dispose of high-level waste and irradiated (spent) fuel.

1.7.2 Acceptance Criteria

To be acceptable, the reviewer should find a summary of the contract and a reference to an appendix in the SAR where a copy of the contract cover letter can be found.

1.7.3 Review Procedures

The reviewer should compare the content of the SAR with that suggested in Section 1.7 of the format and content guidance document. If necessary, the appropriate DOE representatives could confirm the contract.

1.7.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following type of conclusion, which will be included in the safety evaluation report:

- If the applicant or licensee is a university or government agency:

  Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982 states that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. DOE (R. L. Morgan) informed the NRC (H. Denton) by letter of May 3, 1983, that it has determined that universities and other government agencies operating non-power reactors have entered into contracts with DOE that provide that DOE retain title to the fuel and be obligated to take the spent fuel and/or high-level waste for storage or reprocessing. Because (the applicant) has entered into such a contract with DOE, the applicable requirements of the *Waste Policy Act* of 1982 have been satisfied.

- If the applicant/licensee is a corporation:

  Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982 states that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. (The applicant) has entered into a contract with DOE (Contract #XX-XXXX-XX-XX-XXXX) for the ultimate disposal of the fuel in the (applicant reactor). Because (the applicant) has entered into such a contract with DOE, the applicable requirements of the *Waste Policy Act* of 1982 have been satisfied.

1.8 Facility Modifications and History

1.8.1 Areas of Review

If the SAR describes a new facility, this section need only describe the relevant history of applicant activities before the application and the SAR are submitted, including any experience with other non-power reactors.

If the SAR is submitted as part of a license renewal application, this section should describe the history of the facility, including amendments to the license, with dates and purposes. The licensee should also discuss any significant changes in the previous SAR conditions not requiring NRC approval under 10 CFR 50.59 or other regulations. This discussion includes any significant facility modifications and their effect on operations and releases of radioactive effluents to unrestricted areas.

1.8.2 Acceptance Criteria

The reviewer will find this section acceptable if the facility history is complete.

1.8.3 Review Procedures

The reviewer should compare the information in this section of the SAR with information in the facility docket to verify that the application is complete.

1.8.4 Evaluation Findings

The reviewer should confirm that the information (for license renewal) is complete and consistent with the official docket such as 10 CFR 50.59 changes described in annual reports or inspection report observations. The safety evaluation report should contain a short summary of the history of the facility if the application is for license renewal or if the applicant has previous nuclear experience.

6 ENGINEERED SAFETY FEATURES

6.1 Introduction

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 to within acceptable values. The concept of ESFs evolved from the defense-in-depth philosophy with 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 safety analysis report (SAR) analyses of potential 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 analysis will show that ESFs are not needed.

Normal operation of a non-power reactor is defined as operating 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 a major component failure such as the reactor coolant system boundary or a reactivity addition event. Facilities analyze a maximum hypothetical accident that assumes an incredible failure that leads to fuel cladding or fueled experiment containment breech. These postulated accidents are compared to acceptance criteria such as the safety limits from the technical specifications or, where there are radiological consequences, to accepted regulatory limits (10 CFR Parts 20 or 100). The results of the accident analysis are presented in SAR Chapter 13, "Accidents." 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 presented in SAR Chapter 13 include:

- 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 analysis for many non-power reactors may show that ESFs are not required, even for the maximum hypothetical accident. In other cases the results of the accident analysis 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 suspectable to loss of ability to function due to 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.

Besides providing the design and functional characteristics of each ESF, the methods and criteria for testing to demonstrate ESF system operability should be described. 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.

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. Need for site utility redundancy or diversity and the specific design features that provide it for each ESF component should be discussed and demonstrated.

Schematic diagrams should also be provided, 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). A brief description of the instrumentation and control (I&C) system for each ESF should be provided, 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 of ESFs that may be required at non-power reactors are (1) confinement, (2) containment, and (3) 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 also be treated as part of the ESFs of the confinement or containment system. Any additional ESFs should be discussed in a comparable way.

Brief definitions and illustrations of confinement, containment, and ECCS are presented below:

(1) Confinement is an enclosure of the overall facility (for example, 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 in-leakage (i.e., prevent uncontrolled leakage outside of 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 enclosure could be integrated into the HVAC systems, including exhaust stacks or vents to the external environment, filters, blowers, and dampers.

(2) Containment is an enclosure of the facility designed to (a) be at a negative internal pressure to ensure in-leakage, (b) control the release of effluents to the environment, and (c) mitigate the consequences of certain analyzed accidents. The containment structure is designed (a) to be sealed to support a defined pressure differential across the containment and (b) to have a defined upper limit on leakage from the containment. Both design conditions are testable. An accident scenario that might require a containment enclosure 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 structure, and the design should provide for both normal and emergency operational modes. A containment enclosure 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 structure would not be required. As contrasted with a containment structure, a confinement building and 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 leak tight as a containment and the leak rate is normally not confirmable 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 because the walls are thinner, and
- is less resistant 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 containment buildings 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 staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents since the majority of research reactors to date have been able to meet the conservative 10 CFR Part 20 criteria. American National Standard 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 analysis have generally been limited to values from the old 10 CFR Part 20 (10 CFR 20.1 through 20.602 and Appendixes). 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 occupational exposed people and less than 0.5 rem whole body and 3 rem thyroid for members of the public. However, there have been several instances where very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above have been accepted by the staff. Reactors that receive 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 shall 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. References to 10 CFR Part 100 in this chapter refer to test reactors only.

6.2 Summary Description

This section of the SAR should briefly describe all of the ESFs included in the facility design and summarize the postulated accidents for which they are designed to mitigate. These summaries should include the design bases and performance criteria and provide 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 provide the location, basic function, and relationship of each ESF to the facility. Detailed drawings, schematics, data, and analyses should be presented in subsequent sections of this chapter for specific ESFs.

6.3 Detailed ESF Descriptions

This section of the SAR should discuss in detail the particular ESF systems incorporated into the reactor design. Not all of these ESFs would be found in any single design. There may be other systems that are considered ESFs in addition to the systems discussed in this section. Those ESFs should be discussed by the licensee in a manner similar to the discussions in this section.

6.3.1 Confinement

This section should discuss in detail confinement in combination with 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 presented in SAR Chapter 13.

Confinement buildings and HVAC systems may also have functions that are not considered ESFs and need not be addressed in this chapter. Most non-power reactors release small quantities of airborne radioactive material, primarily Ar-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 air flow 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 building and HVAC system 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 provided in Chapter 3, "Design of Structures, Systems, and Components," and Chapter 9, "Auxiliary Systems." Discussions of diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be addressed in Chapter 11, "Radioactive Waste Management and Radiation Protection."

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 ability 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 building 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 of 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 air flow 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&Cs for the ESF system should be described briefly in this section and in detail in Chapter 7.

This section 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 air flow rates, automatic system shutdown and startup, and activation setpoints should be required and specified. See Chapter 14 of this document, "Technical Specifications," for details on what technical specifications requirements should be identified and justified in this section.

6.3.2 Containment

Because the potential risk to the public is generally low from accidents at non-power reactors, few require a containment structure. However, for those non-power reactor facilities requiring a containment structure and associated HVAC system to mitigate the consequences of a postulated accident, these systems 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 building 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 event is accompanied by a pressure surge or a steam release within the room. The thick walls of the containment may also help mitigate the hazard of direct radiation exposure during certain accidents. The analyses presented in Chapter 13 should provide details of the postulated scenario, including the assumptions and justification of the initiating event, the progression of the scenario, the consequence mitigating effects of 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 event recovery 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 ability 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 being that the containment should reduce the consequences to the public, the facility personnel, and the environment to acceptable values as specified above.

This section of the SAR should explain how the design and functional details of the containment structure 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 analysis 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 structure, the reactor building requires 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 able to maintain 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 analysis presented in Chapter 13. 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 accident ventilation 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 provide 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 components of the containment structure 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 document for details on what technical specifications requirements should be identified and justified in this section.

