

United States
Nuclear Regulatory Commission



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

Standard Review Plan and Acceptance Criteria

February 1996

NUREG - 1537 PART 2

Office of Nuclear Reactor Regulation
Division of Reactor Program Management



**Guidelines for Preparing and Reviewing Applications
for the Licensing of Non-Power Reactors**
Standard Review Plan and Acceptance Criteria

Office of Nuclear Reactor Regulation Division of Reactor Program Management

February 1996
NUREG-1537-PAR1

United States
Nuclear Regulatory Commission



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

Standard Review Plan and Acceptance Criteria

February 1996

NUREG-1537-PART-2

Office of Nuclear Reactor Regulation
Division of Reactor Program Management

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars; information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers, and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

Photograph on cover courtesy of General Atomics

ABSTRACT

NUREG-1537, Part 2 gives guidance on the conduct of licensing action reviews to NRC staff who review non-power reactor licensing applications. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

NUREG-1537, Part 2, has been reproduced
from the best available copy.

CONTENTS

	<i>Page</i>
ABSTRACT	iii
ABBREVIATIONS	ix
INTRODUCTION	xiii
Background	xiii
Document Structure	xvi
General Requirements	xviii
Contributors	xix
References	xx
 1 THE FACILITY	 1-1
1.1 Introduction	1-1
1.2 Summary and Conclusions on Principal Safety Considerations	1-2
1.3 General Description	1-5
1.4 Shared Facilities and Equipment	1-7
1.5 Comparison With Similar Facilities	1-8
1.6 Summary of Operations	1-10
1.7 Compliance With the Nuclear Waste Policy Act of 1982	1-12
1.8 Facility Modifications and History	1-13
Appendix 1.1 Introduction From NUREG-1312	
Appendix 1.2 DOE Letter to NRC Concerning the Nuclear Waste Policy Act of 1982	
 2 SITE CHARACTERISTICS	 2-1
2.1 Geography and Demography	2-1
2.2 Nearby Industrial, Transportation, and Military Facilities	2-3
2.3 Meteorology	2-4
2.4 Hydrology	2-6
2.5 Geology, Seismology, and Geotechnical Engineering	2-8
2.6 Bibliography	2-10
 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS	 3-1
3.1 Design Criteria	3-1
3.2 Meteorological Damage	3-3
3.3 Water Damage	3-4
3.4 Seismic Damage	3-5
3.5 Systems and Components	3-7
3.6 Bibliography	3-8

CONTENTS

	<i>Page</i>
4 REACTOR DESCRIPTION	4-1
4.1 Summary Description	4-1
4.2 Reactor Core	4-1
4.2.1 Reactor Fuel	4-2
4.2.2 Control Rods	4-5
4.2.3 Neutron Moderator and Reflector	4-7
4.2.4 Neutron Startup Source	4-9
4.2.5 Core Support Structure	4-12
4.3 Reactor Tank or Pool	4-14
4.4 Biological Shield	4-17
4.5 Nuclear Design	4-19
4.5.1 Normal Operating Conditions	4-19
4.5.2 Reactor Core Physics Parameters	4-22
4.5.3 Operating Limits	4-24
4.6 Thermal-Hydraulic Design	4-26
 5 REACTOR COOLANT SYSTEMS	 5-1
5.1 Summary Description	5-2
5.2 Primary Coolant System	5-2
5.3 Secondary Coolant System	5-7
5.4 Primary Coolant Cleanup System	5-10
5.5 Primary Coolant Makeup Water System	5-12
5.6 Nitrogen-16 Control System	5-14
5.7 Auxiliary Systems Using Primary Coolant	5-16
5.8 Reference	5-18
 6 ENGINEERED SAFETY FEATURES	 6-1
6.1 Summary Description	6-4
6.2 Detailed Descriptions	6-4
6.2.1 Confinement	6-4
6.2.2 Containment	6-7
6.2.3 Emergency Core Cooling System	6-11
6.3 References	6-13
 7 INSTRUMENTATION AND CONTROL SYSTEMS	 7-1
7.1 Summary Description	7-2
7.2 Design of Instrumentation and Control Systems	7-3
7.3 Reactor Control System	7-3
7.4 Reactor Protection System	7-9
7.5 Engineered Safety Features Actuation Systems	7-14
7.6 Control Console and Display Instruments	7-16
7.7 Radiation Monitoring Systems	7-20
7.8 References	7-22

	<i>Page</i>
8 ELECTRICAL POWER SYSTEMS	8-1
8.1 Normal Electrical Power Systems	8-1
8.2 Emergency Electrical Power Systems	8-3
9 AUXILIARY SYSTEMS	9-1
9.1 Heating, Ventilation, and Air Conditioning Systems	9-1
9.2 Handling and Storage of Reactor Fuel	9-4
9.3 Fire Protection Systems and Programs	9-8
9.4 Communication Systems	9-10
9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material	9-11
9.6 Cover Gas Control in Closed Primary Coolant Systems	9-14
9.7 Other Auxiliary Systems	9-16
10 EXPERIMENTAL FACILITIES AND UTILIZATION	10-1
10.1 Summary Description	10-2
10.2 Experimental Facilities	10-3
10.3 Experiment Review	10-8
10.4 References	10-11
11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT	11-1
11.1 Radiation Protection	11-2
11.1.1 Radiation Sources	11-2
11.1.2 Radiation Protection Program	11-5
11.1.3 ALARA Program	11-10
11.1.4 Radiation Monitoring and Surveying	11-11
11.1.5 Radiation Exposure Control and Dosimetry	11-14
11.1.6 Contamination Control	11-18
11.1.7 Environmental Monitoring	11-21
11.2 Radioactive Waste Management	11-23
11.2.1 Radioactive Waste Management Program	11-23
11.2.2 Radioactive Waste Controls	11-25
11.2.3 Release of Radioactive Waste	11-27
11.3 Bibliography	11-28
12 CONDUCT OF OPERATIONS	12-1
12.1 Organization	12-1
12.2 Review and Audit Activities	12-4
12.3 Procedures	12-6
12.4 Required Actions	12-7
12.5 Reports	12-9

CONTENTS

	<i>Page</i>
12.6 Records	12-10
12.7 Emergency Planning	12-10
12.8 Security Planning	12-11
12.9 Quality Assurance	12-11
12.10 Operator Training and Requalification	12-11
12.11 Startup Plan	12-14
12.12 Environmental Reports	12-16
12.13 References	12-17
Appendix 12.1 Environmental Considerations Regarding the Licensing of Research Reactors and Critical Facilities	
13 ACCIDENT ANALYSES	13-1
Bibliography	13-14
14 TECHNICAL SPECIFICATIONS	14-1
References	14-3
15 FINANCIAL QUALIFICATIONS	15-1
15.1 Financial Ability To Construct a Non-Power Reactor	15-1
15.2 Financial Ability To Operate a Non-Power Reactor	15-3
15.3 Financial Ability To Decommission the Facility	15-5
16 OTHER LICENSE CONSIDERATIONS	16-1
16.1 Prior Use of Reactor Components	16-1
16.2 Medical Use of Non-Power Reactors	16-3
17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS	17-1
17.1 Decommissioning	17-1
17.1.1 Preliminary Decommissioning Plan	17-1
17.1.2 Submittal of the Decommissioning Plan	17-2
17.1.3 Review of the Decommissioning Plan	17-3
17.2 Possession-Only License Amendment	17-5
17.2.1 Review of the Application for a Possession-Only License	17-5
Appendix 17.1 NRC Review of Decommissioning Plans for Non-Power Reactors	
18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS	18-1
Appendix 18.1 Standard Review Plan and Acceptance Criteria for HEU to LEU Conversion	

ABBREVIATIONS

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

ABBREVIATIONS

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

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

INTRODUCTION

Background

This document 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.

NRC has published several documents that give guidance that is applicable to 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)" to help commercial power plants in applying for light-water reactor (LWR) licenses. The staff revised RG 1.70 in 1972, 1975, and 1978. In 1975, 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 earlier Standard Review Plan (NUREG-75/087) and published the revision as NUREG-0800. In 1987, the staff revised NUREG-0800.

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 (NUREG-1537, Part 1) for non-power reactor license applicants and is issuing this companion 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 comprises 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 guide contains additional information on the classification of non-power reactors.

All reactors (power and non-power) 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 in other parts of Title 10 that deal with non-power reactors. The licensed thermal power levels of non-power reactors are several orders of magnitude lower than current power reactors. Therefore, the accumulated inventory of radioactive fission products in the fuel (in core) of non-power reactors is proportionally less and requires less stringent and less prescriptive measures to give equivalent protection to the health and safety of the public. Thus, even though many of the regulations of Title 10 apply to both power and non-power reactors, the regulations 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 it 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, operation, 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 research reactor was discussed in an Atomic Safety and Licensing Appeal Board (ASLAB) decision issued May 18, 1972, for the research reactor at Columbia University in New York City. ASLAB stated that "as a general proposition, the Appeal Board does not consider it desirable to use the standards of 10 CFR Part 20 for evaluating the effects of a postulated accident in a research reactor inasmuch as they are unduly restrictive for that purpose. The Appeal Board strongly recommends that specific standards for the evaluation of an accident situation in a research reactor be formulated." The staff has not found it necessary to conform to that recommendation to develop separate criteria for the evaluation of research reactor

accidents, since the majority of research reactors to date have been able to adopt 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 Appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and Appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results for research reactors are less than 5 rem whole body and less than 30 rem thyroid for occupational exposure, and less than 0.5 rem whole body and less than 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analyses that exceed the 10 CFR Part 20 dose limits discussed above.

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

The 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 shall demonstrate that the facility design and the safety conclusions of the previous SAR documents are unchanged.

This standard review plan covers a variety of site conditions and plant designs. Each section contains the necessary procedures and acceptance criteria for all areas of review pertinent to that section. However, not all of the guidance in this standard review plan may be applicable to every non-power reactor type licensed by NRC. There may be instances in which the applicant has not addressed a topic in the format and content guide because the applicant has made a determination that the guidance is not applicable to the particular reactor. The reviewer should be aware of the general non-power reactor types and the differences between the types. If it is not clear to the reviewer that specific guidance is not applicable to the reactor under review, the applicant may be asked why a particular issue is not addressed in the SAR. 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. Rationale for each decision should be documented in the appropriate section of the SAR.

Document Structure

Parts 1 and 2 of this document are complementary; titles and numbers of sections 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 the review, including a description of the systems, components, analyses, data, or other information that is part of the particular safety analysis section under review.
- ***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 comprise such specific criteria 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 this 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 with analyses. These solutions and approaches are presented in this form so that reviewers can take uniform positions for these issues in future reviews.

Although the technical positions in these documents represent solutions and approaches that are acceptable, those solutions and approaches should not be considered as the only solutions and approaches that are acceptable. However, applicants should recognize that, as in the case of regulatory guides, NRC staff spent substantial time and effort preparing the technical positions, and a corresponding amount of effort would probably be required to review and find acceptable 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 performed and is generally a description that the reviewer follows to verify that the applicable safety criteria have been met. The reviewer must document the results of the review in the staff's safety evaluation report by the following means:

- stating the applicable requirements or standards, with specific citation to the source of those requirements or standards
- summarizing the applicant's proposed method for satisfying the requirements or standards
- summarizing the staff's analysis of whether the applicant's proposal does indeed satisfy the requirements or standards

The documented analysis must be a sufficient basis for the evaluation findings which are discussed below.

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

Although not specifically discussed in every section of this standard review plan, each section of the staff's safety evaluation report 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 herein, and the bases for any deviations from this standard review plan.

Selected chapters end with a reference section or a bibliography, which gives full citations for the documents, standards, and other reports referred to in this standard review plan, and which may also list other useful material.

This standard review plan and the format and content guide 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 the format and content guide were prepared by staff who have 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 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.

NRC wrote these documents with three major objectives: (1) to discuss NRC requirements germane to each review topic, (2) to describe how the reviewer determines that 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 bottom of each revised page. The revision numbers and dates need not be the same for all sections because individual sections will be replaced with a newly revised section only as needed. A list of affected pages will indicate the revision numbers for the current sections. As necessary, the staff will make corresponding changes to the format and content guide using these methods.

General Requirements

Most operating licenses for non-power reactors are issued for a 20-year term. 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 a license renewal should submit an SAR that follows the standard format and content guide.

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

accident consequences. The MHA, which assumes an incredible failure that can lead to fuel cladding or to a fueled experiment containment breach, is used to bound credible accidents in the accident analysis.

The SAR is the basic document that gives NRC justification for licensing the facility and gives information for understanding the design bases for the 10 CFR 50.59 change process, for training reactor operators, for preparing reactor operator licensing examinations, and for preparing for NRC inspections. For these reasons and 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 licensees of power reactors, the NRC staff encourages non-power reactor licensees to maintain current SARs on file at NRC after initial licensing or license renewal by submitting replacement pages along with applications for license amendment and along with the annual report that summarizes changes made without prior NRC approval under 10 CFR 50.59.

Although 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. 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 standard letter to the licensee has been developed for this purpose.

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 guide for licensing are also applicable to 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 shall confirm that all safety issues have been addressed.

Contributors

This document was prepared by A. Adams, Jr., Senior Project Manager, Non-Power Reactors and Decommissioning Project Directorate, Division of Project

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

References

U.S. Nuclear Regulatory Commission, "Standard Format and Content for Applications for the Licensing of Non-Power Reactors," NUREG-1537, Part 1, February 1996.

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

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

1 THE FACILITY

Chapter 1 of the safety analysis report (SAR) is an overview or an 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

Areas of Review

In this very brief introduction to the applicant and the facility, areas of review should include the following:

- 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
- unique design features, such as a pressurized primary coolant system or unique fuel design, which would be notable for a non-power reactor licensed by the U.S. Nuclear Regulatory Commission (NRC)

Acceptance Criteria

Acceptance criteria for the information in this section should include the following:

- The purpose of the SAR should be clearly stated.
- The applicant should be identified.
- The location, purpose, and use of the facility should be briefly described.
- The basic characteristics of the facility that affect licensing considerations should be briefly discussed.
- The design or location features included to address basic safety concerns should be outlined.

- Any unique safety design features of the facility different from previously licensed non-power reactor facilities should be highlighted.

Review Procedures

The reviewer should confirm that the applicant submitted all information requested in the format and content guide.

The reviewer should confirm that the introduction contains 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.

Evaluation Findings

The NRC does not write evaluation findings for the introduction of the SAR. Section 1.1 of the staff's safety evaluation report serves as an introduction to the NRC report and has a standard format. Section 1 of the "Safety Evaluation Report Related to the Renewal of the Facility License for the Research Reactor at the Dow Chemical Company," NUREG-1312, April 1989, an example of the standard format, is reproduced here as Appendix 1.1. The statements should be appropriately modified for an initial application for construction and operation. In the introduction to the safety evaluation report, the staff identifies 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.

1.2 Summary and Conclusions on Principal Safety Considerations

Areas of Review

The reviewer should ensure that the SAR discusses all possibilities for radiological exposure to the public that could result from operation of the facility. In this section, the applicant 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.

Areas of review should include the following:

- safety criteria proposed by the applicant
- principal safety considerations of the facility design
- potential radiological consequences of operation and the method of providing protection
- description of safety of unique design features
- discussion of accidents

Acceptance Criteria

The acceptance criteria for the information on principal safety considerations include the following:

- Sufficient design features should be included to protect the health and safety of the public.
- No exposures from normal operation should exceed the requirements of Part 20 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 20) and the guidance of the facility program for keeping exposures as low as is reasonably achievable (ALARA).
- Accidents should be briefly discussed.
- All modes of operation and events that could lead to significant radiological releases and exposure of the public should be discussed.

Review Procedures

The reviewer should confirm that the applicant submitted all information requested in the format and content guide. The reviewer should consider the stated criteria to ensure safety and to evaluate their application to the reactor facility design. The summary discussions and descriptions should 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-hydraulic 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 substitute for the detailed analysis in the SAR; they briefly summarize some of the information in the SAR. The reviewer should examine the detailed discussions as part of the review of other chapters of the SAR.

Evaluation Findings

NRC does not write specific evaluation findings for this section of the SRP. This section of the staff's safety evaluation report contains the summary and conclusions of principal safety considerations as determined by the NRC staff. These conclusions are summarized from the reviewer's analysis of the complete SAR 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 sought 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 an initial application. The conclusions 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 are adequately planned, and safe operation of the facility can reasonably be expected.
- (2) The management organization of the applicant is adequate to maintain the facility, ensure safe operation of the facility, and conduct research activities so that there is no significant radiological risk to the employees or the public.
- (3) The applicant has considered the expected consequences of several postulated accidents and has emphasized 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 Appendices (*for research reactors licensed before January 1, 1994*), or 10 CFR 20.1001 through 20.2402 and Appendices (*for research reactors licensed on or after January 1, 1994*) or 10 CFR Part 100 (*for test reactors*) for doses in unrestricted areas.
- (4) 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.
- (5) 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.

- (6) The financial data demonstrate that the applicant has reasonable access to sufficient revenues to cover (construction) operating costs and eventually to decommission the reactor facility.
- (7) The program for physically protecting the facility and its special nuclear materials complies with the requirements of 10 CFR Part 73.
- (8) The procedures for training its reactor operators and the plan for operator requalification are adequate; they give reasonable assurance the reactor will be operated competently.
- (9) The emergency plan provides reasonable assurance that the applicant is prepared to assess and respond to emergency events.

1.3 General Description

Areas of Review

In this very brief description of the facility, the reviewer should ensure that the applicant's overview of the facility design shows 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 applicant should present a more detailed description in later chapters of the SAR. The applicant should include drawings, tables, and photographs as necessary.

Areas of review should include the following:

- 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

- the design features of the radioactive waste management system or provisions and radiation protection

Acceptance Criteria

Acceptance criteria for the general description of the facility include the following:

- The applicant should briefly describe
 - geographical location of the reactor facility
 - principal characteristics of the site
 - principal design criteria, operating characteristics, and safety systems
 - any engineered safety features
 - instrumentation, control, and electrical systems
 - reactor coolant and other auxiliary systems
 - radioactive waste management provisions or system and radiation protection
 - experimental facilities and capabilities
- The applicant should indicate the general arrangement of major structures and equipment with plan and elevation drawings
- The applicant should briefly identify safety features likely to be of special interest.
- The applicant should highlight unusual characteristics of the site, the containment building, novel designs of the reactor, or unique experimental facilities.

The reviewer should examine full facility descriptions and analysis found in to other sections of the SAR and should evaluate them there.

Review Procedures

The reviewer should confirm that the applicant submitted all information requested in the format and content guide.

Evaluation Findings

NRC does not write evaluation findings on this section of the SAR.

1.4 Shared Facilities and Equipment

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 solely 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. Areas of review for this section should include brief descriptions and discussions of facilities and equipment shared between the facility described in this SAR and others. Additional guidance on what constitutes a shared facility is discussed in the format and content guide.

The reviewer should verify that this section summarized the safety implications and relationships between 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 contain detailed descriptions and safety implications of such shared equipment or functions.

Acceptance Criteria

The acceptance criteria for the information on shared facilities and equipment include the following:

- The non-power reactor facility should be designed to accommodate all uses or malfunctions of the shared facilities without degradation of the non-power reactor safety features.
- The non-power reactor should be 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 these two foregoing criteria are met.

Review Procedures

The reviewer should confirm that all facilities or equipment shared by the non-power reactor have been discussed in the SAR. The reviewer should verify that

the applicant discussed 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 discussion in the SAR of the effect of the shared facilities on the safety of the subject facility. The reviewer may need to review discussions and analyses in other sections of the SAR.

Evaluation Findings

This section of 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 staff's safety evaluation report:

- The shared facilities are clearly and completely listed and the other users are identified. The applicant has shown that a malfunction or a loss of function of these shared facilities would not affect the operation of the non-power reactor, nor would it damage the non-power reactor or its capability to be safely shut down.
- Either normal operation or a 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 Analyses," and are found to be acceptable.

1.5 Comparison With Similar Facilities

Areas of Review

Since the early 1940s, several hundred non-power reactors have been built in the United States, and many more were built in other countries. The first few such reactors established the safety considerations and principles for the non-power reactors that followed.

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 power excursion reactor test (SPERT)
- Chicago Pile #5 or Argonne research reactor (CP-5)

Applicants are expected to use pertinent information from these and other reactors in their design, and the reviewer 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 could include the following:

- Chapter 4, "Reactor Description," and Chapter 13 for the bases of NRC's acceptance of fuel performance [e.g., SPERT, the Oak Ridge Research Reactor (ORRR), the system for nuclear auxiliary power (SNAP), the General Atomics reactor for training and isotope production (TRIGA), the 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, the 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 (reactor shutdown) systems [e.g., BSF, TRIGA, Omega West Reactor (OWR), MTR, and CP-5]
- other specific license conditions acceptable to NRC of other facilities that demonstrate acceptable technical performance (previously licensed facilities with similar thermal power level, similar fuel type, and similar siting considerations)

Acceptance Criteria

The acceptance criteria for the comparison of this facility with similar facilities include the following:

- The comparisons should show that the proposed facility would not exceed the safety envelope of the similar facilities.
- There should be reasonable assurance that radiological exposures of the public would not exceed the regulations and the guidelines of the proposed facility ALARA program.

Review Procedures

The reviewer should confirm that the characteristics of any facilities compared with the proposed facility are similar and relevant. The reviewer should verify that

the operating history of licensed facilities cited by the applicant demonstrates consistently safe operation, use, and protection of the public.

Evaluation Findings

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

- The applicant has 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 the staff.
- The applicant's design does not differ in any substantive way from similar facilities that have been found acceptable to NRC, and should be expected to perform in a similar manner when constructed to that design.
- The applicant has used test data from similar reactor facilities in designing components. The applicant cited the actual facilities with the components. These data provide assurance that the facility can operate safely as designed.

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

1.6 Summary of Operations

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 should 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.

Areas of review should include 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
- assumed releases of radioactive effluents to the unrestricted environment

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

Acceptance Criteria

The acceptance criteria for the applicant's summary of operations include the following:

- 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 applicant should demonstrate that the proposed reactor operation was conservatively considered in the design and safety analyses.
- The proposed operations for license renewal should be consistent with the assumptions in later chapters of the SAR.

Review Procedures

Although 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 should compare them with similar facilities for initial licensing, or with 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 applicant can operate the facility as specified in the SAR. If there are limitations on operation of the reactor, the reviewer should verify that they are represented as license conditions.

Evaluation Findings

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

- The proposed operating conditions and schedules are consistent with those of similar facilities that have been found acceptable to the staff, and with the design features of the facility. The proposed operations are consistent with relevant assumptions in later chapters of the SAR, in which 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 Compliance With the Nuclear Waste Policy Act of 1982

Area of Review

The reviewer should confirm that the applicant has contracted with the U.S. Department of Energy (DOE) to dispose of high-level waste and irradiated (spent) fuel.

Acceptance Criteria

Acceptance criteria for the information on compliance with the Nuclear Waste Policy Act of 1982 should include the following:

- The applicant should have submitted a summary of the contract with DOE to dispose of high-level waste and irradiated (spent fuel).
- The applicant should have indicated where a copy of the contract letter can be found in the SAR.

Review Procedures

The reviewer should compare the content of the SAR with that suggested in this section of the format and content guide. If necessary, the appropriate DOE representatives could confirm the contract.

Evaluation Findings

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

- *(If the applicant is a university or government agency)* The Nuclear Waste Policy Act of 1982, Section 302(b)(1)(B), states that 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 NRC (H. Denton) by letter dated May 3, 1983,* that it had 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 (*insert name of applicant*) has entered into such a contract with DOE, the applicable requirements of the *Nuclear Waste Policy Act* of 1982 have been satisfied.

- (*If the applicant is a corporation*) Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982 states that 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. (*Insert name of applicant*) has entered into a contract with DOE [Contract (*insert contract number*) for the ultimate disposal of the fuel in the (*insert name of applicant's reactor*).] Because (*insert name of applicant*) has entered into such a contract with DOE, the applicable requirements of the *Nuclear Waste Policy Act* of 1982 have been satisfied.

1.8 Facility Modifications and History

Areas of Review

If the SAR describes a new facility, the reviewer need only examine 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, the reviewer should confirm the history of the facility, including amendments to the license, with dates and purposes. The reviewer should also evaluate any significant changes in the previous SAR conditions not requiring NRC approval under 10 CFR 50.59 or other regulations. This discussion should include any significant facility modifications and their effect on operations and releases of radioactive effluents to unrestricted areas.

*The DOE letter is reproduced at the end of this chapter as Appendix 1.2.

Acceptance Criterion

The acceptance criterion for the information on facility modifications and history is the following: The applicant should submit a complete facility history.

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.

Evaluation Findings

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

- 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.
- *(If the application is for license renewal or if the applicant has previous nuclear experience)* The information contains a short summary of the history of the facility.

Appendix 1.1

Introduction from NUREG-1312

1 INTRODUCTION

By letter (with supporting documentation) dated November 14, 1986, as supplemented on June 2, 1987, August 14, 1987, April 29, 1988, and January 10, 1989, the Dow Chemical Company (Dow/licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC/staff) a timely application for a 20-year renewal of the Class 104c Facility Operating License R-108 (NRC Docket No. 50-264) and an increase in operating power level, from the existing 100 kilowatts thermal [kW(t)] to 300 kW(t), for its TRIGA Mark I research reactor facility. The research reactor facility is located in the 1602 Building on the grounds of the Michigan Division of the Dow Chemical Company in Midland, Michigan. The licensee currently is permitted to operate the Dow TRIGA Research Reactor (DTRR) within the conditions authorized in past amendments in accordance with Title 10 of the Code of Federal Regulations (10 CFR), Section 2.109, until NRC action on the renewal request is completed.

The staff's review, with respect to issuing a renewal operating license to Dow, was based on the information contained in the renewal application and supporting supplements plus responses to requests for additional information. The renewal application included financial statements, the Safety Analysis Report, an Environmental Report, the Operator Requalification Program, the Emergency Plan, and Technical Specifications. This material is available for review at the Commission's Public Document Room located at 2120 L Street, NW, Washington, DC 20555. The approved Physical Security Plan is protected from public disclosure under 10 CFR 2.790.

The purpose of this Safety Evaluation Report (SER) is to summarize the results of the safety review of the DTRR and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewal of the license for operation of the DTRR at thermal power levels up to and including 300 kW. The facility was reviewed against the requirements of 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides; and appropriate accepted industry standards [American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series]. Because there are no specific accident-related regulations for research reactors, the staff has compared calculated dose values with related standards in 10 CFR Part 20, the standards for protection against radiation, both for employees and the public.