6.3.3 Emergency Core Cooling System

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

A schematic diagram should show the relationships among the major system components such as valves, spray headers, pumps, piping, and any I&Cs. Special ECCS I&Cs should be described briefly in this section and fully described in Chapter 7. This section should discuss any effects of the ECCS design on normal operations and reactor safety. Analysis 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 presented 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 initiating 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.

This section should also provide 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 document for details on what technical specifications requirements should be identified and analyzed in this section.

6 ENGINEERED SAFETY FEATURES

6.1 Introduction

This chapter provides review and acceptance criteria for active or passive engineered safety features (ESFs) that are designed into the reactor facility to mitigate the consequences of accidents. The concept of ESFs evolved from the defense-in-depth philosophy with multiple design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The applicant determines the need for ESFs from the safety analysis report (SAR) analyses of potential accidents that could occur even though prudent and conservative designs of the facility have made such accidents very unlikely. The NRC reviewer may find that the SAR analysis shows that ESFs are not needed for a particular proposed design.

Normal operation of a non-power reactor is defined as operating 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 generally assume a major component failure such as the reactor coolant system boundary or a reactivity addition event. Facilities analyze a maximum hypothetical accident that assumes an incredible failure that leads to fuel cladding or fueled experiment containment breech. These postulated accidents are compared to acceptance criteria such as the safety limits from the technical specifications or, where there are radiological consequences, to accepted regulatory limits (10 CFR Parts 20 or 100). The results of the accident analysis are presented in SAR Chapter 13, "Accidents." 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 significant radiological risk to the public. Accident scenarios that should be presented by the applicant in SAR Chapter 13 include:

- 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.

In the past, the SAR analysis for many non-power reactors has shown that ESFs are not required, even for the maximum hypothetical accident. In other cases, the results of the accident analysis have shown that ESFs need to be considered in mitigating the potential release of hazardous quantities of radioactive material to the environment.

The accident analyses provided by the applicant should contain 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 accidents should require redundant or diverse ESF systems. Some factors the NRC will review to verify whether redundant or diverse ESF should be required for a particular reactor design are discussed in this chapter.

In addition to reviewing the design and functional characteristics of each ESF, the NRC will also review the methods and criteria proposed by the applicant for testing to demonstrate ESF operability. The necessary components, functional requirements, related setpoints, interlocks, bypasses, and surveillance tests for each ESF will be reviewed and the NRC reviewer will check that they are included in the facility technical specifications. The technical specification surveillance requirements for system components that ensure the integrity and operational capability of the ESFs are also areas to be reviewed.

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 staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents since the majority of research reactors to date have been able to meet the conservative 10 CFR Part 20 criteria. American National Standard ANSI/ANS-15.7, "Research Reactor Site Evaluation," contains additional information on doses to the public from releases of radioactive material.

The design goal of non-power reactor ESFs is to ensure that projected radiological exposures from accidents are kept below the regulatory limits. For a research reactor the reviewer will compare the results of the accident analysis against 10 CFR Part 20. For research reactors licensed before January 1, 1994, the doses calculated in the accident analysis will be acceptable to the reviewer if they are less than the old 10 CFR Part 20 limits (10 CFR 20.1 through 20.602 and Appendixes) of 5 rem whole body and 30 rem thyroid for occupational exposed people and less than 0.5 rem whole body and 3 rem thyroid for members of the public. The reviewer should conduct each review on a case-by-case basis. There have been several instances where very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above have been accepted by the staff. Reactors that receive 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 a research reactor applicant cannot meet the above doses, the reviewer should examine the safety analysis to insure that the evaluation of accidents is not overly conservative.

If the facility meets the definition of a test reactor, the reviewer will compare the results against 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 chapter refer to test reactors only.

The NRC will review the SAR on how the ESFs interact with site utilities, such as electrical power and water, and how the transfer between normal and emergency sources of electricity and water, if applicable, is proposed to be accomplished. The applicant will present any need for site utility redundancy and the specific design features that provide redundancy for components of each ESF.

Schematic diagrams can be provided by the applicant, showing all components, their interrelationships, and the relationship of each ESF to systems used for normal operations (e.g., the emergency core cooling system to the core cooling system or the confinement to the reactor room ventilation system).

Typical of ESFs that may be required for a proposed design are confinement, containment, and emergency core cooling system (ECCS), which are discussed in this chapter of the format and content guide. The postulated accident analyses provided by the applicant determines if a non-power reactor facility needs a confinement system, a containment, an ECCS, or no ESFs. The reviewer will find that heating, ventilation, air conditioning (HVAC) and air exhaust systems at non-power reactors generally serve to limit the release of airborne radioactive material. The reviewer will verify that those features in HVAC systems required to mitigate the consequences of accidents were treated as ESFs. This document gives guidance for the review and evaluation of information on confinement, containment, and ECCS ESFs. Any additional ESFs required at non-power reactors can be evaluated by the reviewer in a similar manner.

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 structure will not be required. If the reviewer confirms that the safety analyses shows that a confinement ESF is sufficient to mitigate the consequences of the most limiting accident to acceptable levels, a containment ESF would not be required. Some licensees have chosen to build containment buildings as an additional design conservatism.

6.2 Summary Description

This section of the SAR briefly describes all of the ESFs included in the facility design and summarizes the postulated accidents whose consequences could be unacceptable without mitigation. A specific postulated accident scenario should indicate the need for each ESF. The details of the accident analyses are provided in Chapter 13 of the SAR and the detailed discussions of the ESFs are provided in Section 6.3 of the SAR. These summaries should include the design bases, performance criteria, and the full range of reactor conditions under which the equipment or systems must maintain function including accident conditions. The evaluation procedures and criteria for confinement, containment and the ECCS are described in Section 6.3 of this document.

The applicant may submit simple block diagrams and drawings which provide the location, basic function, and relationship of each ESF to the facility. The summary description should provide enough information for an overall understanding of the functions and relationships of the ESFs to the operation of the facility. Detailed drawings, schematics, data, and analyses should be presented in Section 6.3 of the SAR for each specific ESF.

6.3 Detailed ESF Descriptions

This section of the SRP discusses in detail particular ESF systems that may be incorporated into the reactor design. Not all of these ESFs would be found in any single design. There may be other systems that are considered ESFs in addition to the systems discussed in this section. Those ESFs should be evaluated by the reviewer in a manner similar to the ESFs in this section.

6.3.1 Confinement

6.3.1.1 Areas of Review

The confinement structure should be discussed in this section of the SAR. If the HVAC and any air exhaust or liquid release systems are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13 and thereby mitigate its consequences, those design features should be considered part of the confinement ESF and should be discussed in this section of the SAR.

Most non-power reactors release a small amount of radioactive material during normal operation. Even though the quantity of radioactive material produced may not be large, the applicant should describe how release to the environment will be controlled. The airborne radionuclide normally released from the envelope of the reactor is Ar-41, which may be continuously swept from the reactor building to diffuse and disperse in the atmosphere. The controlled release must ensure that neither the public nor the facility staff would receive a dose greater than regulatory limits. This function of the confinement and the HVAC system is not considered an ESF. If the effluent control systems provide no unique accident consequence mitigation function, the design bases and detailed discussions of the systems for normal operations should be provided in Chapter 3, "Design of Structures, Systems, and Components," and Chapter 9, "Auxiliary Systems." Discussions and calculations of diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be discussed in Chapter 11, "Radioactive Waste Management and Radiation Protection."

The NRC will review:

- Design bases and functional description of the required mitigative features of the confinement ESFs, derived from the accident scenarios.

- Drawings, schematics, and tables of important design and operating parameters and specifications for the confinement ESFs, including

  - seals, gaskets, filters, and penetrations (e.g., electrical, experimental, air and water)

  - necessary ESF equipment included as part of the confinement structure

  - fabrication specifications for essential and safety-related components.

- Discussion and analyses, keyed to drawings, of how the structure provides the necessary confinement analyzed in Chapter 13 with cross reference to other chapters for discussion of normal operations (such as Chapter 4, "Reactor Description" and Chapter 11), as necessary.

- Description of control and safety instrumentation, including the locations and functions of sensors, readout devices, monitors, and isolation components, as applicable. (Design features should ensure operability in the environment created by the accident.)

- Discussion of the required limitations on release of confined effluents to the environment.

- Surveillance methods and intervals included in the technical specifications that ensure operability and availability of the confinement ESFs, when required.

6.3.1.2 Acceptance Criteria

Acceptance criteria for the information on the confinement and HVAC system ESFs are based on the following:

- The need for a confinement ESF has been properly identified. To be considered an ESF, design features must exist to mitigate the consequences of specific accident scenarios.

- Any ESF in addition to the confinement structure (e.g., HVAC) does not interfere with normal operations or with safe reactor shutdown.

- The ESF design features should ensure that the system is available and operable when the system is required for accident consequence mitigation.

- The minimum design goal of the confinement ESFs should be to reduce below regulatory limits the potential radiological exposures to the facility staff and members of the public for accidents discussed in the introduction of this chapter for test and research reactors. Any additional reduction in potential radiological exposures below the regulatory limits is desirable and should be a design goal if it can be reasonably achieved.

- The design of the confinement system should not transfer undue radiological risk to the health and safety of the public in order to reduce potential exposures to the facility staff.

- The I&C of the confinement ESF systems should be as basic and fail safe as possible. They should be designed to remain functional for the full range of potential operational conditions, including in the environment created by accident scenarios.

- The discussions should identify operational limits, design parameters, surveillances and surveillance intervals which will be included in the technical specifications.

6.3.1.3 Review Procedures

The applicant should show that the confinement ESFs reduce predicted radiological exposures and releases from applicable potential accidents to acceptable levels as discussed in Section 6.1. The staff will examine all accident scenarios analyzed in Chapter 13 that could lead to significant radiological exposures or releases and verify that consequences can be sufficiently mitigated by the confinement ESF. The reviewer will confirm that the design and functional bases of confinement ESFs are derived from the accidents analyzed. The staff will compare the dispersion and diffusion of released airborne radionuclides discussed in SAR Chapters 6 and 13 with methods described in SAR Chapter 11 as applicable.

6.3.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The scenarios for all potential accidents at the reactor facility have been analyzed by the applicant and reviewed by the staff. Mitigation of consequences by a confinement system has been proposed in the SAR analyses for any accident that could lead to potential unacceptable radiological exposures to the public, the facility staff, or the environment.

- The designs and functional descriptions of the confinement ESF have been reviewed; they reasonably ensure that the consequences will be limited to the levels found acceptable in the accident analyses of Chapter 13 of the SAR.

- The designs and functional descriptions of the confinement ESF reasonably ensure that control of radiological exposures or releases during normal operation will not be degraded by the ESF.

- The radiological consequences from accidents to the public, the environment, and the facility staff will be reduced by the confinement ESF to values for research reactors that do not exceed the applicable limits of 10 CFR Part 20 as discussed in Section 6.1 of this document, or for test reactors, 10 CFR Part 100, and in both cases are as far below the regulatory limits as can be reasonable achieved.

6.3.2 Containment

The containment structure should be discussed in this section of the SAR. If the HVAC and any air exhaust or liquid release systems are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13 and thereby mitigate its consequences, those design features should be considered part of the containment ESF and should be discussed in this section of the SAR.

Because the potential risk to the public from accidents at non-power reactors is generally low, most non-power reactors can be designed, sited, and operated so that a containment structure is not required for normal operation or accident mitigation. However, the safety analysis may show that a confinement does not provide sufficient mitigation and a containment is necessary.

Higher power non-power reactors may require a containment for normal operational modes, depending on the operating characteristics or location of the reactor. A containment structure also should be considered necessary for non-power reactor facilities if potential credible accidents, or a maximum hypothetical accident, could lead to unacceptable radiological consequences to the public in the absence of its mitigating functions. There is also the possibility that the analysis of the applicant may show that a confinement is an acceptable ESF, but they choose to construct a containment for additional conservatism.

Most non-power reactors release a small amount of radioactive material during normal operation. Even though the quantity of radioactive material produced may not be large, the applicant should describe how release to the environment will be controlled. The airborne radionuclide normally released from the envelope of the reactor is Ar-41, which may be continuously swept from the reactor building to diffuse and disperse in the atmosphere. The controlled release must ensure that neither the public nor the facility staff would receive a dose greater than regulatory limits. This function of the containment and the HVAC system is not considered an ESF. If the effluent control systems provide no unique accident consequence mitigation function, the design bases and detailed discussions of the systems for normal operations should be provided in Chapter 3, "Design of Structures, Systems, and Components," and Chapter 9, "Auxiliary Systems." Discussions and calculations of diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be discussed in Chapter 11, "Radioactive Waste Management and Radiation Protection."

6.3.2.1 Areas of Review

The areas of review include assumptions and progressions of potential accident scenarios as presented in SAR Chapter 13. The analyses should show if any postulated accident could cause an unacceptable radiological exposure, as discussed above, to the public, the environment, or the facility staff. For any accidents that could cause such an exposure, the analyses should address how the containment ESF prevents rapid release of radiation or radioactive material to the environment and how the ESF design features reduce potential exposures to acceptable levels.

Non-power reactors that are required to have a containment that functions as an ESF during an accident could operate the system as a vented structure for normal operations. For such a use, the applicant should describe the conditions for both uses and the signals and equipment required to initiate switching to the emergency mode. Information on the design of the containment as a vented structure for normal operation should be provided in SAR Chapters 3 and 9 and in Chapter 11 for diffusion and dispersion of airborne radioactivity in restricted and unrestricted environments.

The NRC will review:

- Design bases and functional description of the required mitigative features of the containment structure, derived from the accident scenarios.
- Drawings, schematics, and tables of important design and operating parameters and specifications for the containment structure, including
    - volume and overpressure capability
    - seals, gaskets, filters, and penetrations (e.g., electrical, experimental, air, and water)
    - necessary ESF equipment included as part of the containment structure, and
    - fabrication specifications for essential and safety-related components.
- Discussion and analyses, keyed to drawings, of how the structure provides the necessary containment presented in Chapter 13 with cross reference to other chapters for discussion of normal operation (such as Chapter 4 and Chapter 11), as necessary.
- Description of control and safety instrumentation, including the locations and functions of sensors, readout devices, monitors, and

isolation components, as applicable. (Design features should ensure operability in the environment created by the accident.)

- Discussion of the shielding protection factors provided for direct radiation and the required limitations on leakage or release of contained effluents to the environment.

- Conditions under which operability is required, and surveillance methods and intervals included in the technical specifications that ensure operability and availability of the containment, when required.

6.3.2.2 Acceptance Criteria

Acceptance criteria for the information on the containment ESF are based on the following:

- The need for a containment ESF has been properly identified. To be considered an ESF, design features must exist to mitigate consequences of specific accident scenarios.

- The design meets the minimum design goal of a containment ESF which is to reduce below regulatory limits the potential radiological exposures to the facility staff and members of the public for accidents discussed in the introduction of this chapter. Any additional reduction in potential radiological exposures below the regulatory limits is desirable and should be a design goal if it can be reasonably achieved.

- The containment should not interfere with either normal operation or reactor shutdown.

- The design features and surveillance program should ensure that the system will be available and operable if the ESF system is needed.

- The design of the containment should not transfer undue radiological risk to the health and safety of the public in order to reduce potential exposures to the facility staff.

- The I&C of the containment ESF system should be as basic and fail safe as possible. It should be designed to operate in the environment created by the accident scenario.

- The discussions should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

6.3.2.3 Review Procedures

The staff will review the accident scenarios and the applicable design bases for a containment ESF and the design and functional features of the ESF and the mitigating effects on the radiological consequences evaluated. The net projected radiological exposures will be compared with the limits of 10 CFR Parts 20 or 100 to determine if the design is acceptable.

6.3.2.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The applicant has identified a potential or maximum hypothetical accident from which projected exposures to the public without containment would be greater than acceptable limits.

- The design and functional features proposed for a containment structure reasonably ensure that exposures will be reduced below the limits of 10 CFR Part 20, or Part 100 for test reactors, with an additional factor to achieve residual doses as far below the regulatory limits as can be reasonably achieved. The maximum projected dose to a member of the public is determined from the analyses in SAR Chapter 13 for all analyzed accidents.