This SER was prepared by Alexander Adams, Jr., Project Manager, Division of Reactor Projects III, IV, V, and Special Projects, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Major contributors to the technical review were the Project Manager and R. E. Carter, C. Cooper, and R. Carpenter of the Idaho National Engineering Laboratory under contract to the NRC.

Appendix 1.2

DOE Letter to NRC Concerning the Nuclear Waste Policy Act of 1982



Department of Energy
Washington, D.C. 20585

3 MAY 1983

Mr. Harold Denton
Director, Office of Nuclear Reactor
Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20055

Dear Mr. Denton:

This will serve to clarify whether owners of research reactors will be required to sign a nuclear waste disposal contract under the Nuclear Waste Policy Act of 1982 (Public Law 97-425, "The Act").


Section 302(b)(1)(B) of the Act provides that the Nuclear Regulatory Commission (NRC), as it deems necessary or appropriate may require as a precondition to the issuance or renewal of a license under Section 103 or 104 of the Atomic Energy Act of 1954 (42 U.S.C. 2133, 2134) that the applicant shall have entered into an agreement for the disposal of high-level radioactive waste (HLW) and spent nuclear fuel (SNF) that may result from the use of such license. Section 104 of the Atomic Energy Act relates generally to utilization and production facilities, as those terms are defined in the Act (42 U.S.C. 2014(v), (cc), used for research and development purposes. We understand that the NRC has written to all Section 104 licensees, suggesting that they enter into negotiations with the Department for nuclear waste disposal services in view of Section 302(b)(1)(B) of the Act.

Generally, universities or other Government agencies operating research reactors have existing agreements with the Department whereby DOE provides, through fuel assistance contracts, the funds to purchase the fuel. In such cases, the Department retains title to that fuel and is obligated to take the SNF and/or HLW generated by these reactors for storage or reprocessing at no cost to the research reactor organizations.

Accordingly, the Department of Energy has determined that owners of research reactors who are university or Government entities and have entered into such contracts with DOE, do not need further nuclear waste disposal contracts with the Department as specified under the provisions of Section 302(B)(2) of the Nuclear Waste Policy Act of 1982.

If you would like additional information or clarification on this matter please contact me at (202) 252-6850.

Sincerely,


Robert L. Morgan
Director
Nuclear Waste Policy Act
Project Office

2 SITE CHARACTERISTICS

This chapter provides guidance for reviewing and evaluating Chapter 2 of the applicant's SAR in which the applicant discusses the geological, seismological, hydrological, meteorological, geographic and demographic characteristics of the site and vicinity, in conjunction with present and projected population distributions, industrial facilities and land use, and site activities and controls. The site characteristics should be described in sufficient detail to verify input to design and analyses presented in other chapters of the SAR, e.g., Chapter 3, "Design of Structures, Systems, and Components"; Chapter 11, "Radiation Protection Program and Waste Management"; and Chapter 13, "Accident Analyses." In each case, the reviewer determines how much emphasis to place on the various topics covered by this chapter of the SAR. The reviewer's judgment on the areas to be given attention during the review should be based on an examination of the information presented, the similarity of the information to that recently reviewed for other reactors, and whether any special site characteristics or reactor design or operating features raise questions of safety significance. In 10 CFR 100.10, the staff gives factors to consider in selecting a site and related reactor design for test reactors.

2.1 Geography and Demography

Areas of Review

The reviewer should ascertain that reactor location is identified by latitude and longitude and by the Universal Transverse Mercator (UTM) coordinate system as found on U.S. Geological Survey (USGS) topographical map with respect to State, county, or other political subdivisions and distributions of population; and with respect to prominent natural and manmade features of the area that could affect the safety of reactor operations at that site, and the health and safety of the public. The characteristics of the operations, site, and urban boundaries and rural zones up to 8 kilometers from the reactor should be given. The current and projected population distributions within 1, 2, 4, 6, and 8 kilometers of the reactor location should be included and temporary or seasonal populations located in dormitories or classrooms on a college campus should be given, if applicable.

Acceptance Criteria

The acceptance criteria for the information on geography and demography include the following:

- The geographical and demographic descriptions of the facility and its location are sufficiently accurate and detailed to provide the necessary bases for analyses presented in other chapters of the SAR.

- No geographic or demographic characteristics of the facility site could render the site unsuitable for operation of the proposed reactor. For example, information presented demonstrates that the property and political jurisdictions are sufficiently defined and sufficiently stable that there is reasonable assurance that the applicant can exercise necessary radiological control throughout the facility boundaries.

In addition, land use in the area of the facility is sufficiently stable or well enough planned that likely potential radiological risks to the public can be analyzed and evaluated with reasonable confidence. Existing and projected land-use information includes population distribution, densities, and other relevant characteristics, so that projected doses can be shown not to exceed the applicable limits.

Review Procedures

The information in this section of the SAR forms the basis for evaluations performed in other chapters. Therefore, the reviewer should ascertain that sufficient site-related information supports the subsequent analyses of issues related to the distribution of population around the proposed reactor.

As part of this review, the reviewer should check the exclusion area distances against distances used in analyses presented in Chapters 11 and 13 of the SAR. The map provided should be scaled to check distances specified in the SAR and to determine the distance-direction relationships to area boundaries, roads, railways, waterways, prevailing winds, and other significant features of the area.

A visit to the site under review permits a better understanding of the physical characteristics of the site and its relationship to the surrounding area. It permits the reviewer to gather information, in addition to that supplied in the SAR, which is useful in confirming SAR analyses.

The site should be visited after the initial review of the complete SAR, and after requests for additional information are developed and sent to the applicant.

Evaluation Findings

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

- The information is sufficiently detailed to provide an accurate description of the geography surrounding the reactor facility.

- The demographic information is sufficient to allow accurate assessments of the potential radiological impact on the public resulting from the siting and operation of the proposed reactor.
- There is reasonable assurance that no geographic or demographic features render the site unsuitable for operation of the proposed reactor.

2.2 Nearby Industrial, Transportation, and Military Facilities

Areas of Review

The reviewer should evaluate the reactor site and its vicinity for location and separation distances from existing and planned industrial, military, and transportation facilities and routes. Such facilities and routes include air, ground, and water traffic, pipelines, and fixed manufacturing, processing, and storage facilities. The reviewer should focus on facilities, activities, and materials that may reasonably be expected to be present during the projected lifetime of the non-power reactor. The purpose of this review is to evaluate the information concerning the presence and magnitude of potential hazards to the reactor due to local manmade facilities.

Acceptance Criteria

The acceptance criteria for the information on nearby industrial, transportation, and military facilities include the following:

- The information presents a complete and current overview of facilities, activities, and materials located in the vicinity of the reactor site.
- The information is complete enough to support evaluations of potential risks posed by these facilities to the safe operation and shutdown of the reactor during its projected lifetime.
- The analyses show that none of the expected manmade facilities could cause damage or other hazards to the reactor sufficient to pose undue radiological risks to the operating staff, the public, or the environment. Consequences of such events are analyzed in or are shown to be bounded by accidents considered in Chapter 13 of the SAR.

Review Procedures

The reviewer should confirm that any hazards to the reactor facility posed by normal operation and potential malfunctions and accidents at the nearby manmade

stationary facilities and those related to transportation have been described and analyzed to the extent necessary to evaluate the potential radiological risks to the facility staff, the public, and the environment.

Evaluation Findings

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

- The applicant discusses all nearby manmade facilities and activities that could pose a hazard to reactor operations. There is reasonable assurance that normal operations of such facilities would not affect reactor operations.
- The analyses in Chapter 13 of potential malfunctions or accidents at nearby manmade facilities and consideration of normal activities at those facilities show that safe reactor shutdown would not be prevented, and no undue radiological risk to the public, the environment, or the operating staff is predicted. The potential consequences of these events at nearby facilities are considered or bounded by applicable accidents analyzed in Chapter 13 of the SAR.

On the basis of these considerations, the reviewer should be able to conclude that operations and potential accidents at nearby manmade facilities would not pose sufficient risk to the reactor to render the site unsuitable for construction and operation of the reactor facility, as designed.

2.3 Meteorology

Areas of Review

The reviewer should evaluate information presented by the applicant on documented historical averages and extremes of climatic conditions and regional meteorological phenomena that could affect the designed safety features and siting of the research reactor to determine that the applicant covers the following areas:

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

- historical seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes, waterspouts, thunderstorms, lightning, and hail
- historical and predicted meteorological conditions used as design and operating bases for the reactor facility including:
 - the maximum snow and ice load that the roofs of safety-related structures must be capable of withstanding during reactor operation
 - the maximum wind speed that safety-related structures must be capable of withstanding during reactor operation
 - severe wind loads, e.g., tornado strength including translational speed, rotational speed, and the maximum pressure differential with the projected time interval
- the local (site) meteorology in terms of air flow, temperature, atmospheric water vapor, precipitation, fog, atmospheric stability, and air quality

Acceptance Criteria

The acceptance criteria for the information on meteorology include the following:

- The information regarding the general climate of the region and the local meteorological descriptions of the site area is sufficiently documented so that meteorological impacts on reactor safety and operation can be reliably predicted.
- Historical summaries of local meteorological data based on available onsite measurements and National Weather Service station summaries or summaries from other nearby sources are presented.
- The information on meteorology, and local weather conditions is sufficient to support dispersion analyses for postulated airborne releases. The analyses should support realistic dispersion estimates of normal releases for Chapter 11 analyses and conservative dispersion estimates of projected releases for Chapter 13 analysis of accidental releases at locations of maximum projected radiological dose and other points of interest within a radius of 8 kilometers.
- The information is sufficient to provide design bases for the reactor facility to safely withstand weather extremes predicted to occur during the lifetime of the reactor. The reactor design bases provide reasonable assurance that

the most severe meteorological event predicted would not cause uncontrolled release of radioactive material leading to doses in the unrestricted area that exceed applicable limits.

Review Procedures

The reviewer should verify that sufficient documented and referenced historical information is provided to support the necessary analyses of meteorological effects at the reactor site. These data should address both short-term conditions applicable to accidental releases of radioactive material, and long-term averages applicable to releases during normal reactor operation. The reviewer should also verify that the predicted frequencies of recurrence and intensities of severe weather conditions are documented.

Evaluation Findings

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

- The meteorological history and projections for the reactor site have been prepared in an acceptable form. These projections have been factored into the choice of facility location and design sufficiently to provide assurance that no weather-related event is likely to cause damage to the reactor facility during its lifetime that could release uncontrolled radioactive material to the unrestricted area.
- The meteorological information is sufficient to support analyses applicable to and commensurate with the risks of the dispersion of airborne releases of radioactive material in the unrestricted environment at the site. The methods and assumptions are applied to releases from both normal reactor operations and postulated accidents at the reactor facility.

On the basis of these considerations, the reviewer should be able to conclude that the information provided shows that no weather-related events of credible frequency or consequences at the site render it unsuitable for operation of the reactor facility, as designed.

2.4 Hydrology

Areas of Review

The reviewer should verify that the information in this section of the applicant's SAR describes and discusses all features of the site that could lead to flooding or

other water-induced damage at the site. The information should cover the possible hydrologic events, their causes, historic and predicted frequencies, and potential consequences to the reactor facility. The water table should be located, and the potential for radioactive contamination of ground and surface waters should be discussed.

Acceptance Criteria

The acceptance criteria for the information on hydrology includes the following:

- The facility is located and designed to withstand credible hydrologic events. Locations of particular concern include a flood plain, downriver of a dam, and close to the seashore and sea level.
- Potential events at the site that could cause nearby hydrologic consequences are shown not to present significant risk to the facility.
- Facility design bases are derived sufficiently from predicted hydrologic events that there is reasonable assurance that such events would not preclude safe operation and shutdown of the reactor.
- The reactor facility design bases contain provisions to mitigate or prevent uncontrolled release of radioactive material in the event of a predicted hydrologic occurrence. Potential consequences of such an event are considered or bounded by accidents analyzed in Chapter 13 of the SAR.
- Facility design bases consider leakage or loss of primary coolant to ground water, neutron activation of ground water, and deposition of released airborne radioactive material in surface water.

Review Procedures

The reviewer should verify that the site has been selected with due consideration of potential hydrologic events and consequences, including any that could be initiated by either local or distant seismic disturbances. In addition, the reviewer should ascertain the design bases incorporated into the facility design to address predicted hydrologic events, accidental release or leakage of primary coolant, and radioactive contamination of ground or surface waters.

Evaluation Findings

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

- The applicant considered hydrologic events of credible frequency and consequence in selecting the facility site. The site is not located where catastrophic hydrologic events are credible.
- The applicant considered credible hydrologic events in developing the design bases for the facility, to mitigate or avoid significant damage so that safe operation and shutdown of the reactor would not be precluded by a hydrologic event.
- The applicant selected combinations of site characteristics and facility design bases to provide reasonable assurance that uncontrolled release of radioactive material in the event of a credible hydrologic occurrence would be bounded by accidents analyzed in Chapter 13 of the SAR.
- The facility design bases give reasonable assurance that contamination of ground and surface waters at the site from inadvertent release or leakage of primary coolant, neutron activation, or airborne releases would not exceed applicable limits of 10 CFR Part 20.

On the basis of these considerations, the reviewer should be able to conclude that no credible predicted hydrologic event or condition would render the site unsuitable for operation or safe shutdown of the reactor, as designed.

2.5 Geology, Seismology, and Geotechnical Engineering

Areas of Review

The reviewer should evaluate the information on the geologic structures and features underlying and in the region surrounding the facility site, and the history and predicted potential for seismic activities that could impact reactor safety to determine that the required extent and detail of the information presented is commensurate with the potential consequences to the reactor and to the public, the environment, and the facility staff.

The information on potential seismic effect should be in a form suitable for developing design bases in Chapter 3 for structures, systems, and components.

Acceptance Criteria

The acceptance criteria for the information presented on geology, seismology, and geotechnical engineering include the following:

- The geologic features underlying and in the region surrounding the reactor site are sufficient to provide the stable support required for reactor structures absent any nearby earthquakes.
- The geologic features at the site contain no known faults that could be reactivated by nearby seismic activity.
- The history of seismic activity at the site does not indicate a high probability of a catastrophic earthquake at the site during the projected reactor lifetime.
- Likely seismic activity affecting the site is sufficiently characterized to support development of applicable design criteria for reactor structures.

Review Procedures

The reviewer should confirm that the information presented has been obtained from sources of adequate credibility and is consistent with other available data, such as data from the USGS or in the final safety analysis report (FSAR) of a nearby nuclear power plant. The reviewer should be reasonably assured that the seismic characteristics of the site are considered in the design bases of structures, systems, and other facility features discussed in Chapter 3.

Evaluation Findings

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

- Information on the geologic features and the potential seismic activity at the site has been provided in sufficient detail and in a form to be integrated acceptably into design bases for structures, systems, and operating characteristics of the reactor.
- Information in the SAR indicates that damaging seismic activity at the reactor site during its projected lifetime is very unlikely. Furthermore, if seismic activity were to occur, any radiologic consequences are bounded or analyzed in Chapter 13 of the SAR.
- The SAR shows that there is no significant likelihood that the public would be subject to undue radiological risk following seismic activity; therefore, the site is not unsuitable for the proposed reactor because of potential earthquakes.

2.6 Bibliography

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

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

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

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

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

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

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

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

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter gives guidance for reviewing and evaluating the principal architectural and engineering design criteria for the structures, systems, and components that have been identified by the analyses in this and other chapters of the SAR to ensure reactor facility safety and protection of the public. The bases of some design features may be developed and presented in other chapters of the SAR (e.g., the confinement or the containment, air exhaust stack, and environmental requirements for safety systems) and need only be referenced in this chapter.

3.1 Design Criteria

Areas of Review

Areas of review should include the criteria for the design and construction of the structures, systems, and components that are required to ensure the following:

- safe reactor operation
- safe reactor shutdown and continued safe conditions
- response to anticipated transients
- response to potential accidents analyzed in Chapter 13, "Accident Analyses," of the SAR
- control of radioactive material discussed in Chapter 11, "Radiation Protection Program and Waste Management," of the SAR

Acceptance Criteria

The acceptance criteria for the information on design criteria include the following:

- Design criteria should be specified for each structure, system, and component that is assumed in the SAR to perform an operational or safety function.
- Design criteria should include references to applicable up-to-date standards, guides, and codes. They should be stipulated for those features discussed in the format and content guide for this section, as outlined below:

- design for the complete range of normal reactor operating conditions
- design to cope with anticipated transients and potential accidents
- design redundancy to protect against unsafe conditions in case of single failures of reactor protective and safety systems
- design to facilitate inspection, testing, and maintenance
- design to limit the likelihood and consequences of fires, explosions, and other potential manmade conditions
- quality standards commensurate with the safety function and potential risks
- design bases to withstand or mitigate wind, water, and seismic damage to reactor systems and structures
- analysis of function, reliability, and maintainability of systems and components

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

Review Procedures

The reviewer should compare the specified design criteria with the proposed and analyzed normal reactor operation, response to anticipated transients, and consequences of accident conditions applicable to the appropriate structures, systems, and components assumed to function in each chapter of the SAR.

Evaluation Findings

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

- The design criteria are based on applicable standards, guides, codes, and criteria and provide reasonable assurance that the facility structures, systems, and components can be built and will function as designed and

required by the analyses in the SAR. The design criteria provide reasonable assurance that the public will be protected from radiological risks resulting from operation of the reactor facility.

3.2 Meteorological Damage

Areas of Review

Areas of review should include the design and design bases for all structures, systems, and components that could be affected by wind and other meteorological conditions (e.g., snow and ice) as discussed in Chapter 2, "Site Characteristics," of the SAR. The reviewer should consider wind loads, pressure (including back pressure) effects of potential wind conditions, snow and ice loads, and the facility design features to cope with these conditions.

Acceptance Criteria

The acceptance criteria for the information on meteorological damage include the following:

- The design criteria and designs should provide reasonable assurance that structures, systems, and components would continue to perform their safety functions as specified in the SAR under potential meteorological damage conditions.
- For the design the applicant should use local building codes, standards, or other applicable criteria, at a minimum, to ensure that significant meteorological damage at the facility site is very unlikely.

Review Procedures

The reviewer should examine the description of the site meteorology to ensure that all structures, systems, and components that could suffer meteorological damage are considered in this section of the SAR. This description should include historical data and predictions as specified in Chapter 2 and in the format and content guide for this section. The reviewer should assess the design criteria and the potential for meteorological damage and compare them with local applicable architectural and building codes for similar structures. The reviewer should compare design specifications for structures, systems, and components with the functional requirements and capability to retain function throughout the predicted meteorological conditions.

Evaluation Findings

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

- The design to protect against meteorological damage provides reasonable assurance that the facility structures, systems, and components will perform the safety functions discussed in the SAR, including the capability to maintain safe reactor operation, to effect and maintain safe reactor shutdown conditions, and to protect the health and safety of the public from radioactive materials and radiation exposure.

3.3 Water Damage

Areas of Review

Areas of review should include the design and design bases for all structures, systems, and components that could be affected by predicted hydrological conditions at the site. This should include (1) the impact on structures resulting from the force or submergence of flooding, (2) the impact on systems resulting from instrumentation and control electrical or mechanical malfunction due to water, and (3) the impact on equipment, such as fans, motors, and valves, resulting from degradation of the electromechanical function due to water.

Acceptance Criteria

The acceptance criteria for the information on water damage include the following:

- The design criteria and designs should provide reasonable assurance that structures, systems, and components would continue to perform required safety functions under water damage conditions.
- For the design the applicant should use local building codes, as applicable, to help ensure that water damage to structures, systems, and components at the facility site would not cause unsafe reactor operation, would not prevent safe reactor shutdown, and would not cause or allow uncontrolled release of radioactive material.

Review Procedures

The reviewer should examine the site description to ensure that all safety-related structures, systems, and components with the potential for hydrological (water) damage are considered in this SAR section. The review should include

hydrological historical data and predictions as specified in the format and content guide for this section. For any such structure, system, or component, the reviewer should ensure that the design bases are planned to address the consequences and are described in detail in appropriate chapters of the SAR.

Evaluation Findings

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

- The design bases to protect against potential hydrological (water) damage provide reasonable assurance that the facility structures, systems, and components will perform the functions necessary to allow any required reactor operation to continue safely, to allow safe reactor shutdown, and to protect the health and safety of the public from radioactive materials and radiation exposure.

3.4 Seismic Damage

Areas of Review

Areas of review should include the designs and design bases of structures, systems, and components that are required to maintain function in case of a seismic event at the facility site.

Acceptance Criteria

The acceptance criteria for the information on seismic damage include the following:

- The reactor facility design should provide reasonable assurance that the reactor can be shut down and maintained in a safe condition.
- The seismic design should be consistent with local building codes to provide assurance that significant damage to the facility and associated safety functions is unlikely.
- The applicant should demonstrate that all potential consequences from a seismic event are within the acceptable limits considered or bounded in the accident analyses of Chapter 13 to ensure that conditions due to a seismic event will not pose significant risk to the health and safety of the public.

- Surveillance to verify design functions of associated systems, including applicable instrumentation and controls, should be specified in the technical specifications, and other appropriate SAR chapters should be referenced for details. For example, if a seismically induced scram is a required instrumentation and control protective system, the applicant should propose and justify surveillance of this reactor trip function.

Review Procedures

The reviewer should examine the site description and historical data to ensure that appropriate seismic inputs have been considered in the analysis of the structures, systems, and components discussed in the SAR. For any structure, system, or component damaged, the SAR should contain analyses that show the extent to which potential seismic damage impairs the safety function of the structure, system, or component. The evaluation of seismic damage should be coordinated with the Chapter 13 accident analyses of seismic events or should be shown to be bounded by other accidents considered in Chapter 13.

Acceptable analysis criteria are established in the section on geology and seismology in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.7.

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

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

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

Evaluation Findings

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

- The design to protect against seismic damage provides reasonable assurance that the facility structures, systems, and components will perform the necessary safety functions described and analyzed in the SAR.
- The design to protect against seismic damage provides reasonable assurance that the consequences of credible seismic events at the facility are considered (or bounded) by the results of the Chapter 13 accident analyses, ensuring acceptable protection of the public health and safety.
- The surveillance activities proposed in the technical specifications provide reasonable assurance that the safety-related functions of the structures, systems, and components that are required to respond to or mitigate the consequences of seismic damage to the facility will be maintained.

3.5 Systems and Components

Areas of Review

Areas of review should include the design bases for the electromechanical systems and components that are required to function and are described in detail in this or other SAR sections.

Acceptance Criteria

The acceptance criteria for the information on systems and components include the following:

- The design criteria should include consideration of the conditions required of the electromechanical systems and components to ensure safe reactor operation, including response to transient and potential accident conditions analyzed in the SAR. (Examples of conditions that are important for the electromechanical systems and components are dynamic and static loads, number of cycles, vibration, wear, friction, strength of materials, and effects of the operating environment, including radiation and temperature.)
- Comparisons with similar applicable facility designs may be included (e.g., a reactor of similar design that has operated through its licensed life cycle

and whose electromechanical systems and components have functioned as designed).

Review Procedures

The reviewer should review this and other applicable SAR sections to verify that the electromechanical systems and components that are required to ensure safe reactor conditions are considered and their operating conditions are analyzed to ensure function. The design bases of applicable technical specifications that ensure operability should be evaluated.

Evaluation Findings

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

- The design bases of the electromechanical systems and components give reasonable assurance that the facility systems and components will function as designed to ensure safe operation and safe shutdown of the reactor.
- The surveillance activities proposed in the technical specifications acceptably ensure that the safety-related functions of the electromechanical systems and components will be operable and the health and safety of the public will be protected.

3.6 Bibliography

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

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

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

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

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

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

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

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

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

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

4 REACTOR DESCRIPTION

This chapter gives guidance for evaluating the description in the SAR of the reactor and how it functions as well as the design features for ensuring that the reactor can be safely operated and shut down from any operating condition or accident assumed in the safety analysis. Information in this chapter of the SAR should provide the design bases for many systems and functions discussed in other chapters of the SAR and for many technical specifications. The systems that should be discussed in this chapter of the SAR include the reactor core, reactor tank, and biological shield. The nuclear design of the reactor and the way systems work together are also addressed. In this chapter the applicant should explain how the design and proper operation of a non-power reactor make accidents extremely unlikely. This chapter of the SAR along with the analysis in Chapter 13, "Accident Analyses," should demonstrate that even the consequences of the design-basis accident would not cause unacceptable risk to the health and safety of the public.

4.1 Summary Description

This section of the SAR should contain a general overview of the reactor design and important characteristics of operation. The reviewer need not make any specific review findings for this section. The detailed discussions, evaluations, and analyses should appear in the following sections of the SAR.

This section should contain a brief discussion of the principal safety considerations in selecting the reactor type and the way the facility design principles achieve the principal safety considerations. Included should be summaries for the items requested in this section of the format and content guide and descriptive text, summary tables, drawings, and schematic diagrams.

4.2 Reactor Core

This section of the SAR should contain the design information on all components of the reactor core. The information should be presented in diagrams, drawings, tables of specifications, and text and analysis sufficient to give a clear understanding of the core components and how they constitute a functional non-power reactor that could be operated and shut down safely. Because radiation is one of the essential products from a non-power reactor, a principal design objective is to safely obtain the highest neutron flux densities in experimental facilities.

By reviewing this section, the reviewer gains an overview of the reactor core design and assurance that the SAR describes a complete, operable non-power

43

reactor core. Subsequent sections should contain a description and analysis of the specifications, operating characteristics, and safety features of the reactor components. Although cooling systems and incore experimental facilities should be discussed in Chapters 5, "Reactor Coolant Systems," and 10, "Experimental Facilities and Utilization," of the SAR, respectively, relevant information should also be presented or referenced in this chapter. The information in the following sections should address these systems and components:

- reactor fuel
- control rods
- neutron moderator and reflector
- neutron startup source
- core support structures

The information in the SAR for each core component and system should include the following:

- design bases
- system or component description, including drawings, schematics, and specifications of principal components, including materials
- operational analyses and safety considerations
- instrumentation and control features not fully described in Chapter 7, "Instrumentation and Control Systems," of the SAR and reference to Chapter 7
- technical specifications requirements and their bases, including testing and surveillance, or a reference to Chapter 14, "Technical Specifications"

4.2.1 Reactor Fuel

Areas of Review

With very few exceptions, the fuel used in licensed non-power reactors has been designed and tested under a broad generic development program. Therefore, the information in the SAR should include a reference to the fuel development program and the operational and limiting characteristics of the specific fuel used in the reactor.