- I&C, testing, surveillance provisions and intervals, and related technical specifications reasonably ensure that if required, the containment ESF will be available and operable.

- The design of the containment ESF gives reasonable assurance that it will not interfere with reactor operation or shutdown.

6.3.3 Emergency Core Cooling Systems

6.3.3.1 Areas of Review

For most non-power reactors, heat must be removed from the fuel only during normal operations. In some cases, decay heat from radioactive fission products must be removed from the fuel after the reactor is shut down. Coolant systems described in Chapter 5, "Reactor Coolant Systems," are designed to provide these functions. In the event coolant is accidentally lost, the decay heat in some non-power reactors could be high enough to require a core cooling system to avoid fuel damage from high fuel temperatures.

Each applicant should present in Chapter 13 of the SAR the analysis of a LOCA because at many non-power reactors, the LOCA could be the maximum hypothetical accident that defines the envelope of potential radiological consequences to the facility staff, the public, and the environment.

The NRC will review the design bases and functional requirements of the proposed ECCS for the postulated LOCA through the progression of the accident scenario.

6.3.3.2 Acceptance Criteria

Acceptance criteria on the need for and design of the ECCS are based on the following:

- The design bases and functional description should be derived from a LOCA scenario and presented in SAR Chapter 13.

- The design features ensure that the ECCS will provide the coolant delivery rate for the time interval required by the scenario. The design features ensure that any necessary utility sources, such as normal electricity, emergency power, and coolant, will be available to the ECCS.

- The proposed ECCS should not interfere with either normal operations or reactor shutdown.

- The net consequences of the LOCA event, as mitigated by the ECCS, will not exceed the limits of 10 CFR Parts 20 or 100 and will be as far below the regulatory limits as can be reasonably achieved.

- Technical specifications, containing tests and surveillance, provide reasonable assurance that the ECCS will be operable, if needed.

6.3.3.3 Review Procedures

The staff will review the accidents in Chapter 13 to determine the scenario and consequences for the LOCA and if degradation of the fuel cladding or loss of fuel cladding integrity could result from the LOCA. The reviewer should verify that the proposed ECCS can mitigate degradation of the fuel cladding such as softening or loss of fuel cladding integrity. Fuel limits are discussed in Chapter 14, "Technical Specifications," of this document. The staff will compare the design details of the proposed ECCS with the design and functional requirements of the SAR LOCA accident and the mitigated radiological consequences with 10 CFR Parts 20 or 100 to determine if the design is acceptable.

6.3.3.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The applicant has identified a potential or maximum hypothetical LOCA that could lead to unacceptable fuel degradation or loss of fuel cladding integrity and unacceptable radiological consequences.

- The applicant's analyses of this accident in Chapter 13 includes a proposed ECCS whose design function is to cool the fuel to prevent failure of the fuel cladding.

- The design of the ECCS would not interfere with normal operations and would not prevent safe reactor shutdown.

- The design of the ECCS would not lead to uncontrolled release of radioactive material, including contaminated primary coolant.

- The ECCS is designed and technical specification requirements and procedures exist for periodic surveillance and testing to ensure its operability and availability.

- Design of the ECCS is adequate to operate for the required flow rate and time interval as determined by the accident analysis. The design also considered the availability of normal electrical power and coolant sources and provided alternate sources, if necessary.

- The functioning of the ECCS as designed reasonably ensures that a LOCA at the reactor facility would not subject the public, the environment, or the facility staff to unacceptable radiological exposure.

8 ELECTRICAL POWER SYSTEMS

8.1 Introduction

The purpose of Chapter 8 of the safety analysis report (SAR) is to discuss and describe the electrical power systems at a non-power reactor facility. The electrical power systems to be described in this chapter 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 to assure that, given a loss of normal electric service, sufficient power will be available for mitigating the events discussed in SAR Chapter 13, "Accidents." 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.

8.2 Normal Electrical Power Systems

In this section, the design bases and functional description of the normal electrical power systems for the reactor facility should be provided. 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 instrument 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.

There should be a discussion of 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 SAR 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.

8.3 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. This section of the SAR 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. Examples of uses of emergency electrical power include the following:

- 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. (This section of the SAR should only mention the existence of such emergency electrical power, with details confined to the facility physical security plan.)

- Place or maintain experimental equipment in a safe condition.

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

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

- Power 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 Section 8.3 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, "Radiological Waste Management and Radiological Protection"; and Chapter 13, "Accidents." These systems may range from automatic start generators to wet-cell or dry-cell batteries. This section should present a detailed functional description and circuit diagrams. The design bases should discuss if non-interruptable 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 ov r 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.

8 ELECTRICAL POWER SYSTEMS

8.1 Introduction

This chapter discusses the electrical power systems for a non-power reactor facility. The electrical power systems at non-power reactor facilities are designed to support reactor operations. All non-power reactors require normal electrical service. Some non-power reactors may also require emergency electrical service to perform certain functions related to reactor safety to assure that, given a loss of normal electric service, sufficient power will be available for mitigating the events discussed in safety analysis report (SAR) Chapter 13, "Accidents." The design bases for these functions are provided on a case-by-case basis 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, "Radiological Waste Management and Radiological Protection"; and Chapter 13, "Accidents." Design and functional information in Chapter 8 will be provided under the two categories: normal and emergency electrical power systems.

8.2 Normal Electrical Power Systems

8.2.1 Areas of Review

Normal electrical power systems at non-power reactors are designed for safe operation and shutdown of the reactor, and to provide for reactor use. The areas of review for normal electrical power systems include these functions. In general, non-power reactors are designed for fail-safe passive shutdown by a reactor scram in the event of the loss of offsite electrical services. Therefore, specially designed active systems and components are not generally required. The reactor designer should include high-quality, commercially available components and wiring in accordance with applicable codes in the normal electrical systems.

Specific areas for review for this section are discussed in Section 8.2 of the format and content guide.

8.2.2 Acceptance Criteria

The acceptance criteria for the normal electric power systems at non-power reactor facilities are based on the following:

- The design and functional characteristics should be commensurate with the design bases, which are derived from other chapters of the SAR.
- The facility should have a dedicated substation or a shared system designed to provide reasonable assurance that other uses could not prevent safe reactor shutdown.

- The system should be designed to permit safe reactor shutdown and to prevent uncontrolled release of radioactive material if offsite power is interrupted or lost. Reactor shutdown is generally achieved by a reactor scram.

- Electrical power circuits should be isolated sufficiently to avoid electromagnetic interference with safety-related instrument and control functions.

- Technical specifications should be provided to ensure operability commensurate with power requirements for reactor shutdown and to prevent uncontrolled release of radioactive material.

8.2.3 Review Procedures

The staff will (1) compare the design bases of the normal electrical systems with the requirements discussed in other chapters of the SAR, including Chapters 4, 5, 7, 9, 10, 11, and 13, (2) confirm that the design characteristics and components of the normal electrical system could provide the projected range of services, (3) analyze possible malfunctions, accidents, and interruptions of electrical services to determine their effect on safe facility operation and on safe reactor shutdown, and (4) determine if proposed routing and redundancy, if applicable, of electrical circuits are sufficient to ensure safe reactor operation and shutdown and to avoid uncontrolled release of radioactive material.

8.2.4 Evaluation Findings

Chapter 8 should contain sufficient information to support the following types of conclusions, which are to be included in the staff's safety evaluation report:

- The design bases and functional characteristics of the normal electrical power systems for the facility have been reviewed, and the proposed electrical systems will provide all required services.

- The design of the normal electrical power system provides that in the event of the loss or interruption of electrical power the reactor can be safely shutdown.

- The design and location of the electrical wiring will prevent inadvertent electromagnetic interference between the electrical power service and safety-related instrument and control circuits.

- The design of normal electrical systems provides reasonable assurance that use or malfunction of electrical power systems and controls for experiments could not cause reactor damage or prevent safe reactor shutdown.