The design basis for non-power reactor fuel should be the maintenance of fuel integrity under any conditions assumed in the safety analysis. Loss of integrity is defined as the escape of any fission products from the primary barrier, usually

cladding or encapsulation. The reviewer should be able to conclude that the applicant has included all information necessary to establish the limiting characteristics beyond which fuel integrity could be lost.

Within the context of the factors listed in Section 4.2 of this review plan, the information on and analyses of fuel should include the information requested in this section of the format and content guide. Sufficient information and analyses should support the limits for operational conditions. These limits should be selected to ensure the integrity of the fuel elements and their cladding. Analyses in this section of the SAR should address mechanical forces and stresses, corrosion and erosion of cladding, hydraulic forces, thermal changes and temperature gradients, and internal pressures from fission products and the production of fission gas. The analyses should also address radiation effects, including the maximum fission densities and fission rates that the fuel is designed to accommodate. Results from these analyses should form part of the design bases for other sections of the SAR, for the reactor safety limits, and for other fuel-related technical specifications.

Acceptance Criteria

The acceptance criteria for the information on reactor fuel include the following:

- The design bases for the fuel should be clearly presented, and the design considerations and functional description should ensure that fuel conforms with the bases. Maintaining fuel integrity should be the most important design objective.
- The chemical, physical, and metallurgical characteristics of the fuel constituents should be chosen for compatibility with each other and the anticipated environment.
- Fuel enrichment should be consistent with the requirements of 10 CFR 50.64.
- The fuel design should take into account characteristics that could limit fuel integrity, such as heat capacity and conductivity; melting, softening, and blistering temperatures; corrosion and erosion caused by coolant; physical stresses from mechanical or hydraulic forces (internal pressures and Bernoulli forces); fuel burnup; radiation damage to the fuel and the fuel cladding or containment; and retention of fission products.
- The fuel design should include the nuclear features of the reactor core, such as structural materials with small neutron absorption cross-sections and minimum impurities, neutron reflectors, and burnable poisons, if used.

- The discussion of the fuel should include a summary of the fuel development and qualification program.
- The applicant should propose technical specifications as discussed in Chapter 14 of the format and content guide to ensure that the fuel meets the safety-related design requirements. The applicant should justify the proposed technical specifications in this section of the SAR.

Review Procedures

The reviewer should confirm that the information on the reactor fuel includes a description of the required characteristics. The safety-related parameters should become design bases for the reactor operating characteristics in other sections of this chapter, especially Section 4.6 on the thermal-hydraulic design of the core.

Evaluation Findings

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

- The applicant has described in detail the fuel elements to be used in the reactor. The discussion includes the design limits of the fuel elements and clearly gives the technological and safety-related bases for these limits.
- The applicant has discussed the constituents, materials, components, and fabrication specifications for the fuel elements. Compliance with these specifications for all fuel acquisitions will ensure uniform characteristics and compliance with design bases and safety-related requirements.
- The applicant has referred to the fuel development program under which all fuel characteristics and parameters that are important to the safe operation of the reactor were investigated. The design limits are clearly identified for use in design bases to support technical specifications.
- Information on the design and development program for this fuel offers reasonable assurance that the fabricated fuel can function safely in the reactor without adversely affecting the health and safety of the public.

4.2.2 Control Rods

Areas of Review

The control rods in a non-power reactor are designed to change reactivity by changing the amount of neutron absorber (or fuel) in or near the reactor core. Depending on their function, control rods can be designated as regulating, safety, shim, or transient rods. To scram the reactor, the negative reactivity of the control rods is usually added passively and quickly when the rods drop into the core, although gravity can be assisted by spring action. In the case of control rods fabricated completely of fuel, the rods fall out of the bottom of the core. Because the control rods serve a dual function (control and safety), control and safety systems for non-power reactors are usually not completely separable. In non-power reactors, a scram does not challenge the safety of the reactor or cause any undue strain on any systems or components associated with the reactor.

The areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on control rods include the following:

- The control rods, blades, followers (if used), and support systems should be designed conservatively to withstand all anticipated stresses and challenges from mechanical, hydraulic, and thermal forces and the effects of their chemical and radiation environment.
- The control rods should be sufficient in number and reactivity worth to comply with the "single stuck rod" criterion; that is, it should be possible to shut down the reactor and comply with the requirement of minimum shutdown margin with the highest worth scrammable control rod stuck out of the core. The control rods should also be sufficient to control the reactor in all designed operating modes and to shut down the reactor safely from any operational condition. The design bases for redundancy and diversity should ensure these functions.
- The control rods should be designed for rapid, fail-safe shutdown of the reactor from any operating condition. The discussion should address conditions under which normal electrical power is lost.
- The control rods should be designed so that scrambling them does not challenge their integrity or operation or the integrity or operation of other reactor systems.

- The control rod design should ensure that positioning is reproducible and that a readout of positions is available for all reactor operating conditions.
- The drive and control systems for each control rod should be independent from other rods to prevent a malfunction in one from affecting insertion or withdrawal of any other.
- The drive speeds and scram times of the control rods should be consistent with reactor kinetics requirements considering mechanical friction, hydraulic resistance, and the electrical or magnetic system
- The control rods should allow replacement and inspection, as required by operational requirements and the technical specifications.
- Technical specifications should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects and proposes limiting conditions for operations and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the design bases for the control rods define all essential characteristics and that the applicant has addressed them completely.

Evaluation Findings

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

- The applicant has described the control and safety rod systems for the reactor and included a discussion of the design bases, which are derived from the planned operational characteristics of the reactor. All functional and safety-related design bases can be achieved by the control rod designs.
- The applicant has included information on the materials, components, and fabrication specifications of the control rod systems. These descriptions offer reasonable assurance that the control rods conform with the design bases and can control and shut down the reactor safely from any operating condition.
- The staff has evaluated the information on scram design for the control rods and compared it with designs at other non-power reactors having

similar operating characteristics. Reasonable assurance exists that the scram features designed for this reactor will perform as necessary to ensure fuel integrity and to protect the health and safety of the public.

- (For pulsing reactors) The design and functional description of the transient rod system offer reasonable assurance that pulses will be reproducible and can be limited to values that maintain fuel integrity as determined by the thermal-hydraulic analyses
- The control rod design includes reactivity worths that can control the excess reactivity planned for the reactor, including ensuring an acceptable shutdown reactivity and margin, as defined and specified in the technical specifications.
- Changes in reactivity caused by control rod dynamic characteristics are acceptable. The staff evaluations included maximum scram times and maximum rates of insertion of positive reactivity for normal and ramp insertions caused by system malfunctions.
- The applicant has justified appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods and included them in the technical specifications.

4.2.3 Neutron Moderator and Reflector

Areas of Review

In this section of the SAR, the applicant should describe moderators and reflectors designed into the reactor core and their special features. The cores of most non-power reactors consist of metallic fuel elements immersed in moderator and surrounded by either a liquid or solid neutron reflector. The solid reflectors are chosen primarily for favorable nuclear properties and physical characteristics. In some pool-type reactors (e.g., TRIGA), the fuel elements contain some of the core neutron moderator and reflector material. Section 4.2.1 of the SAR should contain a description of the relationship of all moderators to the core. For most non-power reactors, the water neutron moderator and reflector also function as the coolant, as discussed in Chapter 5. Buildup of contaminating radioactive material in the moderator or coolant and reflector during reactor operation should be discussed in Chapter 11, "Radiation Protection Program and Waste Management," of the SAR.

Areas of review should include the following:

- geometry

- materials
- compatibility with the operational environment
- structural designs
- response to radiation heating and damage
- capability to be moved and replaced, if necessary.

Nuclear characteristics should be discussed in Section 4.5 of the SAR.

Acceptance Criteria

The acceptance criteria for the information on neutron moderators and reflectors include the following:

- The non-nuclear design bases such as reflector encapsulations should be clearly presented, and the nuclear bases should be briefly summarized. Non-nuclear design considerations should ensure that the moderator and reflector can provide the necessary nuclear functions.
- The design should ensure that the moderator and reflector are compatible with their chemical, thermal, mechanical, and radiation environments. The design specifications should include cladding, if necessary, to avoid direct contact with water or to control the escape of gases. If cladding used to avoid direct contact with reactor coolant should fail, the applicant should show that the reactor can continue to be operated safely until the cladding is repaired or replaced or should shut the reactor down until the cladding is repaired or replaced.
- The design should allow for dimensional changes from radiation damage and thermal expansion to avoid malfunctions of the moderator or reflector.
- The design should include experimental facilities that are an integral part of the reflector. If the facilities malfunction, the reflector components should neither damage other reactor core components nor prevent safe reactor shutdown.
- The design should provide for removal and/or replacement of solid moderator or reflector components and systems, if required by operational considerations.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects, and proposes limiting conditions for operations and surveillance requirements. The proposed technical specifications should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the information on the neutron moderator and reflector completely describes the required systems. The bases for the nuclear characteristics should appear in Section 4.5 of the SAR.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The moderator and reflector are integral constituents of a reactor core; the staff's evaluation of the nuclear features appears in Section 4.5. The designs take into account interactions between the moderator or reflector and the reactor environment. Reasonable assurance exists that degradation rates of the moderator or reflector will not affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the unrestricted environment.
- Graphite moderators or reflectors are clad in aluminum (*or state cladding material*) if they are located in an environment where coolant infiltration could cause changes in neutron scattering and absorption, thereby changing core reactivity. Reasonable assurance exists that leakage will not occur. In the unlikely event coolant infiltration occurs, the applicant has shown that this infiltration will not interfere with safe reactor operation or prevent safe reactor shutdown.
- The moderator or reflector is composed of chemically inert materials incorporated into a sound structure that can retain size and shape and support all projected physical forces and weights. Therefore, no unplanned changes to the moderator or reflector would occur that would interfere with safe reactor operation or prevent safe reactor shutdown.
- The applicant has justified appropriate design limits, limiting conditions for operation, and surveillance requirements for the moderator and reflector and included them in the technical specifications.

4.2.4 Neutron Startup Source

Areas of Review

Each nuclear reactor should contain a neutron startup source that ensures the presence of neutrons during all changes in reactivity. This is especially important

when starting the reactor from a shutdown condition. Therefore, the reviewer should evaluate the function and reliability of the source system.

Areas of review should include the following:

- type of nuclear reaction
- energy spectra of neutrons
- source strength
- interaction of the source and holder, while in use, with the chemical, thermal, and radiation environment
- design features that ensure the function, integrity, and availability of the source
- technical specifications

Acceptance Criteria

Acceptance criteria for the information on the neutron startup source include the following:

- The source and source holder should be constructed of materials that will withstand the environment in the reactor core and during storage, if applicable, with no significant degradation.
- The type of neutron-emitting reaction in the source should be comparable to that at other licensed reactors, or test data should be presented in this section of the SAR to justify use of the source.
- The natural radioactive decay rate of the source should be slow enough to prevent a significant decay over 24 hours or between reactor operations.
- The design should allow easy replacement of the source and its holder and a source check or calibration.
- Neutron and gamma radiations from the reactor during normal operation should not cause heating, fissioning, or radiation damage to the source materials or the holder.

- If the source is regenerated by reactor operation, the design and analyses should demonstrate its capability to function as a reliable neutron startup source in the reactor environment.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the information on the neutron startup source and its holder includes a complete description of the components and functions. In conjunction with Chapter 7 of the SAR, the information should demonstrate the minimum source characteristics that will produce the required output signals on the startup instrumentation.

Evaluation Findings

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

- The design of the neutron startup source is of a type (i.e., neutron-emitting reaction) that has been used reliably in similar reactors licensed by NRC (or *the design has been fully described and analyzed*). The staff concludes this type of source is acceptable for this reactor.
- The source will not degrade in the radiation environment during reactor operation. Either the levels of external radiation are not significant or the source will be retracted while the reactor is at high power to limit the exposure.
- Because of the source holder design and fabrication, reactor neutron absorption is low and radiation damage is negligible in the environment of use. When radiation heating occurs, the holder temperature does not increase significantly above the ambient water temperature.
- The source strength produces an acceptable count rate on the reactor startup instrumentation and allows for a monitored startup of the reactor under all operating conditions.

- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the source and included them in the technical specifications.
- The source and holder design operate safely and reliably.

4.2.5 Core Support Structure

Areas of Review

All reactor core components must be secured firmly and accurately because the capability to maintain a controlled chain reaction depends on the relative positions of the components. Controlling reactor operations safely and reliably depends on the capability to locate components and reproduce responses of instrument and control systems, including nuclear detectors and control rods. Predictable fuel integrity depends on stable and reproducible fuel components and coolant flow patterns. Most fixed non-power reactor cores are supported from below. Some are suspended from above, and may be movable. Generally, the control rods of non-power reactors are suspended from a superstructure, which allows gravity to rapidly change core reactivity to shut down the reactor.

Areas of review include the design of the core support structure, including a demonstration that the design loads and forces are conservative compared with all expected loads and hydraulic forces and that relative positions of components can be maintained within tolerances.

Additional areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

Acceptance criteria for the information on the core support structure include the following:

- The design should show that the core support structure will conservatively hold the weight of all core-related components with and without the buoyant forces of the water in the tank or pool.
- The design should show that the core support structure will conservatively withstand all hydraulic forces from anticipated coolant flow with negligible deflection or motion.
- The methods by which core components (individual fuel elements, reflector pieces, control rods, experimental facilities, and coolant systems) are

attached to the core support structure should be considered in the design. The information should include tolerances for motion and reproducible positioning. These tolerances should ensure that variations will not cause reactivity design bases, coolant design bases, safety limits, or limiting conditions for operation in the technical specifications to be exceeded.

- The effect of the local environment on the material of the core support structure should be considered in the design. The impact of radiation damage, mechanical stresses, chemical compatibility with the coolant and core components, and reactivity effects should not degrade the performance of the supports sufficiently to impede safe reactor operation for the design life of the reactor.
- The design should show that stresses or forces from reactor components other than the core could not cause malfunctions, interfere with safe reactor operation or shutdown, or cause other core-related components to malfunction.
- The design for a movable core should contain features that ensure safe and reliable operation. This includes position tolerances to ensure safe and reliable reactor operation within all design limits including reactivity and cooling capability. The description should include the interlocks that keep the reactor core from moving while the reactor is critical or while forced cooling is required, if applicable. The design should show how the reactor is shut down if unwanted motion occurs.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the design bases define a complete core support system.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The applicant has described the support system for the reactor core, including the design bases, which are derived from the planned operational

characteristics of the reactor and the core design. All functional and safety-related design bases can be achieved by the design

- The core support structure contains grid plates that accurately position and align the fuel elements. This arrangement ensures a stable and reproducible reactivity. Hydraulic forces from coolant flow will not cause fuel elements to move or bow.
- The core support structure includes acceptable guides and supports for other essential core components, such as control rods, nuclear detectors, neutron reflectors, and incore experimental facilities.
- The core support structure provides sufficient coolant flow to conform with the design criteria and to prevent loss of fuel integrity from overheating.
- The core support structure is composed of materials shown to be resistant to radiation damage, coolant erosion and corrosion, thermal softening or yielding, and excessive neutron absorption.
- The core support structure is designed to ensure a stable and reproducible core configuration for all anticipated conditions (e.g., scrams, coolant flow change, and core motion) through the reactor life cycle.
- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the core support structure and included them in the technical specifications.

4.3 Reactor Tank or Pool

Areas of Review

The tank or pool (hereinafter referred to as "the tank") of most licensed non-power reactors is an essential part of the primary coolant system, ensuring sufficient coolant. The tank may also provide some support for components and systems mounted to the core supports, beam ports, and other experimental facilities.

The areas of review are the design bases of the tank and the design details needed to achieve those bases. The information that the applicant should submit for review is discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the reactor tank include the following:

- The tank dimensions should include thickness and structural supports, and fabrication methods should be discussed. The tank should be conservatively designed to withstand all mechanical and hydraulic forces and stresses to which it could be subjected during its lifetime.
- The construction materials and tank treatment should resist chemical interaction with the coolant and be chemically compatible with other reactor components in the primary coolant system.
- The dimensions of the tank, the materials used to fabricate the tank, and the position of the reactor core should help avoid radiation damage to the tank for its projected lifetime.
- The construction materials and tank treatment should be appropriate for preventing corrosion in inaccessible locations on the tank exterior.
- A plan should be in place to assess irradiation of and chemical damage to the tank materials. Remedies for damage or a replacement plan should be discussed.
- All penetrations and attachments to the tank below the coolant level, especially those below the top of the core, should be designed to avoid malfunction and loss of coolant.
- The shape and volume of the tank should be designed so that the coolant in it augments solid radiation shields to protect personnel and components from undue radiation exposure. The bases for personnel radiation doses should be derived from Chapter 11 of the SAR. The bases for components should be derived from the descriptions in various sections of the SAR including Section 4.4.
- The coolant should extend far enough above the core to ensure the coolant flows and pressures assumed in thermal-hydraulic analyses.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the design bases describe the requirements for the tank and that the detailed design is consistent with the design bases and acceptance criteria for the tank.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The tank system can withstand all anticipated mechanical and hydraulic forces and stresses to prevent loss of integrity which could lead to a loss of coolant or other malfunction that could interfere with safe reactor operation or shutdown.
- The penetrations and attachments to the tank are designed to ensure safe reactor operation. Safety and design considerations of any penetrations below the water level include analyses of potential malfunction and loss of coolant. The applicant discusses credible loss-of-coolant scenarios in Chapter 13, "Accident Analyses."
- The construction materials, treatment, and methods of attaching penetrations and components are designed to prevent chemical interactions among the tank, the coolant, and other components.
- The outer surfaces of the tank are designed and treated to avoid corrosion in locations that are inaccessible for the life of the tank. Tank surfaces will be inspected in accessible locations.
- The applicant has considered the possibility that primary coolant may leak into unrestricted areas, including ground water, and has included precautions to avoid the uncontrolled release of radioactive material.
- The design considerations include the shape and dimensions of the tank to ensure sufficient radiation shielding to protect personnel and components. Exposures have been analyzed, and acceptable shielding factors are included in the tank design.
- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the tank and included them in the technical specifications.

- The design features of the tank offer reasonable assurance of its reliability and integrity for its anticipated life. The design of the tank is acceptable to avoid undue risk to the health and safety of the public.

4.4 Biological Shield

Areas of Review

The radiation shields around non-power reactors are called biological shields and are designed to protect personnel and reduce radiation exposures to reactor components and other equipment. The principal design objective is to protect the staff and public. The second design objective is to make the shield as thin as possible, consistent with acceptable protection factors. Non-power reactors are sources of radiation used for a variety of reasons. Therefore, their shielding systems must allow access to the radiations internally near the reactor core and externally in radiation beams. Traditional methods of improving protection factors without increasing shield thickness are to use materials with higher density, higher atomic numbers for gamma rays, and higher hydrogen concentration for neutrons. The optimum shield design should consider all these.

Areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the biological shields include the following:

- The principal objective of the shield design should be to ensure that the projected radiation dose rates and accumulated doses in occupied areas do not exceed the limits of 10 CFR Part 20 and the guidelines of the facility ALARA (as low as is reasonably achievable) program discussed in Chapter 11 of the SAR.
- The shield design should address potential damage from radiation heating and induced radioactivity in reactor components and shields. The design should limit heating and induced radioactivity to levels that could not cause significant risk of failure.
- The tank or pool design, the coolant volume, and the solid shielding materials should be apportioned to ensure protection from all applicable radiation and all conditions of operation.
- Shielding materials should be based on demonstrated effectiveness at other non-power reactors with similar operating characteristics, and the

calculational models and assumptions should be justified by similar comparisons. New shielding materials should be justified by calculations, development testing, and the biological shield test program during facility startup.

- The analyses should include specific investigation of the possibilities of radiation streaming or leaking from shield penetrations, inserts, and other places where materials of different density and atomic number meet. Any such streaming or leakage should not exceed the stated limits.
- The shielding at experimental facilities, such as out-of-service beam tubes, should be sufficient to match the shielding factors of the gross surrounding shield.
- Supports and structures should ensure shield integrity, and quality control methods should ensure that fabrication and construction of the shield exceed the requirements for similar industrial structures.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements. The applicant should justify the proposed technical specifications in this section of the SAR.

Review Procedures

The reviewer should confirm that the objectives of the shield design bases are sufficient to protect the health and safety of the public and the facility staff, and that the design achieves the design bases. The reviewer should compare design features, materials, and calculational models with those of similar non-power reactors that have operated acceptably.

Evaluation Findings

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

- The analysis in the SAR offers reasonable assurance that the shield designs will limit exposures from the reactor and reactor-related sources of radiations so as not to exceed the limits of 10 CFR Part 20 and the guidelines of the facility ALARA program.

- The design offers reasonable assurance that the shield can be successfully installed with no radiation streaming or other leakage that would exceed the limits of 10 CFR Part 20 and the guidelines of the facility ALARA program.
- Reactor components are sufficiently shielded to avoid significant radiation-related degradation or malfunction.
- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the shield and included them in the technical specifications.

4.5 Nuclear Design

In this section of the SAR, the applicant should show how the systems described in this chapter function together to form a nuclear reactor that can be operated and shut down safely from any operating condition. The analyses should address all possible operating conditions (steady and pulsed power) throughout the reactor's anticipated life cycle. Because the information in this section describes the characteristics necessary to ensure safe and reliable operation, it will determine the design bases for most other chapters of the SAR and the technical specifications. The text, drawings, and tables should completely describe the reactor operating characteristics and safety features.

4.5.1 Normal Operating Conditions

Areas of Review

In this section of the SAR, the applicant should discuss the configuration for a functional reactor that can be operated safely.

The areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on normal operating conditions include the following:

- The information should show a complete, operable reactor core. Control rods should be sufficiently redundant and diverse to control all proposed excess reactivity safely and to safely shut down the reactor and maintain it in a shutdown condition. The analyses of reactivities should include individual and total control rod effects.

- Anticipated rearrangements of core components should account for uranium burnup, plutonium buildup, and poisons, both fission product and those added by design, for the life of the reactor. All operating core configurations should be compact, allowing no space within the core large enough to accept the addition of a fuel element or the addition of reactivity beyond that analyzed and found acceptable in Chapter 13 of the SAR.
- The analyses should show initial and changing reactivity conditions, control rod reactivity worths, and reactivity worths of fuel elements, reflector units, and such incore components as experimental facilities for all anticipated configurations. There should be a discussion of administrative and physical constraints that would prevent inadvertent movement that could suddenly introduce more than one dollar of positive reactivity or an analyzed safe amount, whichever was larger. These analyses should address movement, flooding, and voiding of core components.
- The reactor kinetic parameters and behavior should be shown, along with the dynamic reactivity parameters of the instrumentation and control systems. Analyses should prove that the control systems will prevent nuclear transients from causing loss of fuel integrity or uncontrolled addition of reactivity.
- The analyses should show that the control systems would prevent reactor damage if incore experimental facilities were to flood or void. This could be shown by reference to the analysis in Chapter 13 of the SAR.
- The information should include calculated core reactivities for the possible and planned configurations of the reactor core and control rods. If only one core configuration will be used over the life of the reactor, the applicant should clearly indicate this. For reactors in which various core configurations could be operated over time, the analyses should show the most limiting configuration (the most compact core and highest neutron flux densities). This information should be used for the analyses in Section 4.6 of the SAR.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that a complete, operable core has been analyzed.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The applicant has described the proposed initial core configuration and analyzed all reactivity conditions. These analyses also include other possible core configurations planned during the life of the reactor. The assumptions and methods used have been justified and validated.
- The analyses include reactivity and geometry changes resulting from burnup, plutonium buildup, and the use of poisons, as applicable.
- The reactivity analyses include the reactivity values for the core components, such as fuel elements, control rods, reflector components, and such incore and in-reflector components as experimental facilities. The assumptions and methods used have been justified.
- The analyses address the steady power operation and kinetic behavior of the reactor and show that the dynamic response of the control rods and instrumentation is designed to prevent uncontrolled reactor transients.
- The analyses show that any incore components that could be flooded or voided could not cause reactor transients beyond the capabilities of the instrumentation and control systems to prevent fuel damage or other reactor damage.
- The analyses address a limiting core that is the minimum size possible with the planned fuel. Since this core configuration has the highest power density, the applicant uses it in Section 4.6 of the SAR to determine the limiting thermal-hydraulic characteristics for the reactor.
- The analyses and information in this section describe a reactor core system that could be designed, built, and operated without unacceptable risk to the health and safety of the public.
- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for minimal operating conditions and included them in the technical specifications. The applicant has also justified the proposed technical specifications.

4.5.2 Reactor Core Physics Parameters

Areas of Review

In this section of the SAR, the applicant should present information on core physics parameters that determine reactor operating characteristics and are influenced by the reactor design. The principal objective of a non-power reactor is to obtain a radiation source that conforms to requirements for use, but does not pose an unacceptable risk to the health and safety of the public. By proper design, the reactor will operate at steady or pulsed power and the reactor systems will be able to terminate or mitigate transients without reactor damage. The areas of review should include the design features of the reactor core that determine the operating characteristics and the analytical methods for important contributing parameters. The results presented in this section of the SAR should be used in other sections of this chapter.

The areas of review are discussed further in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on reactor core physics parameters include the following:

- The calculational assumptions and methods should be justified and traceable to their development and validation, and the results should be compared with calculations of other similar facilities and previous experimental measurements. The ranges of validity and accuracy should be stated and justified.
- Uncertainties in the analyses should be provided and justified
- Methods used to analyze neutron lifetime, effective delayed neutron fraction, and reactor periods should be presented, and the results should be justified. Comparisons should be made with similar reactor facilities. The results should agree within the estimates of accuracy for the methods.
- Coefficients of reactivity (temperature, void, and power) should all be negative over the significant portion of the operating ranges of the reactor. The results should include estimates of accuracy. If any parameter is not negative within the error limits over the credible range of reactor operation, the combination of the reactivity coefficients should be analyzed and shown to be sufficient to prevent reactor damage and risk to the public from reactor transients as discussed in Chapter 13 of the SAR.