- The technical specifications, including testing and surveillance provisions, ensure that the normal electrical system will be operable.

8.3 Emergency Electrical Power Systems

As discussed above, normal electrical power systems should be designed for all normal reactor operations. These systems should allow safe reactor shutdown in the event of offsite power interruptions. Emergency electrical power systems will be required if SAR analyses show that assured power is required to maintain safe reactor shutdown (Chapter 4), to support operation of a required engineered safety feature (Chapters 6 and 13), or to protect the public from release of radioactive effluents (Chapters 11 and 13). For some reactor facilities, emergency electrical power also might be required to avoid damage to an experiment (Chapter 10). For all of these functions, monitoring or sensing channels may also be required to operate on emergency power. Emergency electrical power at a non-power reactor is defined as any temporary substitute for normal electrical service. Some non-power reactor facilities provide emergency electrical power for functions other than those noted above; these should be discussed. The staff review should focus on those uses required to ensure that the health and safety of the public are protected from such unsafe reactor conditions as loss of fuel integrity or uncontrolled release of radioactive material.

8.3.1 Areas of Review

Non-power reactors should be designed for passive reactor shutdown if normal electrical service is interrupted. Some non-power reactors may require emergency power for maintaining safe facility shutdown, for example, decay heat removal, and some non-power reactors may use emergency power to avoid interruption of their research facilities or utilization program. The areas of review for this section are the design bases derived from other chapters of the SAR and their implementation for the emergency electrical power systems at the specific facility. The NRC review will focus on the safety-related features of any emergency electrical power systems.

Other SAR chapters present the calculated response of reactor systems to interruption of offsite power and the potential consequences. This section should describe and discuss any emergency electrical systems designed to avoid fuel damage or the release of radioactive material to the environment. The review should also include events of lesser consequences and the emergency electrical power system design that mitigates them.

8.3.2 Acceptance Criteria

The acceptance criteria for the emergency electric power systems at non-power reactors are based on the following:

- The functional characteristics of the emergency power system should be commensurate with the design bases, which are derived from analyses presented in other chapters of the SAR. In general, the minimum requirement of an emergency electrical power system should be to ensure and maintain safe facility shutdown and to prevent uncontrolled release of radioactive material.

- The source of electrical power (generator, batteries, etc.) should be capable of supplying power for the duration required by the SAR analysis.

- The system should be designed for either automatic or manual startup and switch over.

- The emergency electrical power system should not interfere with or prevent safe facility shutdown.

- Malfunctions of the emergency electrical power system during reactor operation with normal electrical power should not interfere with normal reactor operation or prevent safe facility shutdown.

- Any non-safety-related uses of an emergency electrical power system should not interfere with performance of its safety-related functions.

- Technical specifications should be based on the accident analyses, include surveillance and testing, and provide reasonable assurance of emergency electrical power system operability. The discussions in the SAR should identify the minimum design requirements, the minimum equipment required, and the power and duration of operation required.

8.3.3 Review Procedures

The staff will (1) compare the design bases of the emergency electrical power system with the requirements for emergency electrical power presented in Chapters 4, 5, 7, 9, 10, 11, and 13, (2) compare the design and functional characteristics with the design bases to verify compatibility, (3) verify that an emergency electrical system is not required at those facilities not proposing one, and (4) consider the design features of the emergency electrical power system that help ensure availability, including the mechanisms of startup, source of generator fuel, routing of wiring, and methods of isolation from normal services.

8.3.4 Evaluation Findings

Chapter 8 should contain sufficient information to support the following types of conclusions, which are to be included in the staff's safety evaluation report:

- The design bases and functional characteristics of the emergency electrical power systems have been reviewed, and the proposed system is capable of providing the necessary range of safety-related services.

- The design and operating characteristics of the source of emergency electrical power are basic and reliable, ensuring availability if needed.

- The design of the emergency electrical power system will not interfere with safe facility shutdown or lead to reactor damage if the system malfunctions during normal reactor operation.

- The technical specifications, including surveillance and testing, provide reasonable assurance of necessary system operability and availability.

9 AUXILIARY SYSTEMS

9.1 Introduction

In this chapter the applicant should discuss the auxiliary systems at the reactor facility. Auxiliary systems are defined as systems not fully described in other chapters of the safety analysis report (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 function of the system, 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 ability to shutdown 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," and

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

The design, operation, and use of non-power reactors vary widely. The following typical auxiliary systems may be included in the SAR, as appropriate:

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

- 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.

- the control, storage, and processing systems for byproduct material produced and used 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 special 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,"

- 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."),

- 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, and

- 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 Chapter 9. The descriptions of some auxiliary systems may be better suited to other chapters in the SAR and should be referenced to those chapters in this section. The following sections identify some specific information that the applicant should provide in this chapter.

9.2 Heating, Ventilating, and Air Conditioning Systems

All used spaces in a facility may require HVAC systems to provide acceptable environments for personnel and equipment. 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 explain how airborne radioactive material from operations and experiments is limited in occupied areas to maintain radiation exposures below 10 CFR Part 20 requirements and the facility ALARA (as low as is reasonably achievable) program limits. 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 Chapter 9. There should be discussions of the interactions among airflow patterns in the reactor room, the air exhaust stacks, and effluent and continuous air monitors. If the HVAC systems also are designed to mitigate the consequences of accidents, the ESFs should be noted here 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 applicant should include sufficient information in Chapter 9 for 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 evaluated in Chapter 11.

9.3 Handling and Storage of Reactor Fuel

The applicant should discuss the life cycle of reactor fuel from the time it enters the jurisdiction of the licensee until it is released from such jurisdiction. For most non-power reactors this means from arrival onsite until shipment offsite. The discussions should describe 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 offsite. 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 applicant should describe the physical protection of fuel against theft or diversion in the facility physical security plan. This information is treated as proprietary or safeguards information.

The applicant should address handling, storage, and shipment of new and irradiated fuel. The applicant should discuss 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 include the design of handling tools, transfer casks and other radiation shields, storage facilities, methods of preparing fuel for shipping, and shipping methods. Descriptions of procedures and systems for irradiated fuel storage and handling should include radiation shielding, protection from physical damage, physical control, and sufficient cooling to prevent overheating and surface corrosion. Details of irradiated fuel cooling systems and methods may be described in Chapter 5 if they are integral to the reactor cooling 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 Analysis." Detailed discussions of radiological considerations for storing and handling fuel should be provided in Chapter 11 and should include the implementation of 10 CFR 73.6(b) concerning self-protection for irradiated high-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 may be found in American Nuclear Society Standard ANSI/ANS-15.19-1991, "Shipment and Receipt of Special Nuclear Material (SNM) by Research Reactor Facilities."

9.4 Fire Protection Systems and Programs

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. The applicant should discuss additional active and passive design features required by the reactor design characteristics. Further, the discussion should address the potential for releasing radioactive material from 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 combustibles.

The applicant should describe how the designs for the facility considered the potential release of radioactive materials from fires in the reactor room and other applicable spaces. The discussion should include the reactor and all facilities that store or use special nuclear material and other radioactive materials under the reactor license. It should include any effects of a fire that could affect a safe shutdown of the reactor. The objectives of the fire protection program should include

- preventing fires, including limiting combustibles,

- detecting, controlling, and extinguishing fires to limit consequences, and

- 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.

"Fire Protection Program Criteria for Research Reactors," ANSI/ANS 15.17 (1987) contains general information on fire protection.

9.5 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, the 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.

9.6 Byproduct, Source, and Special Nuclear Material Possession and Use

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, the 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 material 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 license. 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 greater than 31 days will pass before it is placed into the reactor, then the material should be placed 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 Appendix B).

The receipt, possession, or use of materials authorized by the reactor license may be 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 material is removed from the reactor pool. Others take a broad view with all materials produced by the reactor or authorized by the license allowed to be in a variety of 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 to 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 chapter laboratories under the reactor license in which reactor licensed material will be used. The discussions in Chapter 9 should address all five factors noted in Section 9.1 for any such auxiliary laboratories. The discussion 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. 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. Information on the administrative aspects of the use of materials in these areas should be addressed in Chapter 12, "Conduct of Operations."