- Changes in feedback coefficients with core configurations, power level, and fuel burnup should not change the conclusions about reactor protection and safety, nor should they void the validity of the analyses of normal reactor operations, including pulsing, when applicable.
- The methods and assumptions for calculating the various neutron flux densities should be validated by comparisons with results for similar reactors. Uncertainties and ranges of accuracy should be given for other analyses requiring neutron flux densities, such as fuel burnup, thermal power densities, control rod reactivity worths, and reactivity coefficients.
- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that generally accepted and validated methods have been used for the calculations, evaluate the dependence of the calculational results on reactor design features and parameters, review the agreement of the methods and results of the analyses with the acceptance criteria, and review the derivation and adequacy of uncertainties and errors.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The analyses of neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity have been completed, using methods validated at similar reactors and experimental measurements.
- The effects of fuel burnup and reactor operating characteristics for the life of the reactor are considered in the analyses of the reactor core physics parameters.
- The numerical values for the reactor core physics parameters depend on features of the reactor design, and the information given is acceptable for use in the analyses of reactor operation.
- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the reactor core physics parameters and

included them in the technical specifications. The applicant has also justified the technical specifications.

4.5.3 Operating Limits

Areas of Review

In this section of the SAR, the applicant should present the nuclear design features necessary to ensure safe operation of the reactor core and safe shutdown from any operating condition. The information should demonstrate a balance between fuel loading, control rod worths, and number of control rods. The applicant should discuss and analyze potential accident scenarios, as distinct from normal operation, in Chapter 13 of the SAR.

The areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information operating limits include the following:

- All operational requirements for excess reactivity should be stated, analyzed, and discussed. These could pertain to at least the following:
 - temperature coefficients of reactivity
 - fuel burnup between reloads or shutdowns
 - void coefficients
 - xenon and samarium override
 - overall power coefficient of reactivity if not accounted for in the items listed above
 - experiments
- Credible inadvertent insertion of excess reactivity should not damage the reactor or fuel; this event should be analyzed in Sections 4.5 and 4.6 and Chapter 13 of the SAR.
- The minimum amount of total control rod reactivity worth to ensure reactor subcriticality should be stated.

- A transient analysis assuming that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region should be performed. This analysis could also be based on a credible failure of a movable experiment. The analysis should show that the reactor would not be damaged and fuel integrity would not be lost. Reactivity additions under accident conditions should be analyzed in Chapter 13 of the SAR.
- An analysis should be performed that examines reactivity assuming that the reactor is operating at its maximum licensed conditions, normal electrical power is lost, and the control rod of maximum reactivity worth and any non-scrammable control rods remain fully withdrawn. The analysis should show how much negative reactivity must be available in the remaining scrammable control rods so that, without operator intervention, the reactor can be shut down safely and remain subcritical without risk of fuel damage even after temperature equilibrium is attained, all transient poisons such as xenon are reduced, and movable experiments are in their most reactive position.
- On the basis of analysis, the applicant should justify a minimum negative reactivity (shutdown margin) that will ensure the safe shutdown of the reactor. This discussion should address the methods and the accuracy with which this negative reactivity can be determined to ensure its availability.
- The core configuration with the highest power density possible for the planned fuel should be analyzed as a basis for safety limits and limiting safety system settings in the thermal-hydraulic analyses. The core configuration should be compared with other configurations to ensure that a limiting configuration is established for steady power and pulsed operation, if applicable.
- The applicant should propose and justify technical specifications for safety limits, limiting safety system settings, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14 of the format and content guide.

Review Procedures

The reviewer should confirm that the methods and assumptions used in this section of the SAR have been justified and are consistent with those in other sections of this chapter.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The applicant has discussed and justified all excess reactivity factors needed to ensure a readily operable reactor. The applicant has also considered the design features of the control systems that ensure that this amount of excess reactivity is fully controlled under normal operating conditions.
- The discussion of limits on excess reactivity shows that a credible rapid withdrawal of the most reactive control rod or other credible failure that would add reactivity to the reactor would not lead to loss of fuel integrity. Therefore, the information demonstrates that the proposed amount of reactivity is available for normal operations, but would not cause unacceptable risk to the public from a transient.
- The definition of the shutdown margin is negative reactivity obtainable by control rods to ensure reactor shutdown from any reactor condition, including a loss of normal electrical power. With the assumption that the most reactive control rod is inadvertently stuck in its fully withdrawn position, and non-scrammable control rods are in the position of maximum reactivity addition, the analysis derives the minimum negative reactivity necessary to ensure safe reactor shutdown. The applicant conservatively proposes a shutdown margin of α in the technical specifications. The applicant has justified this value; it is readily measurable and is acceptable.
- The SAR contains calculations of the peak thermal power density achievable with any core configuration. This value is used in the calculations in the thermal-hydraulic section of the SAR to derive reactor safety limits and limiting safety system settings, which are acceptable.

4.6 Thermal-Hydraulic Design

Areas of Review

The information in this section should enable the reviewer to determine the limits on cooling conditions necessary to ensure that fuel integrity will not be lost under any reactor conditions (including pulsing, if applicable) including accidents. For many licensed non-power reactors that operate at low power, the fuel temperatures remain far lower than temperatures at which fuel could be damaged. For these reactors, the analyses and discussions may not constitute a critical part of

the SAR. However, for non-power reactors that operate at higher fuel temperatures or power densities, the thermal-hydraulic analyses may be the most important and most limiting features of reactor safety. Because some of the factors in the thermal-hydraulic design are based on experimental measurements and correlations that are a function of coolant conditions, the analyses should confirm that the values of such parameters are applicable to the reactor conditions analyzed.

The areas of review are discussed in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on thermal-hydraulic design include the following:

- The applicant should propose criteria and safety limits based on the criteria for acceptable safe operation of the reactor, thus ensuring fuel integrity under all analyzed conditions. The discussion should include the consequences of these conditions and justification for the alternatives selected. These criteria could include the following:
 - There should be no coolant flow instability in any fuel channel that could lead to a significant decrease in fuel cooling.
 - The departure from the nucleate boiling ratio should be no less than 2 in any fuel channel.
- Safety limits, as discussed in Chapter 14 of the format and content guide, should be derived from the analyses described above, the analyses in Section 4.5.3 of the SAR, and any other necessary conditions. The safety limits should include conservative consideration of the effects of uncertainties or tolerances and should be included in the technical specifications.
- Limiting safety system settings (LSSSs), as discussed in Chapter 14 of the format and content guide of the SAR, should be derived from the analyses described above, the analyses in Section 4.5.3 of the SAR, and any other necessary conditions. These settings should be chosen to maintain fuel integrity when safety system protective actions are conservatively initiated at the LSSSs.
- A forced-flow reactor should be capable of switching to natural-convection flow without damaging fuel and jeopardizing safe reactor shutdown. Loss of normal electrical power should not change this criterion.

- For a pulsing reactor, limits on pulse sizes and transient rod characteristics should ensure that fuel is not damaged by pulsed operations. These limits should be based on the thermal-hydraulic analyses and appear in the technical specifications. Changes in fuel characteristics from steady power operation that affect pulsed operation should be taken into account. Such factors as hydrogen migration, oxidation of cladding, and decrease in burnable poison should be addressed, if applicable.

Review Procedures

The reviewer should confirm that the thermal-hydraulic analyses for the reactor are complete and address all issues that affect key parameters (e.g., flow, temperature, pressure, power density, and peaking). The basic approach is an audit of the SAR analyses, but the reviewer may perform independent calculations to confirm SAR results or methods.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will appear in the staff's safety evaluation report:

- The information in the SAR includes the thermal-hydraulic analyses for the reactor. The applicant has justified the assumptions and methods and validated their results.
- All necessary information on the primary coolant hydraulics and thermal conditions of the fuel are specific for this reactor. The analyses give the limiting conditions of these features that ensure fuel integrity.
- Safety limits and limiting safety system settings are derived from the thermal-hydraulic analyses. The values have been justified and appear in the technical specifications. The thermal-hydraulic analyses on which these parameters are based ensure that overheating during any operation or credible event will not cause loss of fuel integrity and unacceptable radiological risk to the health and safety of the public.

5 REACTOR COOLANT SYSTEMS

This chapter gives guidance for evaluating the design bases, descriptions, and functional analyses of the reactor coolant systems. The principal purpose of the coolant system is to safely remove the fission and decay heat from the fuel and dissipate it to the environment. However, the coolant in the primary systems of most non-power reactors serves more functions than just efficient removal of heat. It can act as radiation shielding for the reactor, fuel storage facilities, and in some designs, experimental facilities and experiments. In open-pool reactors, the coolant is the only vertical shielding. In many designs the reactor coolant also acts as a core moderator and reflector. Because of these many functions of the reactor coolant, the design of the reactor coolant systems is based on choosing among interdependent parameters, including thermal power level, research capability, available fuel type, reactor core physics requirements, and radiation shielding.

The principal licensing basis of non-power reactors is the thermal power developed in the core during operation. This basis also applies to the few non-power reactors licensed to operate at such low power levels that no significant core temperature increases would occur during normal operation. Such reactors may not require an engineered coolant system. For those reactors, the applicant should, in Chapter 4, "Reactor Description," of the SAR, discuss the dissipation of the heat produced, estimate potential temperature increases during reactor operation, and justify why an engineered coolant system is not required. In this chapter the applicant should summarize those considerations and conclusions.

For all other non-power reactors, the applicant should describe and discuss in this chapter systems to remove and dispose of the waste heat. The design bases of the core cooling systems for the full range of normal operation should be derived in Chapter 4 of the SAR. All auxiliary systems and subsystems that use and contribute to the heat load of either the primary or secondary coolant system should also be described and discussed in this chapter. Any auxiliary systems using coolant from other sources, such as building service water, should be discussed in Chapter 9, "Auxiliary Systems." The design bases of any features of the core cooling system designed to respond to potential accidents or to mitigate the consequences of potential accidents should be derived from the analyses in Chapter 13, "Accident Analyses." These features should be summarized in this chapter and discussed in detail in Chapter 6, "Engineered Safety Features," of the SAR. In this chapter the applicant should discuss and reference the technical specifications that are needed to ensure operability consistent with SAR analyses assumptions.

The primary loops of the coolant systems of most licensed non-power reactors are of two basic types, forced-convection and natural thermal-convection. Facilities using forced-convection cooling also may be licensed to operate in the natural-convection mode and should be capable of dissipating decay heat in that mode.

This chapter gives the review plan and acceptance criteria for information on the heat removal systems. The information suggested for this section of the SAR is outlined in Chapter 5 of the format and content guide.

5.1 Summary Description

The applicant should summarize the principal features of the reactor coolant systems, including the following:

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

5.2 Primary Coolant System

Areas of Review

As noted above, non-power reactor design requires choosing among several interdependent variables. Usually, the design represents a compromise between the neutron flux densities required and the need to dissipate thermal power. The final design depends on the intended uses of the reactor, the resources available, and the priorities of the facility owner. The objective of the final design is ensured safety. The primary coolant system is a key component in the design and should have the capability to

- remove the fission and decay heat from the fuel during reactor operation and decay heat during reactor shutdown
- for most non-power reactors, transfer the heat to a secondary coolant system for controlled dissipation to the environment

- maintain high water quality to limit corrosion of fuel cladding, control and safety rods, reactor vessel or pool, and other essential components
- provide radiation shielding of the core and other components such as beam tubes and fuel storage facilities
- provide neutron moderation and reflection in the core
- prevent uncontrolled leakage or discharge of contaminated coolant to the unrestricted environment

The basic requirements for these functions are generally derived and analyzed in other chapters of the SAR. In this chapter the applicant should describe how the coolant system provides these functions. Specific areas of review for this section are discussed in Section 5.2 of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the primary coolant system include the following:

- Chapter 4 of the SAR should contain analyses of the reactor core including coolant parameters necessary to ensure fuel integrity. Safety limits (SLs) and limiting safety system settings (LSSSs) to ensure fuel integrity should be derived from those analyses and included in the technical specifications. Examples of cooling system variables on which LSSSs may be established are maximum thermal power level for operation in natural-convection flow, maximum coolant temperature, minimum coolant flow rate, minimum pressure of coolant at the core, and minimum pool depth above the core. The analyses in this section should show that the components and the functional design of the primary coolant system will ensure that no LSSS will be exceeded through the normal range of reactor operation. The analyses should address forced flow or natural-convection flow, or both for reactors licensed for both modes. The design should show that the passive or fail-safe transition from forced flow to natural-convection flow is reasonably ensured in all forced-flow non-power reactors and that continued fuel integrity is ensured.

The functional design should show that safe reactor shutdown and decay heat removal are sufficient to ensure fuel integrity for all possible reactor conditions, including potential accident scenarios. Scenarios that postulate loss of flow or loss of coolant should be analyzed in Chapter 13 and the results summarized in this section of the SAR.

- The descriptions and discussions should show that sufficient instrumentation, coolant parameter sensors, and control systems are provided to monitor and ensure stable coolant flow, respond to changes in reactor power levels, and provide for a rapid reactor shutdown in the event of loss of cooling. There also should be instrumentation for monitoring the radiation of the primary coolant because elevated radiation levels could indicate a loss of fuel cladding integrity. There should be routine sampling for gross radioactivity in the coolant and less frequent radioactive spectrum analysis to identify the isotopes and concentrations found in the coolant. This spectrum analysis may also detect cladding failure at its earliest stages.
- The primary coolant should provide a chemical environment that limits corrosion of fuel cladding, control and safety rod surfaces, reactor vessels or pools, and other essential components. Aluminum-clad fuel operated at high power density will develop an oxide coating that could decrease heat conductivity (Griess et al., 1964). Chapter 4 of the SAR should contain discussion and analyses of the dependence of oxide formation on water quality and other factors. Other requirements for water purity should be analyzed in the SAR, and proposed values of conductivity and pH should be justified. Experience at non-power reactors has shown that the primary water conditions, electrical conductivity $\leq 5 \mu\text{mho/cm}$ and pH between 5.5 and 7.5, can usually be attained with good housekeeping and a good filter and demineralizer system. Chemical conditions should be maintained, as discussed in Section 5.4 of this standard review plan.
- Most non-power reactors consist of a core submerged in a pool or tank of water. The water helps shield personnel in the reactor room and the unrestricted area from core neutrons and gamma rays. It also decreases potential neutron activation and radiation damage to such reactor components as the pool liner, beam port gaskets, in-pool lead shields, and concrete biological shield. The applicant should discuss these factors in Chapter 4 of the SAR. To ensure that the design of the primary coolant vessel is acceptable, exposure limits on materials discussed in Chapter 4 should not be exceeded, and exposures to personnel, as discussed in Chapter 11, "Radiation Protection Program and Waste Management," should not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility ALARA (as low as is reasonably achievable) program.
- Radioactive species including nitrogen-16 and argon-41 may be produced in the primary coolant. Additional radioactivity may occur as a result of neutron activation of coolant contaminants and fission product leakage from the fuel. Provisions for limiting personnel radiological hazards should maintain potential exposures from coolant radioactivity below the

requirements of 10 CFR Part 20 and should be consistent with the facility ALARA program. To ensure that facilities or components for controlling, shielding, or isolating nitrogen-16 are acceptable, potential exposures should not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility ALARA program. The nitrogen-16 control system is discussed in Section 5.6 of this standard review plan.

Argon-41 is a ubiquitous radionuclide produced at non-power reactors. Because this radionuclide may be the major release to the environment during normal operation, special analyses and discussion of its production and consequences should be given in Chapter 11 of the SAR. If any special design or operational features of the primary coolant system modify or limit exposures from argon-41, they should be discussed in this section of the SAR. This discussion should demonstrate that any facilities or components added to the primary coolant system to modify argon-41 releases can limit potential personnel exposures to the values found acceptable in Chapter 11.

Closed systems also may experience a buildup of hydrogen in air spaces in contact with the coolant. The discussion should show that it is not possible to have hydrogen build up to concentrations that are combustible. This may require gas sweep systems and hydrogen concentration monitoring. These systems should be discussed in Chapter 9.

- Because the primary coolant may provide essential fuel cooling and radiation shielding, the system design should avoid uncontrolled release or loss of coolant. Some design features to limit losses include locating components of the primary coolant system above the core level, avoiding drains or valves below core level in the pool or tank, providing syphon-breaks in piping that enters the primary vessel or pool, and providing check valves to preclude backflow. The designs and locations of such features should provide reasonable assurance that loss of coolant that could uncover the core is very unlikely. A potential accident of rapid loss of coolant should be analyzed in Chapter 13 and summarized in this section of the SAR.

Heavy water systems require additional design features because of the radiological hazards of tritium production in the coolant. These systems should be designed with systems to detect minor leakage. They also should be designed so that heavy water, if lost from the system, will be contained and not released to the environment.

If contaminated coolant were lost from the primary coolant system, the design and analyses should ensure that potential personnel exposures and uncontrolled releases to the unrestricted environment do not exceed

acceptable radiological dose consequence limits derived from the accident analyses. The radiological consequences from the contaminated coolant should be discussed in Chapter 11 and summarized in this section of the SAR. Necessary surveillance provisions should be included in the technical specifications.

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

Review Procedures

The reviewer should compare the functional design and the operating characteristics of the primary coolant system with the bases for the design presented in this and other relevant chapters of the SAR. The system design should meet the appropriate acceptance criteria presented above considering the specific facility design under review.

Evaluation Findings

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

- The primary coolant system is designed in accordance with the design bases derived from all relevant analyses in the SAR.
- Design features of the primary coolant system and components give reasonable assurance of fuel integrity under all possible reactor conditions. The system is designed to remove sufficient fission heat from the fuel to allow all licensed operations without exceeding the established limiting safety system settings that are included in the technical specifications.
- Designs and locations of primary coolant system components have been specifically selected to avoid coolant loss that could lead to fuel failure, uncontrolled release of excessive radioactivity, or damage to safety systems or experiments. Heavy water systems are designed to quickly detect leakage and prevent the release of heavy water to the environment. *(If an emergency core cooling system is required to prevent a loss of fuel integrity, it is evaluated in connection with the review of the engineered safety features.)*
- *(For reactors licensed to operate with forced-convection coolant flow)*
The primary coolant system is designed to convert in a passive or fail-safe method, to natural-convection flow sufficient to avoid loss of fuel integrity.

(This feature is evaluated in conjunction with the reviews of the reactor description and accidents.)

- The chemical quality of the primary coolant will limit corrosion of the fuel cladding (or other primary barrier to release of fission products), the control and safety rod cladding, the inside of the reactor vessel (or pool), and other essential components in the primary coolant system for the duration of the license and for the projected utilization time of the fuel. *(For a closed primary coolant system)* Systems are present that will prevent hydrogen concentrations from reaching combustible limits.
- The size and shape of the primary vessel or pool will provide sufficient radiation shielding to maintain personnel exposures below the limits in 10 CFR Part 20 and will provide a heat reservoir sufficient for anticipated reactor operations.
- Primary coolant system instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary primary coolant system operability for reactor operations as analyzed in the SAR.
- The design bases of the primary coolant system provide reasonable assurance that the environment and the health and safety of the public will be protected.

5.3 Secondary Coolant System

Areas of Review

Secondary coolant systems of non-power reactors are designed to transfer reactor heat from the primary coolant system to the environment. Non-power reactors may be designed in three ways: with a continuously operating secondary coolant system, with an on-demand secondary coolant system, and without a secondary coolant system. For most of the licensed reactors, the coolant systems are designed for continuous operation at licensed power level. Therefore, the secondary coolant system in these reactors must be designed to dissipate heat continuously. Some non-power reactors are designed and licensed to operate at low power levels, or for limited time intervals, so that the primary coolant system can absorb or dissipate the heat without a continuously operating secondary coolant system. Some non-power reactors require no secondary coolant system. In this section of the SAR, the applicant should justify how any necessary heat

dissipation is accomplished. Specific areas of review for this section are discussed in Section 5.3 of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the secondary coolant system include the following:

- The required operating characteristics of the primary coolant system should be given in Section 5.2 of the SAR. The analyses and discussions of Section 5.3 should demonstrate that the secondary coolant system is designed to allow the primary coolant system to transfer the heat as necessary to ensure fuel integrity. The analyses should address primary coolant systems operating with forced flow, natural-convection flow, or both for reactors licensed for both modes. The design should show that the secondary coolant system is capable of dissipating all necessary fission and decay heat for all potential reactor conditions as analyzed in the SAR.
- Some non-power reactors are designed with secondary coolant systems that will not support continuous reactor operation at full licensed power. This is acceptable, provided the capability and such limiting conditions as maximum pool temperature are analyzed in the SAR and included in the technical specifications.
- The primary coolant will usually contain radioactive contamination. The design of the total coolant system should ensure that release of such radioactivity through the secondary coolant system to the unrestricted environment would not lead to potential exposures of the public in excess of the requirements of 10 CFR Part 20 and the ALARA program guidelines. Designs should ensure that the primary coolant system pressure is lower than the secondary coolant system pressure across the heat exchanger under all anticipated conditions, the secondary coolant system is closed, or radiation monitoring and effective remedial capability are provided. The secondary coolant system should prevent or acceptably mitigate uncontrolled release of radioactivity to the unrestricted environment. Periodic samples of secondary coolant should be analyzed for radiation. Action levels and required actions should be discussed.
- The secondary coolant system should accommodate any heat load required of it in the event of a potential engineered safety feature operation or accident conditions as analyzed in Chapters 6 and 13 of the SAR.

- The secondary coolant system design should provide for any necessary chemical control to limit corrosion or other degradation of the heat exchanger and prevent chemical contamination of the environment.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

Review Procedures

The reviewer should verify that all reactor conditions, including postulated accidents, requiring transfer of heat from the primary coolant system to the secondary coolant system have been discussed. The reviewer should verify that the secondary coolant system is capable of removing and dissipating the amount of heat and the thermal power necessary to ensure fuel integrity. The reviewer should also confirm the analyses of secondary coolant system malfunctions including the effects on reactor safety, fuel integrity, and the health and safety of the public.

Evaluation Findings

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

- Design features of the secondary coolant system and components will allow the transfer from the primary coolant system the necessary reactor heat under all possible reactor conditions.
- Locations and design specifications for secondary coolant system components ensure that malfunctions in the system will not lead to reactor damage, fuel failure, or uncontrolled release of radioactivity to the environment.
- Secondary coolant system instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.
- The secondary coolant system is designed to respond as necessary to such postulated events as a loss-of-primary-coolant accident and a loss of forced coolant flow in the primary coolant system.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary secondary coolant system operability for normal reactor operations.

5.4 Primary Coolant Cleanup System

Areas of Review

Experience has shown that potable water supplies are usually not acceptably pure for use as a reactor primary coolant without additional cleanup. Most licensed non-power reactors contain solid fuel elements immersed in the primary coolant water. Experience has also shown that oxide buildup on aluminum-clad fuel operated at high power densities can reduce heat transfer (Griess et al., 1964). The rate of buildup depends on several operational characteristics, including the pH of the coolant. Therefore, this process should be discussed in Chapter 4 and summarized in this section of the SAR if it contributes to establishing requirements for primary coolant purity. The purity of the primary coolant should be maintained as high as reasonably possible for the following reasons.

- to limit the chemical corrosion of fuel cladding, control and safety rod cladding, reactor vessel or pool, and other essential components in the primary coolant system
- to limit the concentrations of particulate and dissolved contaminants that might become radioactive by neutron irradiation
- to maintain high transparency of the water for observation of submerged operational and utilization components

Specific areas of review for this section are discussed in Section 5.4 of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the primary coolant cleanup system include the following:

- The primary coolant quality should be maintained in the ranges established as acceptable in Chapters 4 and 11 of the SAR. These analyses for high-power reactors (> 2 MW) should include the buildup of an oxide film on aluminum cladding. Experience has shown that quality water conditions, electrical conductivity $\leq 5 \mu\text{mho/cm}$ and pH between 5.5 and 7.5, can usually be achieved by good housekeeping and a cleanup loop with particulate filters and demineralizers. Such a system is acceptable unless the SAR analyses establish other purity conditions as acceptable.
- Radioactively contaminated resins and filters should be disposed of or regenerated in accordance with radiological waste management plans

- discussed in Chapter 11, and potential exposures and releases to the unrestricted environment shall not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility ALARA program.
- Location, shielding, and radiation monitoring of the water cleanup system for routine operations and potential accidental events should be such that the occupational staff and the public are protected from radiation exposures exceeding the requirements of 10 CFR Part 20 and acceptable radiological consequence dose limits for accidents.
- Location and functional design of the components of the water cleanup system should ensure the following:
 - Malfunctions or leaks in the system do not cause uncontrolled loss or release of primary coolant.
 - Personnel exposure and release of radioactivity do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program.
 - Safe reactor shutdown is not prevented.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

Review Procedures

The reviewer should compare the design bases for the primary coolant water quality with the design bases by which the primary coolant cleanup system will achieve the requirements. The comparison should include performance specifications, schematic diagrams, and discussion of the functional characteristics of the cleanup system. The reviewer should evaluate (1) design features to ensure that leaks or other malfunctions would not cause inadvertent damage to the reactor or exposure of personnel and (2) the plan for control and disposal of radioactive filters and demineralizer resins.

Evaluation Findings

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

- The design bases and functional descriptions of the primary water cleanup system give reasonable assurance that the required water quality can be

achieved. The design ensures that corrosion and oxide buildup of fuel cladding and other essential components in the primary coolant system will not exceed the acceptable limits or the recommendations of the fuel vendor.

- Experience has shown that the pH of the primary coolant can influence the rate of oxide buildup on aluminum-clad fuel. The pH and the proposed system are consistent with the analysis for the effect of oxide on heat transfer from the fuel.
- The primary coolant cleanup system and its components have been designed and selected so that malfunctions are unlikely. Any malfunctions or leaks will not lead to radiation exposure to personnel or releases to the environment that exceed the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The plans for controlling and disposing of radioactivity accumulated in components of the primary water cleanup system, which results from normal operations and potential accident scenarios, conform with applicable regulations, including 10 CFR Part 20, and acceptable radiological consequence dose limits for accidents.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary primary water cleanup system operability for normal reactor operations.