9.7 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. Some of these reactors use heavy water (D_2O) as a moderator/reflector/coolant, with a need to prevent admixture of atmospheric water vapor and loss of the heavy water. Others employ a primary system that 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 noted in Section 9.1. 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 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.

9.8 Other Auxiliary Systems

As noted earlier, there could be a unique set of auxiliary systems 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 noted in Section 9.1, 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.
- Demonstrate that the auxiliary system could not prevent safe reactor shutdown.
- Provide discussion and the bases for any technical specifications related to the auxiliary system, including testing and surveillance.

9 AUXILIARY SYSTEMS

9.1 Introduction

This chapter provides review and acceptance criteria for information presented pertaining to auxiliary systems in the reactor facility. Auxiliary systems are defined as systems not fully described in other chapters of the safety analysis report (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. There are also auxiliary systems or subsystems that do not have a direct impact on protecting the reactor or the public from exposure to radiation. However, for all auxiliary systems at a non-power reactor, sufficient information should be provided so that the reviewer can understand the design and function of the system. Emphasis should be placed on those aspects of auxiliary systems that might affect the reactor, its safety features, and its safe shutdown, or contribute to the control of radioactivity and radiation exposures.

The design, operation, and use of non-power reactors vary widely, resulting in a wide variety of auxiliary systems. The capability of each auxiliary system to function as designed without compromising the safe operation or shutdown of the reactor facility under the range of operational conditions should be discussed. There also should be a discussion of any functions of auxiliary systems required during analyzed reactor accidents. The discussion the applicant should provide in this section of the SAR for each auxiliary system is listed in Section 9.1 of the format and content guide. The typical auxiliary systems discussed in this section are not intended as a complete list of auxiliary systems to be discussed in Chapter 9 of the SAR. The reviewer should be aware that the discussion of some auxiliary systems could be shared by more than one chapter. The following sections provide guidance for the reviewer to address the five items in Section 9.1 of the format and content guide for the example systems discussed in Sections 9.2 to 9.8 of the guide.

9.2 Heating, Ventilation, and Air Conditioning (HVAC) Systems

9.2.1 Areas of Review

At a non-power reactor, the HVAC systems are designed to provide conditioned air for an acceptable working environment for personnel and equipment. The areas of review for this section include HVAC operating characteristics for the full range of reactor operation. In many non-power reactors, the HVAC system is also designed to limit concentrations and prevent the uncontrolled release of airborne radioactive material to the unrestricted environment. Any operating modes or functions designed to mitigate the consequences of accidents should be discussed in Chapter 6, "Engineered Safety Features." Radiological exposures to airborne radioactive material that result from the full range of reactor operations should be analyzed in detail in Chapter 11, "Radiation Protection Program and Waste Management," where design bases for the full range of reactor operations of the HVAC system are developed.

The NRC will review

- discussion of the characteristics and functions of the HVAC system if no airborne radioactivity is present,

- discussion of all sources of radioactive materials that could become airborne during the full range of reactor operation, and how the HVAC system is designed to affect the distribution and concentration of those materials,

- features of the HVAC system designed to limit exposures of personnel to radiation in the restricted area as a result of the full range of reactor operation,

- features of the HVAC system and associated reactor building designs that prevent inadvertent or uncontrolled release of airborne radioactive material to areas outside the reactor room and to the unrestricted environment,

- modes of operation and features of the HVAC system designed to control (contain or confine) reactor facility atmospheres, including damper closure or flow diversion functions during the full range of reactor operation,

- features of the HVAC system that affect habitability and the working environment in the reactor facility for personnel and equipment, and

- applicable technical specifications and their bases, including testing and surveillance.

9.2.2 Acceptance Criteria

The acceptance criteria for the HVAC systems should be based on the following:

- The system design should ensure that temperature, relative humidity, and air exchange rate (ventilation) are within the design-basis limits for personnel and equipment.

- The system design should address all normal sources of airborne radioactive material and ensure that these sources are diluted, diverted, or filtered so that no occupational doses exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA (as low as is reasonably achievable) program.

- The design features should ensure air flow and relative pressure that prevent inadvertent diffusion or other uncontrolled release of airborne radioactive material from the reactor room.

- The design and operating features of the HVAC system should ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur.

- The analyses of operations of the HVAC systems should show that planned releases of airborne radioactive material to the unrestricted environment will not expose the public to doses that exceed the limits of 10 CFR Part 20 and are consistent with the facility ALARA program. The exposure analyses are detailed in Chapter 11 of the SAR.

- If design bases of the HVAC system include containment or confinement during the full range of reactor operation, the system design and analyses should show how this condition is ensured. If the function is used to mitigate accident scenarios as discussed in Chapter 13, "Accident Analysis," the function should be described in Chapter 6.

- Required technical specifications and their bases should ensure system operability.

9.2.3 Review Procedures

Using the five factors in Section 9.1 of the format and content guide, the staff will evaluate the submittal for all operations and functions of the HVAC systems during the full range of reactor operations. The design bases will be compared with requirements from other chapters of the SAR, especially Chapters 4, "Reactor Description"; 6; 7, "Instrumentation and Control Systems"; 11; and 13, "Accident Analysis." Reviewers will examine whether all acceptance criteria have been met by the HVAC designs for the full range of reactor operations.

9.2.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- A review of the design bases and functional and safety characteristics of the HVAC systems indicate that the proposed systems are adequate to control the release of airborne radioactive effluents during the full range of reactor operations in compliance with the regulations.

- All sources of radioactive material that could become airborne in the reactor room from the full range of reactor operations have been discussed. The analyses have demonstrated that the radioactive material is controlled by the HVAC system and could not inadvertently escape from the reactor room. The analyses show that the distributions and concentrations of the airborne radionuclides in the reactor facility are limited by operation of the HVAC system so that during the full range of facility operations no potential occupational exposures would exceed the design bases derived in Chapter 11.

- If the HVAC system includes a stack to exhaust facility air to the unrestricted environment, the height and flow rate are considered for the design-basis dose rates derived in Chapter 11 for the maximum exposed personnel in the unrestricted environment.

- The HVAC system is an integral part of a containment (confinement) system at the reactor facility. The design of the system and analysis of its operation ensure that the containment (confinement) system will function to limit normal airborne radioactive material to the extent analyzed in this chapter and Chapter 11. The potential radiation doses will not exceed the limits of 10 CFR Part 20 and are consistent with the facility ALARA program.

- Technical specifications, including testing and surveillance, are proposed that will provide reasonable assurance of necessary HVAC system operability for the full range of reactor operations.

9.3 Handling and Storage of Reactor Fuel

The fuel for a non-power reactor is the most important component bearing on the health and safety of the public and on the common security. Protecting the fuel from malfunction or failure is discussed in many chapters in the SAR.

9.3.1 Areas of review

The areas of review for Chapter 9 include the handling, protection, and storage of the fuel when it is not in the reactor core, both before insertion and after removal.

The NRC will review

- equipment, systems, methods, and administrative procedures for receipt of new fuel,

- methods for inspection and verification of new fuel to ensure procurement specifications,

- systems and methods for movement, physical control, and storage of new fuel within the facility,

- methods, analyses, and systems for secure storage of new and irradiated fuel that will prevent criticality (k_{eff} not to exceed 0.90) under all conditions of moderation during storage and movement (The use of criticality monitors will be reviewed, if applicable in accordance with 10 CFR 70.24.),

- tools, systems, and methods for inserting fuel into the reactor core and for removing fuel from the core (This discussion should include physical and administrative methods to ensure that fuel is handled only by authorized persons.),

- systems, components, and methods for radiation shielding and for protecting irradiated fuel from damage during removal from the core, movement within the reactor facility, and storage (Thermal and mechanical damage should be discussed. Details on the cooling of stored irradiated fuel should be found in Chapter 5, "Reactor Coolant Systems," if the system is integral to that function. Otherwise, it should be described in this Chapter.),

- systems, components, and methods used to prepare and ship fuel off site in accordance with applicable regulations (This function should also be discussed for facilities that expect to retain the fuel until reactor decommissioning.), and

- technical specifications that define controls on fuel during handling and storage, including testing and surveillance.