5.5 Primary Coolant Makeup Water System

Areas of Review

During operations at non-power reactors, primary coolant must be replaced or replenished. Coolant may be lost through evaporation in open-pool systems, radiolysis, leaks from the system, and other operational activities. Although each non-power reactor should have a makeup water system or procedure to meet projected operational needs, the system need not be designed to provide a rapid, total replacement of the primary coolant inventory. Specific areas of review for this section are discussed in Section 5.5 of the format and content guide

Acceptance Criteria

The acceptance criteria for the information on the primary coolant makeup water system include the following:

- The projected loss of primary coolant water inventory for anticipated reactor operations should be discussed. The design or plan for supplying makeup water should ensure that those operational requirements are satisfied.
- If storage of treated makeup water is required by the design bases of the primary coolant system, the makeup water system or plan should ensure that such water is provided.
- Not all non-power reactors must provide makeup water through hardware systems directly connecting the reactor to the facility potable water supply. However, for those that do, the makeup water system or plan should include components or administrative controls that prevent potentially contaminated primary coolant from entering the potable water system.
- The makeup water system or plan should include features to prevent loss or release of coolant from the primary coolant system.
- The makeup water system need not have a functional relationship with any installed emergency core cooling system (ECCS). If it does, it should not interfere with the availability and operability of the ECCS.
- The makeup water system or plan should include provisions for recording the use of makeup water to detect changes that indicate leakage or other malfunction of the primary coolant system.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

Review Procedures

The reviewer should compare the design bases and functional requirements for replenishing primary coolant including the quantity and quality of water, the activities or functions that remove primary coolant, and the systems or procedures to accomplish water makeup with the acceptance criteria. The review should focus, as applicable, on safety precautions to preclude overfilling of the reactor coolant system, loss of primary coolant through the nonradioactive service drain system, and the release of primary coolant back through the makeup system into potable water supplies.

Evaluation Findings

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

- The design bases, functional descriptions, and procedures for the primary coolant makeup water system give reasonable assurance that the quantity and quality of water required will be provided.
- The system design or procedures will prevent overfilling of the primary coolant system or malfunction of the makeup water system and will prevent the loss or release of contaminated primary coolant that would exceed the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The system design or procedures will prevent contaminated primary coolant from entering the potable water system through the makeup water system.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary makeup water system operability for normal reactor operations.

5.6 Nitrogen-16 Control System

Areas of Review

Non-power reactors that use either light or heavy water for neutron moderation or cooling will produce nitrogen-16 by the fast neutron-proton reaction in oxygen-16. Nitrogen-16, a high-energy beta and gamma ray emitter with a half-life of approximately 7 seconds, is a potential source of high radiation exposure at water-cooled non-power reactors. It tends to remain dissolved in the primary coolant water as it leaves the core. The quantity and concentration of nitrogen-16 should be considered and provisions made to control personnel exposure. Because of the relatively short half-life, potential doses can be decreased by delaying the coolant within shielded regions. For reactors using natural-convection cooling in a large open pool, stirring or diffusing the convection flow to the surface can produce a delay. For forced-flow cooling, passing the coolant through a large shielded and baffled tank can produce the delay. In some non-power reactor designs, the entire primary coolant system may be shielded. Specific areas of review for this section are discussed in Section 5.6 of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the nitrogen-16 control system include the following:

- The reduction in personnel exposure to nitrogen-16 should be consistent with the nitrogen-16 analyses in Chapter 11 of the SAR. Total dose shall not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility ALARA program.
- System design should not
 - decrease cooling efficiency so that any limiting safety system setting would be exceeded
 - lead to uncontrolled release or loss of coolant if a malfunction were to occur
 - prevent safe reactor shutdown and removal of decay heat sufficient to avoid fuel damage
- The applicant should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

Review Procedures

The reviewer should evaluate the design bases and functional requirements of the system designed to control personnel exposures to nitrogen-16 by

- confirming the amount of nitrogen-16 predicted by the SAR analysis at the proposed power level and the potential personnel exposure rates, including exposures from direct radiation and airborne nitrogen-16
- reviewing the type of system and the decrease in exposure rates
- reviewing the effect of the proposed system on the full range of normal reactor operations
- reviewing the possible effects of malfunctions of the nitrogen-16 control system on reactor safety, safe reactor shutdown, and release of contaminated primary coolant

Evaluation Findings

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

- Design bases and design features give reasonable assurance that the nitrogen-16 control system can function as proposed and reduce potential doses to personnel so that doses do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program.
- Design and functional operation of the nitrogen-16 control system give reasonable assurance that the system will not interfere with reactor cooling under anticipated reactor operating conditions and will not reduce cooling below the acceptable thermal-hydraulic performance discussed in Chapter 4 of the SAR.
- Design features give reasonable assurance that malfunction of the nitrogen-16 control system will not cause uncontrolled loss or release of primary coolant and will not prevent safe reactor shutdown.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary nitrogen-16 control system operability for normal reactor operations.

5.7 Auxiliary Systems Using Primary Coolant

Areas of Review

The primary coolant of a non-power reactor may serve functions other than cooling the reactor fuel. Some of these auxiliary functions involve cooling other heated components, which may affect the heat load of the primary coolant system. Some of the auxiliary functions involve radiation shielding, which may not contribute to the heat load but could require that the primary coolant be diverted or distributed to subsystems not involving core cooling.

Auxiliary uses of the primary coolant could affect its availability as a fuel coolant, which is its principal use. Although the principal discussions of these auxiliary systems should be located in other sections of the SAR, their effects on the coolant systems should be summarized in this section. Auxiliary systems that may use primary coolant include the following:

- experiment cooling
- experimental facility cooling
- experimental facility shielding
- biological shield cooling
- thermal shield cooling
- fuel storage cooling or shielding

Specific areas of review for this section are discussed in Section 5.7 of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the auxiliary systems using primary coolant include the following:

- The system should remove sufficient projected heat to avoid damage to the cooled device.
- The system should not interfere with the required operation of the primary core cooling system.
- Any postulated malfunction of an auxiliary system should not cause uncontrolled loss of primary coolant or prevent a safe reactor shutdown.
- The shielding system using primary coolant should provide sufficient protection factors to prevent personnel exposures that exceed the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The system should not cause radiation exposures or release of radioactivity to the environment that exceeds the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the technical specifications.

Review Procedures

The reviewer should verify that auxiliary cooling or shielding using primary coolant is described in this section of the SAR for any component (other than the core) in which potentially damaging temperature increases or excessive radiation exposures are predicted. If the potential exists for radiation heating of components near the reactor core, the reviewer should verify that the heat source, temperature increases, heat transfer mechanisms, and heat disposal have been discussed and

analyzed. The reviewer should verify that the potential personnel radiation exposures from sources shielded by primary coolant have been analyzed and the protection factors provided by the coolant have been discussed.

Evaluation Findings

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

- The applicant has described and analyzed auxiliary systems that use primary coolant for functions other than in-core fuel cooling, has derived the design bases from other chapters of the SAR, has analyzed any reactor components located in high radiation areas near the core for potential heating that could cause damage to the reactor core or failure of the component, and has planned acceptable methods to remove sufficient heat to ensure the integrity of the components. The coolant for these systems is obtained from the purified primary coolant system without decreasing the capability of the system below its acceptable performance criteria for core cooling.
- The applicant has analyzed any reactor components or auxiliary systems for which primary coolant helps shield personnel from excessive radiation exposures. The use of the coolant for these purposes is acceptable, and the estimated protection factors limit the exposures to the requirements of 10 CFR Part 20 and the facility ALARA program guidelines. There is reasonable assurance that credible and postulated malfunctions of the auxiliary cooling systems will not lead to uncontrolled loss of primary coolant, radiation exposures, or release of radioactivity to the unrestricted environment that exceeds the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The technical specifications, including testing and surveillance, provide reasonable assurance of necessary auxiliary cooling system operability for normal reactor operations.

5.8 Reference

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

6 ENGINEERED SAFETY FEATURES

This chapter gives the review plan and acceptance criteria for active or passive engineered safety features (ESFs) of the reactor facility that are designed to mitigate the consequences of accidents. The concept of ESFs evolved from the defense-in-depth philosophy of 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 SAR analyses of accidents that could occur, even though prudent and conservative designs of the facility have made these accidents very unlikely. The NRC reviewer may find that the SAR analyses show that ESFs are not needed for a proposed design.

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

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

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

In the past, the SAR accident analyses for many non-power reactors have shown that ESFs are not required, even for the maximum hypothetical accident. In other cases, the accident analyses 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 by the applicant should contain the design bases for any required ESF. The ESF design should be as basic and fail safe as practicable. Because non-power reactors are conservatively designed, few accidents should require redundant or diverse ESF systems. Some factors the reviewer should evaluate to verify whether redundant or diverse ESFs 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 reviewer should examine the methods and criteria proposed by the applicant for testing to demonstrate ESF operability. The reviewer should evaluate the necessary components, functional requirements, related setpoints, interlocks, bypasses, and surveillance tests for each ESF and should 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 should also 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 NRC staff has not found it necessary to follow the board recommendation to develop separate criteria for the evaluation of research reactor accidents, since most research reactors to date have been able to meet the conservative 10 CFR Part 20 criteria. American National Standards Institute/American Nuclear Society ANSI/ANS-15.7, "Research Reactor Site Evaluation," contains additional information on doses to the public from releases of radioactive material.

The 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 should compare the results of the accident analyses against 10 CFR Part 20. For research reactors licensed before January 1, 1994, the doses calculated in the accident analyses will be acceptable if they are less than the old 10 CFR Part 20 limits (10 CFR 20.1 through 20.602 and appendices) of 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. The reviewer should conduct each review on a case-by-case basis. In several instances, the staff has accepted very conservative accident analyses with results greater than the 10 CFR Part 20 dose limits discussed above. Research reactors that received their initial operating license after January 1, 1994, must show that exposures meet the

requirements of the revised 10 CFR Part 20 (10 CFR 20.1001 through 20.2402 and appendixes). Occupational exposure is discussed in 10 CFR 20.1201, and public exposure is discussed in 10 CFR 20.1301. If a research reactor applicant cannot meet the above doses, the reviewer should examine the safety analyses to ensure that the evaluation of accidents is not overly conservative.

If the facility meets the definition of a test reactor, the reviewer should compare the results against the doses in 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses in 10 CFR Part 100 are reference values. Any further references to 10 CFR Part 100 in this chapter pertain to test reactors only.

The reviewer should evaluate 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 to be accomplished. The applicant should present any need for site utility redundancy and the specific design features that provide redundancy for the components of each ESF.

The applicant should provide schematic diagrams 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 ESFs that may be required for a proposed design are the confinement, the containment, and the emergency core cooling system (ECCS), which are discussed in this chapter of the format and content guide. The postulated accident analyses by the applicant determine if a non-power reactor facility needs a confinement, a containment, an ECCS, or no ESFs. The reviewer will find that heating, ventilation, and air conditioning (HVAC) and air exhaust systems at non-power reactors generally serve to limit the release of airborne radioactive material. The reviewer should verify that those features in HVAC systems required to mitigate the consequences of accidents were treated as ESFs. This review plan gives guidance for the evaluation of information on confinement, containment, and ECCS ESFs. Information on 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 will not be required. If the reviewer confirms that the safety analyses show 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 containments as an additional design conservatism.

6.1 Summary Description

In this section of the SAR, the applicant should briefly describe all the ESFs in the facility design and summarize 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 should be given in Chapter 13 of the SAR and the detailed discussions of the ESFs in Section 6.2 of the SAR. These summaries should include the design bases, the performance criteria, and the full range of reactor conditions, including accident conditions, under which the equipment or systems must maintain function. The evaluation procedures and criteria for the confinement, the containment, and the ECCS are given in the following section.

The applicant may submit simple block diagrams and drawings that show the location, basic function, and relationship of each ESF to the facility. The summary description should contain enough information for an overall understanding of the functions and relationships of the ESFs to the operation of the facility. Detailed drawings, schematic diagrams, data, and analyses should be presented in Section 6.2 of the SAR for each specific ESF.

6.2 Detailed Descriptions

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

6.2.1 Confinement

If the HVAC and any air exhaust or liquid release systems associated with the confinement are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13 and thereby mitigate its consequences, they 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 releases to the environment will be controlled. The airborne radionuclide normally released from the envelope of the reactor is argon-41, which may be continuously swept from the reactor building to diffuse and disperse in the atmosphere. The applicant should ensure that during this controlled release, neither the public nor the facility staff receive a dose greater

than regulatory limits. This function of the confinement and the HVAC system is not considered a function of 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 given in Chapter 3, "Design of Structures, Systems, and Components," and Chapter 9, "Auxiliary Systems," of the SAR. Discussions and calculations of diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be given in Chapter 11, "Radiation Protection Program and Waste Management."

Areas of Review

The reviewer should evaluate the following:

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

Acceptance Criteria

The acceptance criteria for the information on the confinement and HVAC system ESFs include 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 (e.g., HVAC systems) does not interfere with normal operations or safe reactor shutdown.
- The ESF design features should ensure that the system is available and operable when it is required for mitigating accident consequences.
- 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 the accidents discussed at the beginning 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 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 instrumentation and control (I&C) system 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 the environment created by accident scenarios.
- The discussions should identify operational limits, design parameters, surveillances, and surveillance intervals that will be included in the technical specifications.

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 at the beginning of this chapter. The reviewer should examine all accident scenarios analyzed in Chapter 13 of the SAR that could lead to significant radiological exposures or releases and verify that consequences can be sufficiently mitigated by the confinement ESF. The reviewer should confirm that the design and functional bases of confinement ESFs are

derived from the accidents analyzed. The reviewer should 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.

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 staff has reviewed the designs and functional descriptions of the confinement ESF; 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 that do not exceed the applicable limits of 10 CFR Part 20 for research reactors, or 10 CFR Part 100 for test reactors, and in both cases are as far below the regulatory limits as can be reasonably achieved.

6.2.2 Containment

If the HVAC and any air exhaust or liquid release systems associated with the containment are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13 and thereby mitigate its consequences, they 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 is not required for normal operation or accident mitigation. However, the safety analyses 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 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 applicant's analyses may show that a confinement is an acceptable ESF, but the applicant chooses 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 releases to the environment will be controlled. The airborne radionuclide normally released from the envelope of the reactor is argon-41, which may be continuously swept from the reactor building to diffuse and disperse in the atmosphere. The applicant should ensure that during the controlled release, 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 the function of 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 given in Chapter 3 and Chapter 9 of the SAR. Discussions and calculations of diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be given in Chapter 11.

Areas of Review

The reviewer should evaluate the 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 it 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 given in SAR Chapters 3 and 9 and in Chapter 11 with regard to the diffusion and dispersion of airborne radioactivity in restricted and unrestricted environments.

The reviewer should evaluate the following:

- Design bases and functional description of the required mitigative features of the containment, derived from the accident scenarios.
- Drawings, schematic diagrams, and tables of important design and operating parameters and specifications for the containment, 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
 - 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 Chapters 4 and 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 the surveillance methods and intervals in the technical specifications that ensure operability and availability of the containment, when required.

Acceptance Criteria

The acceptance criteria for the information on the containment ESF include the following:

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

- The design that should reduce below regulatory limits the potential radiological exposures to the facility staff and members of the public for the accidents discussed at the beginning 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 containment 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 system of the containment ESF system should be as basic and fail safe as possible. They 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.

Review Procedures

The reviewer should 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 should be compared with the limits of 10 CFR Parts 20 or 100 to determine if the design is acceptable.

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 as a result of which projected exposures to the public without containment would be greater than acceptable limits.
- The design and functional features proposed for a containment reasonably ensure that exposures will be reduced below the limits of 10 CFR Part 20 for research reactors, or 10 CFR 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 systems, 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.

6.2.3 Emergency Core Cooling System

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. If coolant is accidentally lost, the decay heat in some non-power reactors could be high enough to require a core cooling system to avoid cladding and fuel damage from high fuel temperatures.

Each applicant should present in Chapter 13 of the SAR the analysis of a loss-of-coolant accident (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 reviewer should evaluate the design bases and functional requirements of the proposed ECCS for the postulated LOCA through the progression of the accident scenario.

Acceptance Criteria

The acceptance criteria for the information on the ECCS include 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 ECCS should not interfere with either normal operations or reactor shutdown.
- The consequences of the LOCA event, as mitigated by the ECCS, will not exceed the limits of 10 CFR Part 20 for research reactors or 10 CFR Part 100 for test reactors 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.

Review Procedures

The reviewer should evaluate the accidents in Chapter 13 of the SAR to determine the scenario and consequences for the LOCA and to ascertain 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 the format and content guide. The reviewer should compare the design details of the proposed ECCS with the design and functional requirements of the SAR LOCA and the mitigated radiological consequences with 10 CFR Part 20 for research reactors or 10 CFR Part 100 for test reactors to determine if the design is acceptable.

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 analysis 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 ECCS would not interfere with normal operations and would not prevent safe reactor shutdown.
- 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.
- The design of the ECCS is adequate for operation at 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 for alternative 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.

6.3 References

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

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

7 INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation and control (I&C) systems comprise the sensors, electronic circuitry, displays, and actuating devices that provide the information and the means to safely control the reactor and to avoid or mitigate accidents. Instruments are provided to monitor, indicate, and record such operating parameters as neutron flux density, fuel temperature, coolant flow, temperature, and level; and radiation intensities in selected areas around the reactor. Certain I&C systems will automatically shut down (scram) the reactor when any safety parameter reaches a predetermined setpoint as analyzed in the SAR. I&C subsystems may also be designed to actuate engineered safety features (ESFs) upon the detection of abnormal conditions.

The I&C systems of non-power reactors comprise two basic subsystems:

- (1) the reactor control system (RCS), interlocks, control console instruments, and radiation monitoring systems necessary and sufficient to operate the reactor under the full range of normal conditions
- (2) the safety systems [reactor protection system (RPS), ESF actuation system, and radiation safety monitors] added to the I&C systems because of such events as possible accidents, malfunctions, operator error, or release of radioactive material (some components may be a part of both subsystems)

The RPS would be designed to be independent from the RCS if the risks associated with operating a non-power reactor were large. However, non-power reactors can be designed and operated so they pose an acceptably small or insignificant risk to the facility staff, the public, and the environment. Such a facility need not have an RPS independent in all respects from the I&C systems used for normal operations. Most licensed non-power reactors have been designed on the basis of these principles, and the reviewer should anticipate I&C system designs in which subsystems for normal operation and safety subsystems are intermingled. However, the applicant should justify the design of these combined systems and should clearly distinguish and discuss the two functions, noting which components serve both purposes. The consequences of certain malfunctions of the I&C system may render this design approach unacceptable for high-power test reactors. These cases should be handled individually by the project manager and NRC I&C system experts.

The format and content guide suggests that I&C subsystems and equipment be categorized by the function performed: RCS, RPS, ESF actuation system, control console and display instrument, or radiation monitoring instrument. The applicant should completely identify the I&C systems in each category. Identification should

include such attributes as name, type, function, analog or digital, purpose, and any other distinguishing characteristics.

The I&C system gives the operator information with which to control both the mode of operation and neutron flux (power) level of the reactor. It may also give input to the RCS, allowing changes in reactivity and automatic control of the power level of the reactor by insertion or withdrawal of control rods. Startup is accomplished only by manual control for most non-power reactor designs.

The safety systems (RPS and ESF actuation system) monitor such parameters as neutron flux, fuel temperature, area radiation intensities, and other important parameters to scram the reactor when deemed necessary or to initiate the operation of ESF systems when instruments indicate certain conditions have been met.

The control console and other display instruments present current and past operating parameter and system status information for use in evaluating reactor operating conditions. This information enables the operator to decide on further action, such as when to take manual control of the reactor.

Radiation protection instruments monitor radiation intensities in selected areas that may be occupied in or near the reactor building, or may supply input to the RPS or the ESF actuation system, and may monitor the concentrations or the release of radioactive material in effluent streams from the reactor facility. This information can be used to assess or control personnel radiation exposures.

7.1 Summary Description

Each I&C system for a non-power reactor should be designed to perform functions commensurate with the complexity of the particular facility. Reviewers should anticipate wide variations in design capability and functions of the I&C systems because of the wide variations in such factors as operating thermal power levels and use of non-power reactors. The format and content guide recommends that the SAR should include a summary description of the I&C system: the safety, philosophy, and objectives of its design; the operational characteristics of the reactor that determine or limit the I&C design; and the ways in which the various subsystems constitute the whole and interact to contribute to its essential functions. The format and content guide describes information that should be included in this summary, such as block, logic, and flow diagrams illustrating the various subsystems. The summary description may compare the reactor-specific I&C design with similar ones that NRC has found acceptable for other non-power reactors, including the bases for redundancy and diversity of sensor channels, safety channels, and control elements. The acceptance of the summary description

should be based on its completeness in addressing the factors listed in the format and content guide.

7.2 Design of Instrumentation and Control Systems

In this section of the format and content guide, the staff discusses various topics that the applicant should include in this chapter of the SAR. The reviewer should confirm that this type of information is in the SAR for each of the I&C systems in its entirety and for each category of subsystem. The SAR should address the following:

- design criteria
- design bases
- system description
- system performance analysis
- conclusion

The remaining sections of this chapter discuss specific information to be included in the SAR for each of the subsystems and how the reviewer should evaluate each subsystem.

7.3 Reactor Control System

Areas of Review

The RCS contains most of the I&C subsystems and components designed for the full range of normal reactor operation. The areas of review for the RCS should include a discussion of the factors requested in Section 7.2 of the format and content guide. The information for the RCS may be presented under the following subtopics:

- nuclear instruments—including all detector channels designed to monitor or measure nuclear radiations, and possibly fuel temperature within the reactor for operational purposes
- process instruments—instruments designed to measure and display such parameters as coolant flow, temperature, or level; fuel temperature; or air flow parameters within or from the reactor room
- control elements—types, number, function, design, and operating features of reactivity control devices other than fuel elements (coordinate with the review of Chapter 4, "Reactor Description")

- interlocks—circuits or devices to inhibit or prevent an action, such as control rod motion, unless a specified precondition exists. Interlocks are intended to protect personnel or other subsystems from harm.

The areas of review for the RCS should also include the following:

- bases, criteria, standards, and guidelines used for the design of the RCS
- description, including logic, schematics, and functional diagrams, of the overall system and component subsystems
- analysis of the adequacy of the design to establish conformance to the design bases and criteria for reactor power, rate of power change, and pulsing information
- analysis of the adequacy of the design to establish conformance to the design bases and criteria for information on required process variables to control reactor operation
- application of the functional design and analyses to the development of bases of technical specifications, including surveillance tests and intervals
- RCS failure modes to determine if any malfunction of the RCS could prevent the RPS from performing its safety function, or could prevent safe shutdown of the reactor.

Acceptance Criteria

The acceptance criteria together with the use of good engineering practice will help the reviewer to conclude whether the RCS is designed to provide for the reliable control of reactor power level, rate of change of power levels, and pulsing (if applicable) during reactor startup, the full range of normal operation, and shutdown. Acceptance criteria include the following:

- The range of operation of sensor (detector) channels should be sufficient to cover the expected range of variation of the monitored variable during normal and transient (pulsing or square wave) reactor operation.
- The RCS should give continuous indication of the neutron flux from subcritical source multiplication level through the licensed maximum power range. This continuous indication should ensure about one decade of overlap in indication is maintained while observation is transferred from one detector channel to another.

- The sensitivity of each sensor channel should be commensurate with the precision and accuracy to which knowledge of the variable measured is required for the control of the reactor.
- The system should give reliable reactor power level and rate-of-change information from detectors or sensors that directly monitor the neutron flux.
- The system should give reliable information about the status and magnitude of process variables necessary for the full range of normal reactor operation.
- The system should be designed with sufficient control of reactivity for all required reactor operations including pulsing, and ensures compliance with analyzed requirements on excess reactivity and shutdown margins.
- The RCS should not be designed to fail or operate in a mode that would prevent the RPS from performing its designed function, or prevent safe reactor shutdown.
- Hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers Systems in Safety Systems of Nuclear Power Generating Stations," and Regulatory Guide (RG) 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," Revision 1, which is attached to Chapter 7 of the format and content guide as Appendix 7.1, and software should meet the guidelines of ANSI/ANS 10.4-1987, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," that apply to non-power reactor systems.

ANSI/ANS 15.15-1978, "Criteria for the Reactor Safety Systems of Research Reactors," and ANSI/ANS 15.20 (draft), "Criteria for the Control and Safety Systems for Research Reactors," are general guides for the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable. A digital control system developed by General Atomics has been reviewed by the staff, found acceptable, and installed in several NRC-licensed TRIGA reactors (see Amendment No. 19 to Facility Operating License No. R-84, Docket No. 50-170 for the Armed Forces Radiobiology Research Institute TRIGA reactor, July 23, 1990, and Amendment No. 29 to Facility Operating License No. R-38, Docket No. 50-89 for the General Atomics TRIGA Mark I Reactor, October 4, 1990). A digital control system developed by Atomic Energy Canada Limited has been reviewed by the staff, found

acceptable, and installed in an NRC-licensed TRIGA reactor (see Amendment No. 30 to Facility Operating License No. R-2, Docket No. 50-5 for the Penn State Breazeale Reactor, August 6, 1991).

- For I&C systems that are being upgraded to systems based on digital technology, the applicant should consult NRC Generic Letter 95-02, "Use of NUMARC/EPRI Report TR-102348, Guideline on Licensing Digital Upgrades, in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59."
- The RCS should be designed for reliable operation in the normal range of environmental conditions anticipated within the facility.
- The RCS should be designed to assume a safe state on loss of electrical power.
- The subsystems and equipment of the RCS should be readily tested and capable of being accurately calibrated.
- Technical specifications, including surveillance tests and intervals, should be based on SAR analyses and should give the necessary confidence in availability and reliable operation of detection channels and control elements and devices.
- If required by the SAR analysis, the system should give a reactor period or a startup rate indication that covers subcritical neutron multiplication, the approach to critical, through critical, into the operating power range.
- The RCS should give redundant reactor power level indication through the licensed power range.
- The location and sensitivity of at least one reactor startup channel, along with the location and emission rate of the neutron startup source, should be designed to ensure that changes in reactivity will be reliably indicated even with the reactor shut down (see Chapter 4).
- A startup channel with interlock should give indication of neutrons and should prevent reactor startup (increase in reactivity) without sufficient neutrons in the core.
- The startup and low-power range detectors should be capable of discriminating against strong gamma radiation, such as that present after long periods of operation at full power, to ensure that changes in neutron flux density are reliably measured.

- At least one neutron flux measuring channel should give reliable readings to a predetermined power level. For reactors with power as a safety limit, the measurable power level should be above the safety limit. For reactors without power as a safety limit, the measurable power level should be high enough to show that the basis for limiting licensed power level is not exceeded.
- The automatic and manual control element absorber, drive, and display systems should be designed to limit reactor periods and power oscillations and levels to values found acceptable in the reactor dynamic analyses in Chapter 4 of the SAR, and rod and driver positions should be clearly indicated for operator or interlock use.
- For reactors designed for pulsing or "square-wave" operation, the transient rod and its driver mechanism, interlocks, mode switching, detector channels, other related instruments, and limiting technical specifications should be designed for the highest possible reliability to ensure that analyzed fuel safety limits will not be exceeded, and personnel hazards will be controlled. Designs should be compared with such systems accepted by NRC for similar operations or reactors.
- The applicant should plan and discuss how all control elements, their driver and release devices, and display or interlock components will be calibrated, inspected, and tested periodically to ensure operability as analyzed in the SAR.
- The applicant should describe in the SAR interlocks to limit personnel hazards or prevent damage to systems during the full range of normal operations. Interlocks on such systems as the following should be described, including provisions for testing and bypassing, if shown to be acceptable: transient rod drives; power level or reactor period recorders; startup neutron counter; gang operation of control elements; coolant flow or temperature conditions; beam ports, thermal column access, irradiation chambers, pneumatic or hydraulic irradiators, high radiation areas; confinement or containment systems; experiment arrangements and beam lines; or special annunciator or information systems. Interaction with the RPS, if applicable, should be described.
- If analyses of an experiment or experimental facility could show hazard to itself or the reactor, direct interacting or interlocking with reactor controls may be justified. Any such automatic limiting devices should demonstrate that function of the RPS will not be compromised, or safe reactor shutdown will not be prevented (see Chapter 10, "Experimental Facilities and Utilization").

Review Procedures

This chapter of the SAR should describe the I&C subsystems that apply to all normal functions and parameters of the entire reactor facility; these subsystems constitute the RCS. The reviewer should confirm that I&C information for all normal functions and systems described in other chapters of the SAR is addressed in this section.

The RCS comprises several subsystems; therefore, the reviewer should anticipate that the information in the SAR will be further subdivided, as noted in the section on the areas of review. The subdivisions should address all of the factors listed in Section 7.2 for each subsystem and should state how and where the subsystems interact and interface functions as a total RCS for normal operations. The reviewer should verify that all design bases are justified, and that the designs themselves accurately and completely implement the applicable bases and acceptance criteria. The reviewer should obtain the assistance of experts in the I&C Branch to review computer systems

Evaluation Findings

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

- The applicant has analyzed the normal operating characteristics of the reactor facility, including thermal steady-state power levels, pulsing capability (*if included*), and the planned reactor uses. The applicant has also analyzed the functions of the reactor control system (RCS) and components designed to permit and support normal reactor operations, and confirms that the RCS and its subsystems and components will give all necessary information to the operator or to automatic devices to maintain planned control for the full range of normal reactor operations.
- The components and devices of the RCS are designed to sense all parameters necessary for facility operation with acceptable accuracy and reliability, to transmit the information with high accuracy in a timely fashion, and control devices are designed for compatibility with the analyzed dynamic characteristics of the reactor.
- The applicant has ensured sufficient interlocks, redundancy, and diversity of subsystems to avoid total loss of operating information and control, to limit hazards to personnel, and to ensure compatibility among operating subsystems and components in the event of single isolated malfunctions of equipment.

- The RCS was designed so that any single malfunction in its components, either analog or digital, would not prevent the reactor protection systems from performing necessary functions, or would not prevent safe shutdown of the reactor.
- Discussions of testing, checking, and calibration provisions, and the bases of technical specifications including surveillance tests and intervals give reasonable confidence that the RCS will function as designed.
- The applicant has evaluated descriptions of planned interlocks or feedback controls from experimental apparatus to decrease postulated deleterious effects on the reactor. This review was coordinated with the effort for Chapters 10, "Experimental Facilities and Utilization," and 13, "Accident Analyses," and with Section 7.4, "Reactor Protection System." Furthermore, the design bases for such interlocks for future (not fully planned) experiments have been reviewed. The designs and design bases of the RCS give reasonable assurance that experiments will be planned and accomplished with due regard for protection of the reactor.

7.4 Reactor Protection System

In this section, the applicant should thoroughly discuss and describe the RPS, listing the protective functions performed by the RPS, and the parameters monitored to detect the need for protective action. The principal action designed for the RPS is to rapidly place the reactor in a subcritical condition by automatically inserting the control and safety rods whenever any of the selected parameters exceeds predetermined limits, in order to prevent reactor operation in regions in which fuel damage events could occur. The automatic insertion may also be initiated manually by the operator. Parameters typically monitored for this purpose include core neutron flux, fuel temperature (primarily in TRIGA designs), primary coolant flow and temperature, coolant level or radioactivity, and reactor area radiation levels. Redundant and diverse channels should normally monitor these parameters so that a single failure or malfunction cannot disable the protective function. As noted previously, unless analyses in the SAR require it, the RPS and the RCS need not be isolated and independent for non-power reactors. The objective of this review is to confirm that the RPS is designed to perform the safety functions stated in the SAR.

Areas of Review

In evaluating this system, the reviewer should include the following: sensors, signal handling equipment, isolation devices, bistable components, logic matrices, computer hardware and software, bypasses and interlocks associated with the trip and control circuitry, power supplies, and actuation devices that are designed to

initiate automatic reactor shutdown or runback. The reviewer should examine how the RPS automatically initiates rapid operation of the reactivity control devices to verify that reactor design limits analyzed in the SAR are not exceeded. The SAR should contain the information recommended in the format and content guide, such as:

- Design bases, acceptance criteria, and guidelines used for design of the RPS.
- Descriptive information, including system logic and schematic diagrams, showing all instruments, computer hardware and software, electrical, and electromechanical equipment used in detecting reactor conditions requiring scram or other reactor protective action and in initiating the action
- Analysis of adequacy of the design to perform the functions necessary to ensure reactor safety, and its conformance to the design bases, acceptance criteria, and the guidelines used.
- Assessment of the suitability of detector channels for initiating reactor protection (scrams). The reviewer should coordinate this effort with the review of other SAR sections
- Proposed trip setpoints, time delays, accuracy requirements, and actuated equipment response to verify that the RPS is consistent with the SAR analyses of safety limits, limiting safety system settings (LSSS), and limiting conditions of operation (LCOs), and that this information is adequately included in the technical specifications as discussed in Chapter 14, "Technical Specifications."
- Computer hardware, software, and software verification and validation programs for reactor designs that use computerized protection sub-systems.
- Verification that surveillance tests and intervals give confidence that the equipment will reliably perform its safety function. Coordinate this effort with the review of the technical specifications.
- Consideration of the SAR analyses for the RPS to be designed to perform its safety function after a single failure and to meet requirements for seismic and environmental qualification, redundancy, diversity, and independence.

Acceptance Criteria

Most non-power reactors can be designed and operated with an acceptable small or insignificant radiological risk to the public or to the environment. The SAR should address the separation and independence of the RCS and the RPS with consideration of the radiological risk of reactor operation, because these systems include most of the same types of subsystems and components and similar functions. If the safety analysis in the SAR shows that safe reactor operation and safe shutdown would not be compromised by combination of the two systems, they need not be separate, independent, or isolated from each other. In practice, the reactor protection function for non-power reactors has been reliably accomplished by adding an automatic trip and rod release subsystem to the RCS or adding safety channels. Since many licensed non-power reactors have been designed on that principle, this section of the review guidance is based on its continuing applicability and acceptability.

The acceptance criteria for the RPS should include the following:

- The design bases for the protection function should be provided.
- Detector channels and control elements should be redundant to ensure that a single random failure or malfunction in the RCS or RPS could not prevent the RPS from performing its intended function, or prevent safe reactor shutdown.
- The logic, schematic, and circuit diagrams should be included and should show independence of detector channels and trip circuits.
- The RPS is sufficiently distinct in function from the RCS that its unique safety features can be readily tested, verified, and calibrated.
- Technical specifications, including surveillance tests and intervals, should be based on discussions and analyses in the SAR of required safety functions.
- The reactor should have operable protection capability in all operating modes and conditions, as analyzed in the SAR. For example, at low reactor power, a reactor period scram may be needed to ensure that inadvertent transients could not propagate risks to personnel or the reactor.
- The range of operation of sensor (detector) channels should be sufficient to cover the expected range of variation of the monitored variable during normal and transient (pulsing or square wave) reactor operation.

- The sensitivity of each sensor channel should be commensurate with the precision and accuracy to which knowledge of the variable measured is required for the protective function.
- The automatic reactor runback or shutdown (scram) subsystem should be fail-safe against malfunction and electrical power failure, should be as close to passive as can be reasonably achieved, should go to completion once initiated, and should go to completion within the time scale derived from applicable analyses in the SAR.
- The RPS should be designed for reliable operation in the normal range of environmental conditions anticipated within the facility.
- The scram operator should be able to operate the system by means of readily available switches, or by interlock activation.
- The scram system should be designed to annunciate the channel initiating the action, and to require resetting to resume operation.
- The scram system should be designed to maintain reactor shutdown without operator action to at least the shutdown margin as defined in Chapter 4 and the technical specifications.
- The RPS function and time scale should be readily tested to ensure operability of at least minimum protection for all reactor operations.
- Information about the RPS detector or sensor devices should be sufficient to verify that individual safety limits are protected by independent channels, and that LSSS and LCO settings can be established through analyses and verified experimentally.
- Hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993 and RG 1.152, Revision 1, and software should meet the guidelines of ANSI/ANS 10.4-1987 applicable to non-power reactor systems. ANSI/ANS 15.15-1978 and draft ANSI/ANS 15.20 are useful as general guides for the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable.
- Consult NRC Generic Letter 95-02 for I&C systems that are being upgraded to systems based on digital technology.

Review Procedures

The reviewer should compare the design bases for the RPS with SAR analyses of possible hazards to the reactor or to personnel that could be prevented or mitigated by timely protective action. The RPS design and planned functional operation should be compared with the design bases and with the acceptance criteria for this section. The review should be sufficiently detailed to allow assessment of the complexity of the RPS and evaluation of opportunities for malfunction or operability failure during reactor operation. The reviewer should compare the RPS logic and design features with acceptable systems on similar reactors whose operating history is available. The reviewer should obtain the assistance of experts in the I&C Branch to review computer systems.

Evaluation Findings

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

- The applicant has analyzed the design and operating principle of the reactor protection system (RPS) for the (*insert name of facility*). The protection channels and protective responses are sufficient to ensure that no safety limit, limiting safety system setting, or RPS-related limiting condition of operation discussed and analyzed in the SAR will be exceeded.
- The design reasonably ensures that the design bases can be achieved, the system will be built of high-quality components using accepted engineering and industrial practices, and the system can be readily tested and maintained in the designed operating condition.
- The RPS design is sufficient to provide for all isolation and independence from other reactor subsystems required by SAR analyses to avoid malfunctions or failures caused by the other systems.
- The RPS is designed to maintain function or to achieve safe reactor shutdown in the event of a single random malfunction within the system.
- The RPS is designed to prevent or mitigate hazards to the reactor or escape of radiation, so that the full range of normal operations poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

7.5 Engineered Safety Features Actuation Systems

Non-power reactors can generally be designed and operated so they pose an acceptably small or insignificant radiological risk to the public. If the SAR analyses show that no unacceptable radiation doses would result from any postulated accident, even without consequence mitigation, such a facility need not include ESFs. The reviewer should, therefore, study Chapter 6, "Engineered Safety Features," and Chapter 13 of the SAR to determine if there is a requirement for ESFs and their related ESF actuation system. The guidance in this section applies to any non-power reactor for which an ESF is required.

Areas of Review

The reviewer should evaluate the information in Chapter 6 describing the ESFs, the scenarios of the postulated accidents in Chapter 13 which involve the use of an ESF, and the detector channels that sense the need for mitigation of possible consequences. The information to be reviewed in this section should also include the design criteria of each ESF actuation system, and the design bases and functional requirements for the ESF actuation systems. Additional information for review should include details of the design and operating characteristics of the actuation systems such as the following:

- logic and schematic diagrams
- description of instruments, computer hardware and software, electromechanical components, detector channels, trip devices and set points
- discussion of the bases of technical specifications, including surveillance tests and intervals that are designed to ensure operability

Acceptance Criteria

Acceptance criteria for ESF actuation systems should include the following considerations:

- The engineering design of ESF actuation systems and the components procured for them should be of high quality to ensure reliable operation. This quality is essential because these systems are designed to mitigate the consequences of postulated accidents.
- The ESF actuation system should be designed not to fail or operate in a mode that would prevent the RPS from performing its designed function, or prevent safe reactor shutdown.

- The ESF actuation system should be designed to assume a safe state on loss of electrical power.
- The range and sensitivity of ESF actuation system sensors should be sufficient to ensure timely and accurate signals to the actuation devices.
- The equipment should be designed to operate reliably in the ambient environment until the accident has been brought to a stable condition.
- The equipment should be designed to be readily tested and calibrated to ensure operability.
- Technical specifications including surveillance tests and intervals should ensure availability and operability of the ESF actuation system.
- ESF actuation systems should be designed to be operable whenever an accident could happen for which the SAR shows consequence mitigation is necessary.
- Hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993 and RG 1.152, Revision 1, and software should meet the guidelines of ANSI/ANS 10.4-1987 that apply to non-power reactor systems. ANSI/ANS 15.15-1978 and draft ANSI/ANS 15.20 are general guides for the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable.
- Consult NRC Generic Letter 95-02 for I&C systems that are being upgraded to systems based on digital technology.

Review Procedures

The reviewer should compare the design criteria and bases of the ESF actuation system with the designs of the ESFs and the accident scenarios and possible consequences. The reviewer should also compare the design and functional descriptions of the ESF actuation system with the acceptance criteria and with applicable criteria and functions discussed in Chapters 6 and 13. The reviewer should obtain the assistance of experts in the I&C Branch to review computer systems.

Evaluation Findings

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

- The applicant has analyzed the scenarios for all postulated accidents at the facility, including all accidents for which consequence mitigation by engineered safety features (ESFs) is required or planned. The staff evaluated the ESFs and has determined that the designs of their actuation systems give reasonable assurance of reliable operation if required.
- The applicant has considered the environments in which the ESFs are expected to operate, and the applicable actuation systems have been designed accordingly to function as required.
- The design considerations of the ESF actuation system give reasonable assurance that the system will detect changes in measured parameters as designed and will initiate timely actuation of the applicable ESF.
- The bases for technical specifications, including surveillance tests and intervals for the ESF actuating system, give reasonable assurance of actuation of ESFs when required.

7.6 Control Console and Display Instruments

Areas of Review

The non-power reactor control room, containing the control console and other status display instruments is the hub for reactor facility operation. It is the location to which all information necessary and sufficient for safe and effective operation of the facility is transmitted, and the primary location from which control and safety devices are actuated either manually or automatically. The console contains most of the circuitry and hardware for organizing and processing the information either analytically or digitally, applying decision logic, and routing signals to display devices or automatic action of other subsystems or both. The reviewer should evaluate the control console and display instruments to determine that the following are included:

- signals from instrument systems monitoring the reactor and other system process variables
- analytically or digitally processed outputs based on monitored variables
- indication of RCS or RPS status
- recording of selected variables and operating data
- annunciators and alarms

- personnel and equipment protection interlock status
- inputs to the RCS or RPS
- analog or computer hardware and software that manages the combination and presentation of reactor and process variable information for the operators

An objective of this review is to evaluate whether displays and operator control systems are designed and located to promote ease and efficiency in the performance of operations necessary for the safe control of the reactor. The information should include the following:

- design criteria, bases, and guidelines used to design the control console and information display system
- descriptive information such as logic, functional control and schematic diagrams, and equipment location drawings showing interrelationships in the control console
- analysis of the adequacy of the design to perform the necessary control and protection actuation, and information management, storage, and display functions
- coordination with review of other SAR chapters to verify control inputs and displayed parameters apply for the systems involved
- coordination with technical specifications review to verify that appropriate surveillance tests and intervals are specified to ensure that the instruments and equipment will perform their functions as designed

Acceptance Criteria

Acceptance criteria for the control console and display instruments should be based on good engineering practice and should include the following considerations:

- The control console instruments and display systems should be designed to work with applicable systems, either analog or digital computers
- A control console instrument system failure should not prevent the RPS from performing its safety function and should not prevent safe reactor shutdown.

- The control console, display instruments, and equipment should be readily testable and capable of being accurately calibrated.
- The designed range of operation of each device should be sufficient for the expected range of variation of monitored variables under conditions of operation.
- When required by the safety analysis, the control console instruments and equipment should be designed to assume a safe state on loss of electrical power or should have a reliable source of emergency power sufficient to sustain operation of specific devices.
- The bases for technical specifications, including surveillance tests and intervals for control console devices, should be discussed in this section of the SAR
- The outputs and display devices showing reactor nuclear status should be readily observable by the operator while positioned at the reactor control and manual protection systems.
- Control, safety, and transient rod position indication and limit lights should be displayed on the console and should be readily accessible and understandable to the reactor operator.
- Other controls and displays of important parameters that the operator should monitor to keep parameters within a limiting value, and those which can affect the reactivity of the core should be readily accessible and understandable to the reactor operator.
- Annunciators or alarms on the control console should clearly show the status of systems such as operating systems, interlocks, experiment installations, pneumatic rabbit insertions, ESF initiation, radiation fields and concentration, and confinement or containment status.
- Hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993 and RG 1.152, Revision 1, and software should meet the guidelines of ANSI/ANS 10.4-1987 that apply to non-power reactor systems. ANSI/ANS 15.15-1978 and draft ANSI/ANS 15.20 are general guides for the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable. The reviewer should consult Generic Letter 95-02 for guidance on I&C systems that are being upgraded to systems based on digital technology.

- Reactor operation should be prevented and not authorized without use of a key or combination input at the control console.

Review Procedures

The reviewer should coordinate the review of this section with all other applicable chapters of the SAR because the control console and other display instruments in the reactor control room could be linked to numerous systems and subsystems in the reactor facility. The reviewer should compare the design bases and functional requirements of other reactor systems with those for the control console equipment. The reviewer should also compare the design of the console system with the acceptance criteria. The reviewer should study the arrangement of parameter displays, control devices, and the planned operator station to determine whether the operator can quickly understand information and take proper action. The reviewer should obtain the assistance of experts in the I&C Branch to review computer systems. The reviewer should note the discussion in the format and content guide about digital operator information display systems or operator aids.

Evaluation Findings

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

- The applicant has shown that all nuclear and process parameters important to safe and effective operation of the (*insert name of facility*) non-power reactor will be displayed at the control console. The display devices for these parameters are easily understood and readily observable by an operator positioned at the reactor controls. The control console design and operator interface are sufficient to promote safe reactor operation.
- The output instruments and the controls in the control console have been designed to provide for checking operability, inserting test signals, performing calibrations, and verifying trip settings. The availability and use of these features will ensure that the console devices and subsystems will operate as designed.
- The annunciator and alarm panels on the control console give assurance of the operability of systems important to adequate and safe reactor operation, even if the console does not include a parameter display
- The locking system on the control console reasonably ensures that the reactor facility will not be operated by unauthorized personnel.

7.7 Radiation Monitoring Systems

Areas of Review

In this section of the SAR, the applicant should address all equipment, devices, and systems used for monitoring or measuring radiation intensities or radioactivity, except for nuclear instruments. Information in this section should be reviewed in close coordination with those sections of Chapter 11, "Radiation Protection Program and Waste Management," that discuss the use of radiation-monitoring systems to assess, evaluate, or control personnel or environmental radiological exposures. Chapter 11 should include sufficient information about the radiation-monitoring systems to support confident use of the exposure and dose results and this section should detail the operating principles, designs, and functional performance of the I&C aspects of the system. Radiation measurements at a reactor facility may also be used for reactor diagnostic or safety purposes, and the applicable equipment should be discussed in this section. Examples of such functions may include reactor coolant level, coolant radioactivity, fuel inventory measurements for self-protection, confinement or containment initiation, and experimental measurements.

The reviewer should evaluate radiation detectors and sampling equipment; signal processing equipment; computer hardware and software that controls sampling, detection, signal processing and logic; power supplies; and actuation systems that accomplish a function for the system. In determining if the I&C systems are designed to accomplish the radiation measurement functions, the reviewer should evaluate the following:

- design bases, criteria, and guidelines used to design the system
- descriptive information including functional operation, instrument logic and schematic diagrams
- analysis of the adequacy of the design to perform the stated function or purpose of the systems and conformance to the design bases, criteria, and guidelines used
- proposed trip, annunciation, or alarm setpoints, time delays, accuracy requirements, and actuated equipment response to verify that they are consistent with applicable analyses and limiting conditions for operation in the SAR
- coordination with review of other applicable SAR chapters to assess the suitability of the monitored parameters for accomplishing the purposes

- coordination with applicable technical specifications review to verify that surveillance tests and intervals are specified to give confidence that the system and equipment will be operable and reliably perform its function
- consideration of the need for single failure protection, seismic and environmental qualification protection, and diversity.

Acceptance Criteria

Acceptance criteria for radiation monitoring systems should include the following:

- The systems should be designed to interface with either analog or digital computerized RCS or RPS if applicable.
- The systems should be designed not to fail or operate in a mode that would prevent the RPS from performing its safety function, or prevent safe reactor shutdown.
- The systems and equipment should be readily tested and capable of being accurately calibrated.
- The systems and equipment should be designed for reliable operation in the environment in which they will function.
- The instrument ranges should be sufficient to cover the expected range of variation of the monitored variable under the full range of normal operation and if assumed in the SAR analysis, accident conditions.
- The sensitivity of each system should be commensurate with the precision and accuracy to which knowledge of the variable is required by analysis or design basis.
- The bases of technical specifications, including surveillance tests and intervals, should be sufficient to ensure that the systems will be operable and will perform their designed functions.
- Hardware and software for computerized systems should meet the guidelines of IEEE 7-4.3.2-1993 and RG 1.152, Revision 1, and software should meet the guidelines of ANSI/ANS 10.4-1987 that apply to non-power reactor systems. ANSI/ANS 15.15-1978 and draft ANSI/ANS 15.20 are general guides for the design, implementation of I&C systems for non-power reactors and should be used where applicable. The reviewer should consult NRC Generic Letter 95-02 for additional guidance on I&C systems that are being upgraded to systems based on digital technology.

Review Procedures

The reviewer should confirm that the design bases for the radiation monitoring systems and equipment I&Cs are consistent with giving reliable indication of the presence of radiation or release of radioactive material in the various areas monitored and in the monitored effluent streams from the reactor. The reviewer should establish that the design includes sufficient monitoring systems and equipment to perform the functions discussed elsewhere in the SAR. The reviewer should compare the equipment and designs and functions with the design bases and acceptance criteria. The reviewer should obtain the assistance of experts in the I&C Branch to review computer systems.

Evaluation Findings

This section of the SAR should include sufficient information to support the following types of conclusions, which will be included in the staff's safety evaluation report (the second and third conclusions may be presented in Section 11.1.4):

- (1) The designs and operating principles of the instrumentation and control of the radiation detectors and monitors have been described, and have been shown to be applicable to the anticipated sources of radiation.
- (2) The staff found that the SAR discusses all likely radiation and radioactive sources anticipated at the (*insert name of facility*) and describes equipment, systems, and devices that will give reasonable assurance that all such sources will be identified and accurately evaluated.
- (3) The radiation monitoring systems described in the SAR give reasonable assurance that dose rates and effluents at the facility will be acceptably detected, and that the health and safety of the facility staff, the environment, and the public will be acceptably protected.

7.8 References

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

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

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

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

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

U.S. Nuclear Regulatory Commission, "Use of NUMARC/EPRI Report TR-102348, 'Guideline on Licensing Digital Upgrades, in Determining the Acceptability of Performing Analog-to-Digital Replacements Under 10 CFR 50.59'," Generic Letter 95-02, April 26, 1995.

8 ELECTRICAL POWER SYSTEMS

In this chapter of the SAR, the applicant discusses and describes the electrical power systems at a non-power reactor facility 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 ensure that, given a loss of normal electric service, sufficient power will be available for mitigating the events discussed in SAR Chapter 13, "Accident Analysis." The 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, "Radiation Protection Program and Waste Management"; and Chapter 13. Design and functional information in Chapter 8 should be provided under the two categories: normal and emergency electrical power systems.

8.1 Normal Electrical Power Systems

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 design should use 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 8.1 of the standard format and content guide.

Acceptance Criteria

The acceptance criteria for the information on normal electrical power systems at non-power reactor facilities include 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 instrumentation 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.

Review Procedures

The reviewer should (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.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will 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 shut down.
- The design and location of the electrical wiring will prevent inadvertent electromagnetic interference between the electrical power service and safety-related instrumentation and control circuits.

- The design of normal electrical systems gives 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.

8.2 Emergency Electrical Power Systems

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 reviewer 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.

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 reviewer should focus on the safety-related features of any emergency electrical power systems.

In other SAR chapters, the applicant presents the calculated responses 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 reviewer should also evaluate events of lesser consequences and the emergency electrical power system design that mitigates them.

Acceptance Criteria

The acceptance criteria for the information on emergency electrical power systems at non-power reactors include 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 switchover.
- 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, should include surveillance and testing, and should 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.

Review Procedures

The reviewer should (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 no emergency electrical system is 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.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will 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

This chapter contains guidance for evaluating the information on auxiliary systems in the reactor facility. Auxiliary systems are those systems not fully described in other chapters of the SAR that are important to the safe operation and shutdown of the reactor and to the protection of the health and safety of the public, the facility staff, and the environment. 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 their design and functions. 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 applicant should discuss 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. Any functions of auxiliary systems required during analyzed reactor accidents also should be discussed. The information the applicant should provide in this chapter of the SAR for each auxiliary system is given at the beginning of Chapter 9 of the format and content guide. The typical auxiliary systems listed there are not intended to be a complete list of auxiliary systems to be discussed in this chapter of the SAR. The reviewer should be aware that some auxiliary systems could be discussed in more than one chapter. The following sections contain guidance pertaining to the five items listed at the beginning of Chapter 9 of the format and content guide for the systems discussed in Sections 9.1 to 9.7 of the guide.

9.1 Heating, Ventilation, and Air Conditioning Systems

Areas of Review

At non-power reactors, the heating, ventilation, and air conditioning (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 system operating characteristics for the full range of reactor operation. In many non-power reactors, the HVAC systems are 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," of the SAR. 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 should be developed.