9.3.2 Acceptance Criteria

Acceptance criteria for fuel handling and storage should be based on the following:

- The design of all systems, components, and methods for handling, moving, or storing fuel outside of the reactor core should ensure with a high confidence level that the neutron multiplication, k_{eff}, will not exceed 0.90 under any possible conditions. (Existing usage with k_{eff} greater than 0.90 will be acceptable if the usage was previously reviewed and approved by NRC.) Neutron multiplication requirements for shipping containers will be determined by their specific licenses.

- All systems, components, and methods for handling, moving, or storing fuel, including insertion and removal from the reactor, should be designed to prevent mechanical damage that could significantly decrease integrity or release fission products. In the case of irradiated fuel outside the reactor core, these analyses should demonstrate how loss of cladding integrity from excess temperatures will be prevented.

- The design of all systems, components, and methods for handling, moving, or storing fuel should demonstrate that facility staff and the public are protected from radiation and radiation exposures do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program.

- All systems, components, and methods for handling, moving, or storing fuel should be designed to control special nuclear material to the extent required by applicable regulations, such as 10 CFR Part 73. The discussions related to diversion and theft of the fuel should be withheld from public disclosure and should be contained in the facility physical security plan.

- If 10 CFR 73.6(b), self-protection, applies to the storage of irradiated high-enriched uranium (HEU) fuel, the SAR should discuss measurement methods or other techniques to ensure compliance.

- The technical specifications should contain limitations on storage conditions necessary to ensure subcriticality, prevent thermal failure, and administratively and physically control the fuel (special nuclear material) because of its potential for fission and potential hazards as a radiation source.

9.3.3 Review Procedures

The staff will review the systems and methods used to handle and store new and irradiated fuel. The reviewer will compare the design bases with any requirements provided in this and other chapters of the SAR, such as Chapters 4, 11, and 13, and the requirements of the regulations. The reviewer will focus on the design features that maintain fuel cladding integrity, control radiation, and prevent criticality.

9.3.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The discussions of plans for receiving, inspecting, and documenting the arrival of new fuel give reasonable assurance that all special nuclear material will be accounted for and that the fuel will comply with procurement specifications.

- The SAR analyses show that fuel storage features will ensure that criticality cannot occur. Even under optimum neutron moderation and reflection conditions, the maximum neutron multiplication could not exceed 0.90 (for license renewals, the maximum neutron multiplication previously approved by NRC). Plans to implement the applicable requirements of 10 CFR 70.24 for criticality monitoring are acceptable, if applicable.

- Tools and procedures for inserting and removing fuel from the core are specially designed to avoid damaging a fuel element. Provisions for controlling access to fuel handling tools give reasonable assurance that only authorized persons will insert or remove fuel from the core.

- Methods for assessing irradiated fuel radioactivity and potential exposure rates are adequate to avoid overexposure of staff.

- Methods for shielding, cooling, and storing irradiated fuel give reasonable assurance that

  - Potential personnel doses will not exceed the regulatory limits of 10 CFR Part 20 and are consistent with the facility ALARA program.

  - Irradiated fuel can be cooled as necessary to avoid loss of integrity and corrosive deterioration during both moving and storage within the facility.

- Acceptable provisions have been described in the SAR to ensure compliance with 10 CFR 73.6(b), self-protection for HEU fuel, as applicable.

9.4 Fire Protection Systems and Programs

9.4.1 Areas of Review

The applicant should briefly address potential causes and consequences of fires at the facility. The reviewer should verify that fire protection plans and protective equipment used to limit the consequences of a fire are discussed, including defense in depth in the event of escalation of a fire. The SAR should list the objectives of the program, as well as discuss the organizations, methods, and equipment for attaining the objectives. All passive designs or protective barriers planned to limit fire consequences should be discussed. These discussions should include features of the facility that could affect a safe reactor shutdown or release radioactive material in the event of a continuing fire. The applicant should indicate the source of facility fire protection brigades and their training and briefly summarize the more detailed discussions of these personnel and offsite fire protection forces in the facility emergency plan. The areas of review will also include SAR discussions of compliance with local and national fire and building codes applicable to fire protection.

9.4.2 Acceptance Criteria

Acceptance criteria for the information on the fire protection plans are based on the following:

- The plan discusses the prevention of fires, including limiting the types and quantities of combustibles.
- Methods are available to detect, control, and extinguish fires.
- The facility is designed and protective systems exist to ensure a safe reactor shutdown and prevent the uncontrolled release of radioactive material if a fire occurs.

9.4.3 Review Procedures

The staff will evaluate the SAR discussions of potential fires; provisions for early detection, including times when the buildings are not occupied; methods for isolating, suppressing, and extinguishing fires; passive features designed into the facility to limit fire consequences; response organization training and availability to fight fires as detailed in the emergency plan; designs of reactor systems that can ensure safe reactor shutdown in the event of fire; and potential radiological consequences to the public, staff, and environment if fire fighting efforts are unsuccessful.

9.4.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The plans for preventing fires ensure that the facility meets the requirements of local and [illegible] fire and building codes.
- The systems designed to detect and combat fires at the facility can function as described and limit damage and consequences at any time.
- Personnel training programs as described in the facility emergency plan and in Chapter 12, "Conduct of Operations," provide reasonable assurance that training for fire protection is adequately planned.
- The potential radiological consequences of a fire do not prevent safe reactor shutdown, and any fire-related release of radioactive material from the facility to the unrestricted environment has been adequately addressed in the appropriate sections of the facility emergency plan.
- Any release of radioactive material as a result of fire would not cause radiation exposures that exceed the limits and guidelines of 10 CFR Part 20 and are consistent with the facility ALARA program.
- Acceptable technical specifications related to fire protection have been proposed and justified, as applicable. These technical specifications include acceptable requirements for testing and surveillance to ensure operability of fire detection and protection equipment.

9.5 Communications Systems

9.5.1 Areas of Review

The NRC will review in this section of the SAR methods for communication between all necessary locations during the full range of reactor operations and a summary of emergency communications, which are discussed in detail and evaluated by the staff in the review of the physical security and emergency plans. The discussion should show how two-way communication is provided between the reactor control room and other reactor facility locations.

9.5.2 Acceptance Criteria

The acceptance criteria for the information on communication systems should be based on the following:

- The system design should allow the reactor operator on duty to contact the supervisor on duty, health physics staff, and other technical specification required personnel at any time the reactor is operating.

- The communication system should allow the operator, or other designated staff member, to announce the existence of an emergency in all areas of the reactor site.

- The communication system should allow two-way communication between all operational areas, such as between the control room and the reactor fuel-loading location and the control room and reactor experiment halls.

9.5.3 Review Procedures

The SAR discussions will be reviewed for inclusion of the five items noted in Section 9.1 of the format and content guide to formulate conclusions about the adequacy of the communications systems.

9.5.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The facility communications systems are designed to provide two-way communication between the reactor control room and all other locations necessary for safe reactor operation.

- The communications systems allow the reactor operator on duty to communicate with the supervisor on duty and with health physics personnel.

- The communication systems allow a facility-wide announcement of an emergency.

- The communications systems provide for summoning emergency assistance from designated personnel, as discussed in detail in the physical security and emergency plans.

- All technical specifications related to the facility communications systems are acceptable, providing for minimum necessary communication.

9.6 Byproduct Material Possession and Use

9.6.1 Areas of Review

The operating license for a non-power reactor authorizes the possession and operation of the reactor and the possession of all radioactive material that is a byproduct of that operation. The license also will specify the spaces and areas within the site associated with reactor operations. Licenses granted under 10 CFR Part 50 also may authorize the possession of other specified byproduct and special nuclear material used by the reactor for research and development purposes. Byproduct materials may be used at the licensee facilities or shipped offsite to be used by others under a different license. If the materials will be used at the licensee facilities, either they should be transferred to another license (NRC or an Agreement State) or the 10 CFR Part 50 license should explicitly state which facilities and materials are covered, which should be described in the SAR as auxiliary reactor systems.

The NRC will review

- the types and quantities of radionuclides authorized,
- the rooms, spaces, equipment, and procedures to be used,
- the general types of uses, such as research and development, processing, or packaging for shipment,
- provisions for controlling and disposing of radioactive wastes, including special drains for liquids and chemicals, and air exhaust hoods for air-borne materials, with design bases derived in Chapter 11,
- provisions for radiation protection, including shielding materials and radiation survey methods, with design bases derived in Chapter 11,
- the relationship between these auxiliary facility designs and the physical security and emergency plans, and
- required technical specifications and their bases, including testing and surveillance.

9.6.2 Acceptance Criteria

The acceptance criteria for information on the auxiliary facilities to possess and use reactor-made radioactive byproduct material under the 10 CFR Part 50 license are based on the following:

- The design of spaces and equipment and procedures should ensure that no uncontrolled release of radioactive materials (solid, liquid, or airborne) can occur from the facilities.
- The design and procedures should ensure that personnel exposures to radiation, including ingestion or inhalation, do not exceed limiting values in 10 CFR Part 20 as verified in Chapter 11 and are consistent with the facility ALARA program as described in Chapter 11.
- The design and procedures should ensure compliance with all regulations subsumed within the 10 CFR Part 50 license, such as 10 CFR Parts 30 and 70.
- The operating procedures for auxiliary facilities ensure that only radioactive byproducts produced by the reactor are permitted, unless specifically authorized by conditions of the 10 CFR Part 50 license or an additional license.

- The facilities are addressed specifically in the emergency plan, physical security plan, and fire protection provisions, as applicable.

- The proposed technical specifications covering these auxiliary facilities ensure the protection of the health and safety of the public, reactor users, and the environment, and the control of licensed byproduct and special nuclear materials.

9.6.3 Review Procedures

The five items listed in Section 9.1 of the format and content guide will be evaluated by the reviewer for auxiliary systems and facilities that possess or use byproduct materials produced in the reactor or special nuclear material, as allowed by the 10 CFR Part 50 license.

The design bases for systems and procedures will be compared with the requirements developed in other chapters of the SAR, especially Chapters 11 and 12. The reviewer will evaluate the design features against experience with possession and use of radioactive materials at other facilities and laboratories. Compliance with the acceptance criteria will be evaluated.

An important aspect of the control and use of this material are found in operations and health physics procedures. While the review of the actual procedures by the reviewer is not necessary, the reviewer will review the basis for the procedures and the method for review and approval of facility procedures described in Chapter 12. In some cases, the reviewer may conduct an audit of selected procedures as part of the review. Normally, procedures are selectively reviewed by inspectors as part of the construction/startup inspection process or as part of the ongoing inspection program for existing facilities.

9.6.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- Auxiliary facilities and systems are designed for the possession and use of byproduct materials produced by the reactor and, if applicable, special nuclear material. The design bases include limits on potential personnel exposures that are in compliance with 10 CFR Part 20 and are consistent with the facility ALARA program, as described in Chapter 11.

- To ensure that radiation exposures are acceptably limited, the design features and license conditions specify upper limits on source strengths of radionuclides authorized for possession or use in the auxiliary facilities under the 10 CFR Part 50 license. The authorized spaces for use of material are described.

- Design features and procedures provide reasonable assurance that uncontrolled release of radioactive material to the unrestricted environment will not occur.

- Design features and procedures ensure that the use of the auxiliary facilities is covered by the emergency plan, physical security plan, and fire protection provisions, as applicable.

- Technical specifications are proposed that will ensure that possession and use of the auxiliary facilities will not endanger the health and safety of the public, users, or the environment.

9.7 Cover Gas Control in Closed Primary Coolant Systems

9.7.1 Areas of Review

Some non-power reactor designs include a reactor core tank as part of the primary coolant loop that is sealed against the atmosphere. Some of these reactors use heavy water (D_2O) that functions as a moderator, reflector, or coolant. The heavy water must be protected to prevent admixture of atmospheric water vapor and loss of the heavy water. Other non-power reactors have closed primary systems that operate with ordinary light water at or above atmospheric pressure. In these systems, radiolytic decomposition of the water leads to a hydrogen-oxygen mixture that could reach explosive concentration unless it is controlled or processed.

The NRC will review

- design bases for the closed systems, addressing the types of gases to be contained and controlled in them,

- systems for assessing and maintaining any required pressure differential with the external atmosphere,

- systems for assessing the required purity or concentrations of the contained gases,

- methods and systems for circulating, processing, decontaminating, recovering, and storing the contained gases,

- methods for assessing and recombining hydrogen and oxygen gases that could result from radiolysis of the coolant,

- analyses of the potential effect on reactor safety or operation if the characteristics of the gas mixture are changed, including type of majority gas and concentrations of minority gases, and

- any technical specifications and their bases, including testing and surveillance, required to ensure operability of the cover gas systems, if applicable.

9.7.2 Acceptance Criteria

The acceptance criteria for information on the auxiliary cover gas systems in non-power reactors with closed primary coolant systems are based on the following:

- The systems should be designed to perform the design bases functions.
- The system should be designed to ensure the control and detection of leaks so that no uncontrolled release of radioactive material could occur and safe reactor shutdown is not compromised by the system.
- Any necessary processing, recombining, or storing of the cover gases is designed to ensure the safety of the reactor and personnel.
- Technical specifications and their bases including surveillances are provided as required to ensure system operability.

9.7.3 Review Procedures

The five items listed in Section 9.1 of the format and content guide will be evaluated by the reviewer for auxiliary systems that provide and control cover gas for non-power reactors with closed primary coolant systems. The design will be compared with the design bases and with the acceptance criteria.

9.7.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The reactor is designed to operate with a closed primary coolant system, and the design of the cover gas control system described helps provide that function. The cover gas control system is designed to avoid uncontrolled release of radioactive material and to avoid interference with safe reactor operation or shutdown.
- The cover gas control system is designed to ensure that the required type of gas, the acceptable concentrations of constituents, and design-basis pressure are maintained.
- Processing, storing, and recombining of radiolytic gases, as applicable, have been acceptably incorporated into the design, as well as safe disposal of any radioactive waste.
- Technical specifications and their bases that are necessary to protect the health and safety of the public and safe reactor operation have been provided.

9.8 Other Auxiliary Systems

The auxiliary systems addressed above are typical examples of systems found at non-power reactor facilities. As noted in the introduction, many non-power reactors will have additional auxiliary systems and some will have facility-unique systems. Not all possible systems can be adequately addressed here. For other systems, the reviewer should apply the following review and evaluation approach.

9.8.1 Areas of Review

The five factors noted in Section 9.1 of the format and content guide should be addressed in the SAR.

9.8.2 Acceptance Criteria

The acceptance criteria for information on additional auxiliary systems are based on the following:

- The system design and functional description should ensure that it conforms with the design bases.
- The design, functions, and potential malfunctions of the auxiliary system should not cause accidents to the reactor or uncontrolled release of radioactivity.
- In the event radioactive material is released by the operation of an auxiliary system, potential radiation exposures should not exceed the limits of 10 CFR Part 20 and should be consistent with the guidelines of the facility ALARA program.
- No function or malfunction of the auxiliary system should interfere with or prevent safe shutdown of the reactor.
- The technical specifications and bases applicable to an auxiliary system should be provided.

9.8.3 Review Procedures

Review of the discussion in the SAR of additional auxiliary systems should compare the design and functional descriptions with the design bases. The reviewer will review the discussion and analyses of the functions and potential malfunctions with respect to safe reactor operation and shutdown, impact on reactor safety systems, and the potential of the auxiliary system to initiate or affect the uncontrolled release of radioactive material.

9.8.4 Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:

- The system has been designed to perform the functions required by the design bases.

- The design of the system considers functions and potential malfunctions that could affect reactor operations. No analyzed functions or malfunctions could initiate a reactor accident, prevent safe reactor shutdown, or initiate uncontrolled release of radioactive material.

- The technical specifications and their bases proposed in the SAR give reasonable assurance that the system will be operable, as required by the design bases.

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