Areas of review should include the following:

- 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 of the way 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 designed to 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
- applicable technical specifications and their bases, including testing and surveillance

Acceptance Criteria

The acceptance criteria for the information on the HVAC systems include 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 occupational doses do not 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 airflow 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 system should ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur.
- The analyses of operations of the system 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 the facility ALARA program guidelines. The exposure analyses should be given in detail in Chapter 11 of the SAR.
- If design bases of the 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 Analyses," of the SAR, the function should be described in Chapter 6.
- Required technical specifications and their bases should ensure system operability.

Review Procedures

Using the five items listed at the beginning of Chapter 9 of the format and content guide, the reviewer should evaluate the submittal for all operations and functions of the HVAC systems during the full range of reactor operations. The design bases should be compared with requirements from other chapters of the SAR, especially Chapters 4 ("Reactor Description"), 6, 7 ("Instrumentation and Control Systems"), 11, and 13. The reviewer should determine whether the HVAC system designs agree with all acceptance criteria for the full range of reactor operations.

Evaluation Findings

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

- A review of the design bases and functional and safety characteristics of the HVAC systems shows 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.

- The applicant has discussed all sources of radioactive material that could become airborne in the reactor room from the full range of reactor operations. The analyses demonstrate that the radioactive material is controlled by the HVAC system and could not inadvertently escape from the reactor room. They 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 reactor operations, no potential occupational exposures would exceed the design bases derived in Chapter 11.
- The applicant has considered the height and flow rate of the stack that exhausts facility air to the unrestricted environment 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 containment (*confinement*) system and analysis of its operation ensure that it 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.
- The applicant has proposed technical specifications, including testing and surveillance, that will provide reasonable assurance of necessary HVAC system operability for the full range of reactor operations.

9.2 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 the common security. Protecting the fuel from malfunction or failure should be discussed in many chapters in the SAR.

Areas of Review

The reviewer should evaluate the handling, protection, and storage of the fuel when it is not in the reactor core, both before it is inserted and after it is removed.

Areas of review should include the following:

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

- Methods for inspection and verification of new fuel to ensure that procurement specifications have been met.
- 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, if applicable, should be reviewed, 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. The cooling of stored irradiated fuel should be discussed in detail in Chapter 5, "Reactor Coolant Systems," of the SAR, if the system is integral to that function. Otherwise, it should be described in this section.)
- 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. The reviewer should note that the applicant may be discussing events many years in the future and that some degree of uncertainty may exist.)
- Technical specifications that define controls on fuel during handling and storage, including testing and surveillance.

Acceptance Criteria

The acceptance criteria for the information on the handling and storage of reactor fuel include the following:

- The design of all systems, components, and methods for handling, moving, or storing fuel outside 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 should 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, the 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 the facility staff and the public are protected from radiation and that 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 highly enriched uranium (HEU) fuel, the applicant 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.

Review Procedures

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

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the staff's 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 meet procurement specifications.
- The 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 (*or 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 applicant has to adhere to 10 CFR 70.24).
- 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 the staff.
- Methods for shielding, cooling, and storing irradiated fuel give reasonable assurance of the following:
 - 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.
- Provisions to ensure compliance with 10 CFR 73.6(b), self-protection for HEU fuel, are acceptable (*if self-protection for HEU fuel is applicable*).

9.3 Fire Protection Systems and Programs

Areas of Review

Areas of review should include the following:

- brief discussion of potential causes and consequences of fires at the facility
- discussion of fire protection plans and protective equipment used to limit the consequences of a fire, including defense in depth in the event of escalation of a fire
- list of the objectives of the fire protection program, as well as the discussion of the organizations, methods, and equipment for attaining the objectives
- all passive designs or protective barriers planned to limit fire consequences, including features of the facility that could affect a safe reactor shutdown or release radioactive material in the event of a continuing fire
- the source of facility fire protection brigades and their training and the summary of the more detailed discussions of these personnel and offsite fire protection forces in the facility emergency plan
- compliance with local and national fire and building codes applicable to fire protection

Acceptance Criteria

The acceptance criteria for the information on the fire protection systems and programs include the following:

- The fire protection plan should discuss the prevention of fires, including limiting the types and quantities of combustible materials.
- Methods to detect, control, and extinguish fires should be stated in the plan.
- The facility should be designed and protective systems should exist to ensure a safe reactor shutdown and prevent the uncontrolled release of radioactive material if a fire should occur.

Review Procedures

The reviewer should evaluate the discussions of potential fires; provisions for early detection, including during those 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, the staff, and the environment if firefighting efforts are unsuccessful.

Evaluation Findings

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

- The plans for preventing fires ensure that the facility meets local and national 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 will 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 exceeded the requirements of 10 CFR Part 20.
- Acceptable technical specifications related to fire protection have been proposed and justified (*if applicable*). These technical specifications include acceptable requirements for testing and surveillance to ensure operability of fire detection and protection equipment.

9.4 Communication Systems

Areas of Review

Areas of review should include the following:

- the methods of communication between all necessary locations during the full range of reactor operations
- the summary of emergency communications that are discussed in detail in the physical security and emergency plans and evaluated by the staff
- the way two-way communication is provided between the reactor control room and other locations in the reactor facility

Acceptance Criteria

The acceptance criteria for the information on communication systems include the following:

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

Review Procedures

The reviewer should determine where discussions address the five items listed at the beginning of Chapter 9 of the format and content guide to formulate conclusions about the adequacy of the communication systems.

Evaluation Findings

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

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

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

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 specifies 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 at the reactor for research and development purposes. Byproduct materials may be used at the licensee facilities or shipped off site 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 materials and facilities are covered. These facilities should be described in the SAR as auxiliary reactor systems.

Areas of review should include the following:

- 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

- the provisions for controlling and disposing of radioactive wastes, including special drains for liquids and chemicals, and air exhaust hoods for airborne materials, with design bases derived in Chapter 11 of the SAR
- the 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
- required technical specifications and their bases, including testing and surveillance

Acceptance Criteria

The acceptance criteria for information on the possession and use of byproduct, source, and special nuclear material under the 10 CFR Part 50 license include the following:

- The design of spaces and equipment and the procedures should ensure that no uncontrolled release of radioactive materials (solid, liquid, or airborne) from the facilities can occur.
- 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 should ensure that only radioactive byproducts produced by the reactor are permitted, unless specifically authorized by the 10 CFR Part 50 license or an additional license.
- The facilities should be addressed specifically in the emergency plan, physical security plan, and fire protection provisions, as applicable.
- The proposed technical specifications covering these auxiliary facilities should 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.

Review Procedures

The reviewer should evaluate the five items listed at the beginning of Chapter 9 of the format and content guide for auxiliary systems and facilities that possess or use byproduct material produced in the reactor, source material, or special nuclear material, as allowed by the 10 CFR Part 50 license.

The reviewer should compare the design bases for systems and procedures with the requirements developed in other chapters of the SAR, especially Chapters 11 and 12, "Conduct of Operations"; should evaluate the design features against experience with possession and use of radioactive materials at other facilities and laboratories; and evaluate agreement with the acceptance criteria.

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

Evaluation Findings

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

- Auxiliary facilities and systems are designed for the possession and use of byproduct materials produced by the reactor and *(if applicable)* source and 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 applicant has described the authorized spaces for use of the material.
- 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 byproduct material produced by the reactor and (*if applicable*) source and special nuclear material and the auxiliary facilities where this material is stored or used is covered by the emergency plan, physical security plan, and fire protection provisions (*as applicable*).
- Technical specifications are proposed that will ensure that the use and possession of byproduct material produced by the reactor and (*if applicable*) source and special nuclear material and the use of the auxiliary facilities where this material is stored or used will not endanger the health and safety of the public, users, or the environment.

9.6 Cover Gas Control in Closed Primary Coolant Systems

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. At some of these reactors, heavy water (D_2O) functions as a moderator, reflector, or coolant. The heavy water must be protected to prevent the admixture of atmospheric water vapor and the loss of the heavy water. At other non-power reactors, primary systems 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.

Areas of review should include the following:

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

Acceptance Criteria

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

- The systems should be designed to perform the design-bases functions.
- The systems 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 could not be compromised by the system.
- Cover gases should be processed, recombined, or stored in such a way that the safety of the reactor and personnel is ensured.
- Technical specifications and their bases, including surveillances, should be provided as required to ensure system operability.

Review Procedures

The reviewer should evaluate the five items listed at the beginning of Chapter 9 of the format and content guide for auxiliary systems that provide and control cover gas for non-power reactors with closed primary coolant systems. The design should be compared with the design bases and with the acceptance criteria.

Evaluation Findings

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

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

9.7 Other Auxiliary Systems

The auxiliary systems addressed in the previous sections are typical examples of systems found at non-power reactor facilities. As noted at the beginning of this chapter, 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.

Areas of Review

The reviewer should determine whether the applicant has addressed the five items listed at the beginning of Chapter 9 of the format and content guide.

Acceptance Criteria

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

- The design and functional description of the auxiliary system 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 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.

Review Procedures

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

Evaluation Findings

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

- The system has been designed to perform the functions required by the design bases.
- Functions and potential malfunctions that could affect reactor operations have been considered in the design of the system. 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.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

This chapter contains guidance for evaluating the information on the experimental facilities at the reactor, their use, and associated safety considerations. The applicant should provide sufficient information in the SAR to demonstrate that no proposed operations involving experimental facilities would result in unacceptable radiological risk to reactor operations personnel, experimenters, or the general public.

The guidance in this chapter is consistent with Regulatory Guides 2.2 and 2.4. The reviewer should be familiar with these documents.

Non-power reactors have many experimental, educational, and service uses. The experimental facilities may penetrate the reactor core or reflector, be located near the core, or be an integral part of the reactor. Samples can be irradiated in the core or the reflector, or neutron or other radiation beams can be extracted from the core region through the biological shield.

Utility, integrity, longevity, versatility, diversity, and safety should be considered for the experimental facilities in the same manner they are considered for the reactor core and its operational components and systems. The safety analyses of the reactor facility should include the experimental facilities and their interactions with the core and its other reactor systems. If changes in reactor operating characteristics are proposed, the reviewer should check to see that potential interactions between the core and the experimental facilities are analyzed as appropriate.

The reviewer will probably evaluate SARs in which the analyzed safety envelope for experiments and experimental facilities is very broad and the technical specifications are performance based so that the applicant can take maximum advantage of the requirements of 10 CFR 50.59 for changes in the experimental program and facilities.

Some non-power reactors are operated as critical facilities to demonstrate fuel loading and perform reactor physics studies. In such cases, the reactor itself should also be considered an experimental facility. The reviewer will find that this section of the SAR and the experimental technical specifications will focus on allowable core configurations and reactor physics constraints for operating entire or partial cores as experiments.

This chapter of the SAR should contain an analysis demonstrating that the reactor and experimental facilities can be operated safely during normal operations and

accidents including malfunctions of experimental facilities. The applicant should also analyze the possibility of the experimental facility causing a malfunction of the reactor systems. The analysis should support the requirement that there be no undue risk to the health and safety of the public. In some cases, most notably with fueled experiments, the failure of the experiment can be the maximum hypothetical accident (MHA) for the reactor facility. Also, experiment failures could result in the maximum reactivity addition accident. Experiment failures and consequences should be analyzed in Chapter 13, "Accident Analyses," of the SAR.

The reviewer should examine the design bases, facility descriptions, functional and safety analyses, and the applicant's safety conclusions for all experimental facilities. For those experimental facilities that are permanently attached to the reactor support structure, reactor vessel, or pool hardware, this chapter of the SAR should contain an analysis of the structural design and potential impact on reactor operation. For those experimental facilities that penetrate the reactor vessel below any primary coolant water level, an analysis of the experimental design should demonstrate that the design is resistant to failure and that if failure occurs, it is considered or bounded by the analysis in Chapter 13 for a loss-of-coolant accident (LOCA). The placement or utilization of experimental facilities shall not compromise the functionality of any reactor safety systems or engineered safety features. The applicant should discuss the capabilities, limitations, and controls on reactor operation, including engineering or procedural controls for experiments, that ensure personnel radiation doses do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility program for maintaining exposure to radiation as low as is reasonably achievable (ALARA).

Because of the potentially unlimited variety of experiments that can be performed in a non-power reactor, it is important that administrative controls are adequate to ensure that the health and safety of the public are protected. Not all of the actual experiments to be performed need be discussed in detail in this chapter of the SAR, but the limiting and enveloping features of the experiments and the administrative procedures used by the applicant to safely review, approve, and control experiments should be described. The applicant should provide the bases for the experiment-related technical specification limiting conditions for operation (LCOs) and a detailed description and justification of the experiment review and acceptance program that are then specified in the technical specifications.

10.1 Summary Description

In this section of the SAR, the applicant should briefly describe the principal features of the experimental and irradiation facilities associated with the reactor. The reviewer should ensure that the applicant has discussed the scope of the experimental program and defined what is considered to be an experiment. Some applicants consider operation to conduct surveillances to be an experiment. This is

acceptable as long as this is clearly defined. Discussions should include experimental compatibility with normal reactor operations and show how interference with safe reactor shutdown and adequate fuel cooling is avoided.

Areas of review should include the following:

- general focus of the experimental program (radiation science, medical, materials testing, teaching, etc.)
- experimental facilities
- basic type of experiments that will be irradiated (incore, thermal column, external beam, etc)
- limiting experimental characteristics (e.g., reactivity, contents)
- monitoring and control of the experiments and the interaction between the experiment and the reactor control and safety systems
- design requirements for the experiment and the review and approval process

The applicant should include a discussion and analysis of all proposed experimental facilities. Typical experimental facilities are described briefly in the format and content guide.

The applicant may use simple block diagrams and drawings to show the location, basic function, and relationship of each experimental facility to the remainder of the reactor. The summary description should give the reviewer enough information to gain an overall understanding of the functions of the experimental facilities and the experiment review and approval process.

10.2 Experimental Facilities

Areas of Review

Experimental facility design requires choosing among several interdependent variables. In the final design, the applicant should strive to meet the experimental requirements for optimum neutron or gamma fluxes, while ensuring that risks to the public, the staff, and the experimenters are acceptable

Areas of review should include the following:

- Design of the experimental facility: the facility design aspects to contain or withstand any postulated pressure pulse, temperature change, or reactivity effect to preclude any inadvertent primary coolant leakage or facility collapse; facility materials of fabrication and compatibility of materials with projected radiation and chemistry requirements; physical size, including all dimensions; and simplified engineering drawings or schematics, if used, especially for more complex facilities.
- Incore/in-reflector experimental facilities, including important design and operating parameters and specifications: location of the facility in relation to the core, safety systems, core support, neutron detectors, coolant system components, and other in-pool structures; and reactivity effects of these facilities.
- Features of the experimental facility that prevent interference with safe reactor shutdown or with adequate core cooling.
- Radiological considerations associated with the design and use of the experimental facilities: generation of radioactive gases (including argon-41), release of fission products or other radioactive contaminants, and exposure of personnel to neutron and gamma beams, in relation to any facility-related technical specifications; direct radiation streaming from the experimental facilities and the effect of scattered (sky shine) radiation; any radiation monitors specifically designed and placed to detect experiment radiation and to monitor personnel; and any physical restraints, shields, beam catchers or beam stops, temporary and permanently installed, used to restrict access to radiation areas associated with experimental facilities.
- Experiment safety system and the functional interface between this system and the reactor protection system: the experimental facility safety instrumentation, including the location and function of sensors, readout devices, and scram or interlock capabilities, and all permanently installed instrumentation and control systems.
- The need for experiment cooling and the source of coolant and any dependence on or interaction with the reactor coolant system. The reviewer should refer to Chapter 5, "Reactor Coolant Systems," of this review plan.
- Proposed technical specifications for the experimental facilities.

Acceptance Criteria

The acceptance criteria for the information on the experimental facilities include the following:

- Safety limits (SLs), limiting safety system settings (LSSSs), and LCOs for the reactor should be derived from analyses in this and other chapters and included in the reactor facility technical specifications. The analysis in this section of the SAR should show that the components and the functional design of the experimental facilities will ensure that no reactor LSSS or LCO will be exceeded during normal operations of these experimental facilities and that no SL will be exceeded during accident conditions. Consideration should include the full range of reactivity, thermal power level, fuel temperature, pressure, and pool depth above the core.
- The physical dimensions should be such that the volume is not sufficient to produce more of a positive reactivity insertion if suddenly voided or flooded than that analyzed in Chapter 13 of the SAR. The experimental facility should not skew the neutron flux distribution sufficiently to mask safety signals or peak the neutron flux or power densities beyond values analyzed in other applicable chapters.
- The thickness of the walls and the strength of the welds of the experimental facility should preclude any deformation that would result from the maximum postulated pressure, pulse or steady, that could damage surrounding fuel, compromise the integrity of the experimental facility, or interfere with the reactor safety or control system. The applicant should thoroughly investigate the chemical and physical properties of the materials, including vulnerability to corrosion, erosion, and oxidation, compatibility, and strength, for the interior and exterior environments of the experimental facility.
- The mechanical means of securing the experimental facility should provide assurance that the facility cannot move inadvertently. In general, positive fastening methods and materials, such as welds, bolts, sleeves, and collars, are required.
- With regard to the shielding of, and control of access to, experimental facility areas, this section of the SAR should contain descriptions and analysis showing that the placement, dimensions, and materials are sufficient to limit the expected radiation dose to experimenters, reactor operators, and other personnel to levels that are below those required by 10 CFR Part 20 and that are consistent with the facility ALARA program. The analysis should show all pertinent radiation sources, distances,

dimensions, materials, angles of reflection, and material attenuation factors. Exposure to radioactivity produced by experimental operations should be consistent with the analysis in Chapter 11, "Radiation Protection Program and Waste Management," of the SAR. The potential exposure from malfunction or failures of an experimental facility must be within the values analyzed in Chapter 13. Special radiation detectors used for experiments should be discussed in Chapters 7, "Instrumentation and Control Systems," or 11 of the SAR or in the experiment protocol.

- For most non-power reactors, any required experiment or facility cooling might be accomplished through convection heat transfer to the surrounding reactor moderator or coolant water. The thermal-hydraulic analysis should demonstrate that experimental facility cooling is designed to prevent failure of the facility under all operating conditions of the reactor or the experiments. Any interdependence of the experiment cooling and reactor coolant systems should not compromise adequate fuel cooling or prevent safe reactor shutdown.

For any experimental facilities that require a special cooling system independent from the reactor primary coolant system, the technical evaluation considerations should be basically the same as those for the reactor coolant system. The applicable guidance in Chapter 5 of this review plan should be followed.

- At many non-power reactors, beam ports penetrate the reactor vessel below the water level of the pool and in most of those cases below the top of the reactor core. Therefore, the LOCA analysis in Chapter 13 of the SAR is very important. The applicant should show that the design of experimental facilities does not introduce new mechanisms to initiate a LOCA, or that the potential consequences of a LOCA caused by the failure of experimental facilities are considered acceptable by the Chapter 13 analysis. The applicant should show that fuel integrity would not be lost as a result of a LOCA initiated or affected by the design of an experimental facility or by the malfunction of an experiment or experimental facility.
- With regard to the experimental facility safety systems and the functional interface between these systems and the reactor protection system, the applicant should demonstrate that in the event of any credible malfunction in the experimental facility, the design offers reasonable assurance that the safety system will be capable of protecting the reactor, the experiment, and the health and safety of the public.
- For any large-volume irradiation facilities, such as an exposure room or dry chamber, an acceptable design should include provisions for (1) preventing

reactor operation if personnel are in the irradiated volume, (2) controlling airborne radioactive materials, (3) maintaining acceptable biological shielding in occupied areas, (4) limiting effects on reactivity due to changes of experiments within the irradiated volume to values found acceptable, and (5) when applicable, automatically shutting down the reactor if the reactor or shielding is moved during operation.

Review Procedures

The reviewer should evaluate the following:

- The purposes and projected uses of the reactor facility for experimental programs.
- The experimental facilities and the other related reactor systems to determine if they are designed with comparable engineering expertise and emphasis on safety.
- The failure and malfunction modes of experimental facilities and consequences to the reactor, the staff, the experimenters, and the public.
- The analyses of potential hazards to the reactor, users, and public from experimental facilities during normal operations and accidents and the production and release of radioactive materials, including argon-41. The reviewer can evaluate effluents from experimental facilities as part of the review of Chapter 11 of the SAR.
- The analyses of how postulated reactor accidents affect the experimental facilities and their experiment contents.

Evaluation Findings

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

- Planned operation and utilization of the experimental facility will not exceed the limiting safety system setting or limiting condition for operation of the technical specification requirements for the reactor facility during normal operations. The applicant has proposed and justified acceptable technical specifications for the experimental facilities.
- The design and functional information in the SAR gives reasonable assurance that the experimental facility is capable of retaining necessary

integrity during all anticipated operations and postulated accidents and is secured appropriately. The design also accounts for the location of the facility with respect to pool water level, and the LOCA analyses include the effect of the failure of the facility on the consequences of the LOCA.

- The reactivity insertion by rapid flooding, voiding, or other malfunction of an experimental facility is limited to acceptable limits. Reactor behavior resulting from rapid reactivity insertions of this limit does not exceed acceptable conditions.
- Radiological controls ensure that personnel and public radiation doses do not exceed the requirements in 10 CFR Part 20 and are consistent with the facility ALARA program.
- The experiment instrumentation and control systems can perform their design functions. Interfaces between the experiment control systems and the reactor protection system protect the reactor, the experiment, and the public.
- Consequences of the malfunction or failure of an experimental system are considered in the analyses of reactor accidents in Chapter 13 of the SAR.

10.3 Experiment Review

Because of the variety of experiments that can be performed in a non-power reactor, the applicant's administrative controls should be adequate to ensure the protection of the public. The administrative procedures used by the applicant to approve an experiment should be discussed in detail in Chapter 10 of the SAR, summarized in Chapter 12, "Conduct of Operations," and included in the technical specifications. The applicant should state the requirements for the experiment safety analysis and the experiment review and approval methodology and discuss the experiment review and approval process.

Areas of Review

The reviewer should verify that the applicant has presented in the SAR a methodology and procedures to review experiments thoroughly and to grant approval.

Areas of review should include the following:

- review committee composition, review and approval criteria, and approval authority

- experiment design requirements and classification methodology
- administrative controls, including appropriate review and approval of safety analysis in accordance with 10 CFR 50.59 for experiments not included in the SAR
- allowed and possible capsule materials, experimental materials, and structural composition and types
- radiation heating and other radiation damage
- malfunction or failure modes of experimental facilities and experiments, including radiological hazards and controls

Acceptance Criteria

The acceptance criteria for the information on experiment review include the following:

- The experiment safety review committee should consist of individuals with extensive experience in reactor operations, radiation protection, conduct of experiments, and the mechanical, electrical, radiological, and chemical behavior of materials in the operational environment. A functional organization chart should be provided showing in detail the composition and lines of communication of the committee with the reactor operations and administration staffs. The committee should be organizationally independent and operate without interference from the experimenters or reactor operations staff. The committee should function under a charter, and criteria for experiment approval should be provided, including application of 10 CFR 50.59.
- The applicant should describe the methodology that will be used to categorize proposed experiments according to risk potential. The types of categories used at the reactor facility should be stated and requirements for each category listed. The appropriate level of review authority required to approve experiments in each category should be discussed. The information in this section of the SAR should be quantitative where possible (e.g., giving gram amounts, temperature degree limits, radioactivity limits, or reactivity limits) in delineating the bounds of the risk categories. Methodology for using 10 CFR 50.59 to review all new experiments should be described, as well as how Regulatory Guides 2.2 and 2.4 are used.

- The applicant should list the administrative controls used to protect facility personnel and the public from radiation or other possible hazards during the experimental program. Discussion should concentrate on the possibility that reactor and experiment operations may be conducted under separate authority and by different personnel. Areas of discussion should include access to experimental facilities and areas, lockout procedures, communications with reactor operating personnel, and alarms. The administrative controls should address basic protection and recovery procedures after a malfunction of experiments or experimental facilities.
- The applicant should discuss the method used to assess the experimental materials and any limitations. How Regulatory Guides 2.2 and 2.4 were considered should be discussed. The assessment of experimental materials should include the following:
 - radioactive materials, including fissile materials, and radiological risks from radiation fields or release of radioactive material
 - neutron absorption and activation in trace elements and impurities
 - effects on reactivity, both positive and negative
 - explosive, corrosive, and highly reactive chemicals
 - radiation-sensitive materials
 - flammable or toxic materials
 - cryogenic liquids
 - unknown materials
 - radiation heating or damage that could cause experiment malfunction

Review Procedures

The reviewer should evaluate the composition of the experiment safety review committee and the methodology used for reviewing and approving experiments. Reviewed should be the use of references and the content of applicable technical specifications. The reviewer should compare the program under review with comparable programs at other non-power reactor facilities.

Evaluation Findings

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

- The applicant has shown an independent organization for the experiment safety review committee, which has diverse and independent membership as well as acceptable experience and expertise.
- The procedures and methods used at the facility ensure a detailed review of all potential safety and radiological risks that an experiment may pose to the reactor facility and the public.
- The administrative controls are sufficient to protect the operations personnel, experimenters, and general public from radiation and other potential hazards caused by the experiments. The expected radiation doses do not exceed the limits of 10 CFR Part 20 and are consistent with the facility ALARA program.
- The administrative controls ensure that all proposed new or changed experiments will be reviewed in accordance with the requirements of 10 CFR 50.59.
- Technical specifications ensure acceptable implementation of the review and approval of experiments.

10.4 References

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," November 1973.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.4, "Review of Experiments for Research Reactors." May 1977.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter provides guidance for the review and evaluation of Chapter 11 of the applicant's SAR, which should contain information about radiation protection and radioactive waste management provisions at the facility. Information should include radiological design bases of the reactor structures, systems, components, experimental facilities, and laboratories under the reactor license; procedures, policies, and practices employed to ensure compliance with applicable standards and regulations on radiation doses and protection; procedures, policies, and practices to ensure that radioactive wastes are managed in compliance with applicable regulations and standards; and the program to keep radiation exposure at the facility as low as is reasonably achievable (ALARA). The responsibilities of the health physics organization at the reactor facility and of any other onsite radiation protection and radioactive waste management organizations should also be described. Throughout this chapter, the applicant should show that licensed activities will be conducted in compliance with applicable regulations, with emphasis on 10 CFR Part 20.

This chapter should address all radioactive materials and radiation sources that are produced in the reactor or used within the reactor facility and that are possessed under the authority of the reactor operating license. Radioactive standards, check sources, and other byproduct material used in the reactor program, reactor startup sources, fuel and other special nuclear material, and source material that may be under the authority of the reactor license should be included.

The complexity of reactor facilities will vary widely from one non-power reactor to another, as will the risks due to radiation. Furthermore, a non-power reactor facility may be only a small component of a large organization, such as a university or corporation, and could obtain its radiation protection and radioactive waste management services from other parts of the organization. Therefore, the scope and magnitude of the radiation protection and radioactive waste management programs should be expected to vary, and may be found acceptable as long as the program is consistent with a uniform requirement to adequately protect the health and safety of the public.

In some places in this chapter, reference is made to conservative best estimates or conservative but realistic calculations. This means that estimates or calculations performed by the applicant should always give results that are conservative. However, the applicant should try to avoid such large levels of conservatism that results are orders of magnitude from the expected true answer. In some cases, non-power reactor applicants have used assumptions or calculation methods that have produced very conservative, but acceptable results. Subsequently, regulatory

requirements changed or the applicant made changes to the facility or utilization program that resulted in these conservative results being unacceptable. At that point, the applicant had to perform an analysis with assumptions and calculational methods that were more realistic in order to demonstrate compliance with regulatory requirements, and also had to explain the conservatism in the original analysis.

In this chapter of the SAR, the applicant should discuss the capabilities of the reactor facility to control, collect, handle, process, store for short or long periods, and dispose of liquid, gaseous, and solid radioactive wastes related to reactor operations and utilization in a manner planned to protect the public, the environment, and facility staff. The instrumentation and methods used to monitor radiation exposures to personnel and the release of radioactive materials, including sampling methods, should be discussed by the applicant.

11.1 Radiation Protection

The provisions for radiation protection should be described completely in the sections that follow.

11.1.1 Radiation Sources

Areas of Review

To develop a comprehensive radiation protection program, it is important to understand all sources of radiation exposure at the facility. Therefore, the applicant should provide complete listings and discussion of all expected radiation and radioactive sources. The reviewer should evaluate information requested of the applicant in the corresponding section of the format and content guide. As indicated there, airborne, liquid, and solid sources, including radioactive wastes, should be discussed.

Acceptance Criteria

The acceptance criteria for information concerning sources of radiation include the following:

- All sources of radiation should be discussed by the applicant. This discussion should include the physical and chemical form, type (e.g., neutron, gamma), curie strength or exposure rates, energy level, encapsulation (sealed or unsealed), use, storage conditions and locations, and planned program for disposal of all radioactive material subject to the reactor license.

- The applicant should present the best estimates of the maximum annual dose and the collective doses for major radiological activities during the full range of normal operations for facility staff and members of the public. The doses shall be shown to be within the applicable limits of 10 CFR Part 20.
- Airborne radioactive material sources should be described in sufficient detail to provide the bases for the design and assessment of structures, systems, and components, and of personnel protective measures and dose commitments.
- Conservative best estimates of the predicted concentrations, locations, and quantities of airborne radionuclides during the full range of normal operation in areas occupied by personnel should be discussed.
- Conservative best estimates of the predicted locations and magnitude of external radiation fields during the full range of normal operation in areas occupied by or accessible to personnel should be discussed.
- The applicant should identify models and assumptions that are used for predicting and calculating the dose rates and accumulative doses from such radionuclides as argon-41 (Ar-41), nitrogen-16 (N-16), and vapors, aerosols, and airborne radioactive particulates in both restricted, controlled (if present), and unrestricted areas. The applicant should identify (1) locations of specific sources (e.g., coolant water, beam tubes, and gas- or air-driven rabbit systems), (2) expected production rates, release rates, and concentration distributions in occupied areas and resultant personnel doses or dose rates, and (3) release points from the control of the reactor facility, dilution air (quantities and sources), quantities and concentrations expected to be released, dispersion and diffusion, concentration at point of interest, applicable average atmospheric conditions, plume spread, expected concentration distributions in unrestricted areas, and applicable radiation dose rates, including gamma-ray shine from elevated plumes and inhaled or ingested dose commitments. The analysis should contain conservative best estimates of the predicted annual total doses to at least the following in the unrestricted areas: (1) the maximum exposed individual, (2) the nearest permanent residence, and (3) any location of special interest, such as a classroom or campus dormitory. The discussion and calculations should show that the sums of internal and external doses to the facility staff and the public are within the limits of 10 CFR Part 20 and that ALARA principles are applied.
- Liquid effluent volumes and radionuclide concentrations should be shown to be within the requirements of 10 CFR Part 20. The discussion should

include any dilution that occurs before or at the point of release to the unrestricted environment.

- Estimates should be given of solid radioactive waste curie content and volume. The methods used to process solid waste should be briefly discussed.

Review Procedures

The reviewer should determine that all sources of radiation in the facility are adequately discussed and that the specific topics discussed in the standard format and content guide are complete and sufficiently described. Some of this information may be verified during site visits associated with the licensing process, and some may be assessed by comparison with similar operating facilities that NRC has found acceptable.

If the applicant describes processes involving radioactive material, the reviewer should compare the description of the types of radioactive materials present with the applicable process description, including radionuclide inventories and mass balances and chemical and physical forms, to verify that all radioactive materials associated with the process have been identified.

For a licensed reactor that is in operation, material balance inventory forms (NRC/DOE Forms 741 and 742) can be reviewed to verify quantities and general locations of special nuclear material (SNM).

The reviewer should examine the description and discussion of all sources of radiation to verify that they are described in sufficient detail to provide the bases for the design and assessment of personnel protective measures and dose commitments. The reviewer should evaluate the models used to predict airborne and liquid radionuclide concentrations and the physical and chemical forms of the radionuclide inventories to verify that they are appropriate for the facility and process conditions. If radionuclide data (inhalation or ingestion exposure data or concentration and inventory data) are available for the applicant or for facilities with similar processes and configurations, the reviewer should compare the predicted liquid and airborne radionuclide concentration distributions and possible doses with measured exposure data to validate the conservatism of the best estimates of the radionuclide concentrations. To evaluate consistency, the reviewer should use the applicant's summary of the calculated doses resulting from radionuclides predicted or detected during normal operations in areas that could be occupied by facility staff and the public.

The reviewer should confirm that all solid sources of radiation at the facility are described and discussed in sufficient detail to permit evaluation of all significant

radiological exposures related to normal operation, utilization, maintenance, and radioactive waste management including processing and shipment. The reviewer should determine the origin of the radiation (e.g., the reactor core and radiation beams), predicted exposure, access control, provisions for source control and storage, and interim or ultimate disposition.

Evaluation Findings

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

- The description of potential radiation sources and associated doses including the inventories, chemical and physical forms, and locations of radioactive materials, and other facility radiation and operational parameters related to radiation safety presented in the SAR have been reviewed. This review included a comparison of the bases for identifying potential radiation safety hazards with the process and facility descriptions to verify that such hazards were accurately and comprehensively identified. This review and evaluation confirm that the SAR identifies the potential radiation safety hazards associated with (*insert name of facility*) and this provides an acceptable basis for the development and independent review of the radiation protection program.

11.1.2 Radiation Protection Program

Areas of Review

The reviewer should evaluate the design and effectiveness of the radiation protection program required by 10 CFR 20.1101 to determine that they include the following:

- the radiation protection program that implements the regulations to ensure compliance with the requirements for radiation protection
- organizational structure within which the applicant will administer the radiation protection program and ensure radiation protection, including staffing levels, positions of authority and responsibility, position qualifications, standards, charters, procedures, or other documents that specify the authority, duties, and responsibilities of the personnel in the organization
- interfaces and interrelationships of the radiation protection organization with other facility safety organizations and reactor facility operations

- policy governing the program and the allocation of policy-making responsibilities, including the administrative plans and procedures that implement the facility policy, and how the organization, policy, and program are designed for effective radiation protection
- plans and procedures for radiation protection, including the document control measures employed
- radiation protection training program, including the scope and content of training provided or required for all personnel, including facility operations and utilization personnel, health physics personnel, non-facility research and service personnel, security, fire, and other emergency personnel, and visitors
- purpose, organization, and functions of any committees with responsibilities relating to radiation protection, including each committee's charter of responsibilities, frequency of meetings, audit and review responsibilities, scope of any audit and reviews, qualifications and requirements for committee members, and relationship to other standing or ad hoc committees for radiation protection at the facility or within the parent organization
- the program for conducting facility radiation protection reviews and audits of all functional elements of the radiation protection program, including the scope of the reviews and audits, the basis for scheduling the review and audits, the qualifications of the auditors, and the process and office responsible for following up on audit findings
- the system for evaluating experience from the radiation protection program, including problems and incidents so that these experiences can be used to improve facility design and the radiation protection program and to develop "lessons learned," identify root causes, and implement effective corrective actions
- the radiation protection program recordkeeping process, including record retention periods, accessibility, review, and archiving, any special review of radiation safety records for accuracy and validity, and the use of records for developing trend analyses, informing management, planning radiation-related actions, and reporting to the regulatory authority and other duly authorized entities

Acceptance Criteria

Acceptance criteria for the radiation protection program include the following:

- The scope and content of the radiation protection program should be clearly based on a commitment of facility management to protect the facility staff, the environment, and the public from unacceptable exposure to radiation.
- The facility organization chart should show that the management of the radiation protection program is independent of the reactor operations management.
- The program should provide for compliance with all applicable regulations.
- The program should show clearly that all review areas (discussed above) have been addressed and all expected radiation doses have been addressed.
- The program procedures should establish and describe clear lines of responsibility and clear methods for radiation protection under normal and emergency conditions.
- Procedures should be organized and presented for convenient use by operators and technicians at the appropriate locations, and should be free of extraneous material. Supplementary or revised procedures should be issued when necessary to reflect radiation protection changes and improvements. Procedures should be periodically reviewed by supervisors and the review committee. New or revised procedures affecting radiation protection should be reviewed and approved by the radiation protection staff, appropriate management, and the review committee.
- All employees and visitors granted unescorted access to the facility should receive training concerning the radiological health and protection program, commensurate with their job duties and functions, or purposes of their visits. Certified individuals, including operators and their supervisors, should be given classroom and on-the-job training in radiation control practices. The radiation protection training program should be part of the ongoing training program established and maintained by the facility to train and requalify individuals as required.
- The review committee or committees responsible for radiation protection should report to a level of management sufficiently high to take any necessary corrective action; should have clearly written charters that describe their purposes, functions, authority, responsibility, and

composition, and quorum, meeting frequency, and reporting requirements; should maintain records of recommendations and subsequent actions in sufficient detail to permit reviews of its effectiveness; should have membership that is technically competent in the radiation protection disciplines within the committee's area(s) of responsibility; and should operate in a manner that provides for group discussions among members on all but the more routine matters.

- The committee or committees responsible for auditing the radiation protection program should audit all functional elements of the radiation protection program as often as necessary. The audits should be performed by individuals whose expertise covers the range of technical fields encountered in the audit. Audits should be performed by individuals who are not directly responsible for the activities audited. Audits should be performed in such areas as personnel external and internal dosimetry, portable and fixed instrumentation, respirators (if used by the facility), contamination control, radiological monitoring, the ALARA program, nuclear accident dosimetry, radiation source material control, radiological health and safety training, posting of radiological areas, and radiation protection program records.
- The facility should have a radiation work permit or similar program to control tasks with significant radiation hazards that are not described in the SAR.
- The radiation protection program records management system should include such records as ALARA program records, individual occupational dose records, monitoring and area control records, monitoring methods records, and training records.

Review Procedures

The reviewer should evaluate the responsibilities and authorities assigned to the radiation protection organization against the acceptance criteria. The reviewer should also confirm that individuals assigned specific radiation protection responsibilities have adequate and clearly defined authority to discharge these responsibilities effectively. The reviewer should evaluate whether the radiation protection organization has sufficient staff to discharge its assigned responsibilities, and should examine the interfaces and interrelationships with other facility safety organizations, including the reactor operating organization and the radioactive waste management organization if that responsibility is not part of the radiation protection organization. The reviewer should examine how responsibility is assigned to operations supervisors for the radiation protection of personnel under their control and how mechanisms are established to request and obtain necessary

assistance from the radiation protection organization. The reviewer should evaluate whether the administrative plans and procedures provide a framework for the radiation protection organization to discharge its responsibilities independently in an effective manner, including interdiction of perceived unsafe practices and communication with upper management to ensure that radiation protection issues are properly resolved. The use of procedures to carry out the radiation protection program should be examined by the reviewer. The reviewer should examine the radiation protection training program and the radiation protection review and audit committee. The reviewer should examine the description of the records management program for the radiation protection program. During the conduct of the review, the reviewer should consider the regulations, guides, standards, and staff reports (NUREGs) in the bibliography at the end of this chapter, as they apply to the non-power reactor facility. Please note that this list may not be complete and other documents may be available.

Evaluation Findings

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

- The staff has reviewed the radiation protection program presented in the SAR for the (*insert name of facility or operation*). This review included an evaluation of (1) the roles, responsibilities, authorities, organization, and staffing of the radiation protection organization; (2) the roles, responsibilities, authorities, staffing, and operation of committees responsible for the review and audit of the radiation protection program; (3) the effectiveness and comprehensiveness of the radiation protection training program; (4) radiation protection plans and information that form the bases of procedures and the management systems employed to establish and maintain them; (5) the effectiveness and comprehensiveness of the program for independent oversight reviews and audits of the radiation protection program; (6) the effectiveness and comprehensiveness of the process to evaluate the radiation protection program to improve the program and the process to examine problems and incidents at the facility, and (7) the management of records relating to the radiation protection program. This review confirms that the radiation protection program presented in the SAR both complies with applicable requirements and gives reasonable confidence that management's commitment to radiation protection in all activities will protect the facility staff, the environment, and the public from unacceptable exposure to radiation.

11.1.3 ALARA Program

Areas of Review

To evaluate the provisions at the facility for maintaining worker and public doses and radiological releases ALARA, the reviewer should verify that the applicant's submittal includes, but is not limited to, the following:

- A description of the ALARA program for the facility.
- A description of the methods to establish and change policy for this program, including the management level and authority by which the policy is established.
- A description of how this program is implemented for all activities at the facility to maintain radiation doses of all personnel and releases of effluents to the unrestricted area ALARA. The description should include criteria for considering economic factors in the ALARA analyses, for establishing ALARA goals, and for revising or terminating a proposed activity.

Acceptance Criteria

Acceptance criteria for information concerning the ALARA program include the following:

- The facility ALARA program should require that radiation doses received by facility staff and the public are maintained as low as is reasonably achievable, economic factors having been taken into account. The facility should have established ALARA goals.
- The highest levels of facility management should be committed to the ALARA program. (For universities, this commitment should come from the upper university administration.) The commitment should be shown by active management involvement in the program and should be clearly stated in writing to all personnel.
- Supervisory personnel should be required to periodically review exposure records for the personnel under their control to determine the trends and factors that contribute to personnel exposures and the methods for reducing exposure.
- Facility management should ensure that sufficient emphasis is placed on and sufficient resources are given to ALARA considerations during design, construction, and operation of facilities, in the planning and implementation

of reactor utilization, in maintenance activities, and in the management and disposition of radioactive wastes.

Review Procedures

The reviewer should determine that the facility ALARA program satisfies the acceptance criteria discussed above. The reviewer should evaluate the provisions of the ALARA policy and program to determine whether applicable requirements, including 10 CFR Part 20, are satisfied.

Evaluation Findings

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

- The staff has reviewed the ALARA program at the facility. The policies and the bases for procedures give reasonable assurance that doses to occupational workers and the public will be maintained below regulatory limits and ALARA. The controls and procedures for limiting access and personnel exposure (including allowable doses, effluent releases, ALARA goals and criteria used for the action levels in radiation alarm systems) meet the applicable radiation protection program requirements and provide reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The ALARA program is adequately supported at the highest levels of management for the facility.

11.1.4 Radiation Monitoring and Surveying

Areas of Review

The reviewer should evaluate the procedures and equipment at the facility for routinely monitoring and sampling workplaces and other accessible locations to identify and control potential sources of radiation exposure and release. The specific topics to be reviewed are discussed in this section of the format and content guide.

Acceptance Criteria

Acceptance criteria for information concerning radiation monitoring include the following:

- The procedures and equipment should be designed to ensure that air, liquids, solids, and reactor radiation beams and effluents are monitored and sampled as necessary.
- The bases of the methods and procedures used for detecting contaminated areas, materials, and components should be clearly stated.
- The bases of the methods and procedures used for monitoring exposures of personnel, including those working in radiation and high radiation areas, should be clearly stated.
- Records should be kept as required by the regulations in 10 CFR Part 20 to document the applicability, quality, and accuracy of monitoring and sampling methods, techniques, procedures, and results.
- A complete range of radiation monitoring and sampling equipment, appropriate to the facility, should be employed throughout the facility, including equipment employed by experimental and operations support personnel. The applicant should discuss the bases of procedures for selection, use, locations, and functions of each monitoring device, including but not limited to the following:
 - continuous air monitors (CAMs), including fixed and moving filter CAMs, and gaseous monitors
 - portable survey instruments
 - remote area monitors (RAMs)
 - effluent monitors
 - samplers
 - environmental monitors
 - contamination-monitoring equipment
 - personal dosimeters
 - portal monitors
 - radioactive waste storage and release monitors
 - criticality monitors

- The calibration of radiation protection instrumentation and procedures should be discussed, including the calibration equipment, procedures and standards governing calibration, control of the calibration process, associated sensitivities to environmental and other conditions, and verification of proper operation. The program should conform to recognized national standards to help ensure that radiation protection instrumentation will measure radiation accurately and will function as needed. The program should ensure that recommended and routine periodic calibrations will be performed on time.
- The applicant should discuss the routine radiation monitoring and sampling provisions at the facility, including the methods used to survey radioactive material releases and the methods used to verify that waste materials will be appropriately monitored and controlled.
- In coordination with the information presented in Chapter 7, "Instrumentation and Control Systems," the applicant should describe the instrumentation and control systems used for radiation monitoring purposes.
- In coordination with the information presented in Chapter 6, "Engineered Safety Features," the applicant should describe the interface between the radiation monitoring system and the engineered safety features.
- Listings of required equipment, limiting conditions for operation, and surveillance requirements as discussed in Chapter 14, "Technical Specifications," should be discussed and justified in this section.

Review Procedures

The reviewer should evaluate the design of the instrumentation systems used for both routine and special radiation monitoring and sampling to ensure compliance with the acceptance criteria. The reviewer should confirm that the applicant plans to position air sampling or monitoring equipment in the appropriate locations to measure airborne concentrations of radioactive material to which people are exposed. If the SAR shows that general area air sampling is not adequate to estimate worker intakes, a program of personal breathing-zone air sampling may be required, and the reviewer should evaluate its provisions for and applicability to the subject facility.

The reviewer should confirm that radiation monitoring and alarms, as described in the SAR, provide adequate warning and coverage and are of sufficient sensitivity to ensure that any significant increase in radiation exposure rates or concentration of airborne radioactive material within the restricted area, controlled area (if

present), or in the unrestricted area would be detected and would initiate appropriate annunciation or action. The reviewer should coordinate this review with the Chapter 7 review and should evaluate the design of the radiation instrumentation systems used for radiation monitoring and dosimetry to verify compliance with the acceptance criteria. The reviewer should also verify that these radiation monitors and alarm systems will be maintained, operated, calibrated, and subjected to surveillance in compliance with the appropriate standards and are addressed in the technical specifications.

The reviewer should confirm that the facility warning and annunciator systems are designed to alert personnel to a radiological hazard or abnormal condition in sufficient time to enable them to respond in a planned appropriate manner. The reviewer should also confirm that the interface between the radiation monitoring system and the engineered safety features (as discussed in Chapter 6) and the discussion of the radiation monitoring system in the emergency plan are appropriate.

Evaluation Findings

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

- The staff has reviewed the design of radiation monitoring and sampling provisions at the facility. The fixed and portable equipment used for radiation monitoring and sampling inside the facility is selected, located, calibrated, tested, and maintained in accordance with guidance contained in recognized national standards and the manufacturers' instructions, and with applicable regulations. The methods and bases of procedures used to determine the placement of the equipment, the circumstances under which the equipment is used, and the selection of the equipment function and sensitivity are appropriate to the facility, and give reasonable assurance that appropriate types of radiation in significant intensities will be detected, monitored, and sampled consistent with 10 CFR Part 20 requirements and the facility ALARA program

11.1.5 Radiation Exposure Control and Dosimetry

Areas of Review

The reviewer should verify that the applicant submitted (1) the design bases for the equipment and procedures utilized for controlling radiation exposures to personnel and releases of radioactive materials from the facility and (2) dosimetry and

methods used to assess exposure to radiation and radioactive materials. The topics to be reviewed are discussed in this section of the format and content guide.

Acceptance Criteria

Acceptance criteria for information concerning radiation exposure control and dosimetry include the following:

- The design of the facility (e.g., containment/confinement) should prevent uncontrolled radiation releases to the environment or to the work areas during normal operations.
- The design of entry control devices (e.g., alarms, signals, or locked entry ways) should alert workers to, or prevent unauthorized entry into, high radiation areas and very high radiation areas, as appropriate.
- The design bases of radiation shielding, ventilation, and remote handling and decontamination equipment should be planned so radiation doses are maintained ALARA and should be within the regulatory limits
- The personnel protective equipment and materials (e.g., self-contained airpaks) employed in the facility, the facility conditions for which this equipment should be employed, and any testing, calibration, and training required for their use, should be discussed and should be within the applicable regulations and standards.
- Acceptable radiation exposure and dose limits should be administratively established for all accessible locations of the facility, including the exposure limits established for facility personnel, non-facility research and service personnel, and visitors. Acceptable administrative exposure limits may also be established for other groups (e.g., embryos and fetuses, declared pregnant women, minors, and students) at the facility.
- The applicant should discuss the bases used for developing the ALARA radiation exposure limits and how they are enforced, including the plans and procedures for exposure control and dosimetry during the full range of normal operations and postulated accident conditions, rescue and recovery, and planned special exposures.
- Applicable dosimetry should be used for external radiation monitoring (e.g., whole body, extremities). The frequency of dosimeter readings and action levels should be appropriate, and the dosimetry chosen should be suitable for the radiation sources expected and observed. The applicant should appropriately consider allowances for measurement uncertainties in

the dosimetry program and the determination of exposure levels, and the standards for the issuance and the accuracy of self-reading personal dosimeters. Applicable and adequate methods should be used for determining internal doses.

- The applicant should maintain records to establish the conditions under which individuals were exposed to radiation, including the historical and current exposures to personnel and any associated trends (both individual and facility). Methods of maintaining records should be established to assist in planning radiation-related activities, implementing the ALARA program, reporting to appropriate regulatory agencies, and meeting the requirements of 10 CFR Part 20.

Review Procedures

The reviewer should examine the facility exposure control and dosimetry programs for both external exposures and internal exposures to facility personnel, the environment (if measured), and the public to confirm that plans and the bases of procedures for the control of external dose to workers and the public consider the following:

- equipment and equipment design
- shielding
- radiation monitors and alarms
- personnel protective equipment
- the dosimetry used for external radiation monitoring, including the frequency of dosimeter readings, action levels, and the suitability of the dosimetry chosen with respect to the radiation sources expected and observed at the facility

The reviewer should also verify that procedures for the control of internal exposure consider the following:

- equipment and equipment design
- engineered controls such as containment/confinement or ventilation systems
- personnel protective equipment

- radiation monitors, alarms and samplers (if used)
- bioassay methods, frequency, and action levels
- the models and methods used for internal dose evaluation

The reviewer should examine the engineered controls used to ensure radiation protection safety for each of the sources of radiation and radioactive material described in Section 11.1.1. Some systems (e.g., containment or confinement or ventilation system) may have been reviewed in other chapters of the SAR. Reference may be made here to those evaluations. The reviewer should confirm that radiation protection measures have been implemented for sources of radiation and radioactive material. The reviewer should evaluate the radiation safety controls to determine the following:

- The acceptance criteria are met.
- Radiation protection engineering controls (e.g., the provision of shielding, facility and equipment layout to limit activities in radiation areas, use of confinement or containment systems, design of ventilation systems to control the potential for contamination and control release of radioactive material, and provision of remote handling systems) have been used.
- There is evidence of a commitment to reduce radiation doses to levels that are ALARA. The SAR should adequately justify any use of administrative controls instead of engineered controls.

The reviewer should confirm that the radiation dose limits and bases are identified and the plans and programs to control doses are documented. The reviewer should examine the descriptions of facility exposure conditions and methods used to derive administrative radiation dose limits. The reviewer should verify that dose limits and bases consider all groups (including, e.g., embryos and fetuses, declared pregnant women, minors, and students). The reviewer should examine the bases used for developing these limits and how the limits are enforced.

The reviewer should evaluate how the radiation protection controls provide assurance of the following:

- The acceptance criteria contained in this section of the review plan are met.
- Radiation protection engineered controls (e.g., the provisions of shielding, ventilation systems, remote handling systems) have been designed to reduce the potential for uncontrolled exposure or release and have been incorporated in the facility.

- There is evidence of a commitment to maintain radiation doses ALARA.

The reviewer should examine how records are kept to establish the conditions under which individuals were exposed to radiation. For facilities with an operating history, the reviewer should also look for trends. Records of historical and current doses to personnel should be consistent with 10 CFR Part 20.

Evaluation Findings

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

- The engineered radiation exposure controls employed at the facility have been reviewed. The applicant has given sufficient information about the design of the confinement (containment), radiological shielding, ventilation, remote handling, decontamination equipment, and entry control devices to allow for an assessment of the design of these radiological protection features. The entry control devices employed are adequate to alert workers to, or prevent entry into, radiological areas, including high or very high radiation areas. The confinement (containment) system design provides reasonable assurance that uncontrolled radiological releases to the unrestricted environment, controlled area (if present), or the restricted work area will not occur during any anticipated normal operations.
- The applicant has discussed the procedures for use of personal dosimetry at the facility. Provisions have been made for external and internal radiation monitoring of all individuals required to be monitored. The proposed dosimetry program meets the requirements of the regulations in 10 CFR Part 20.
- The provisions incorporated for personal dosimetry, shielding, ventilation, remote handling, and decontamination equipment provide reasonable assurances that radiation doses are maintained ALARA and within applicable regulations.

11.1.6 Contamination Control

Areas of Review

At a non-power reactor facility, controlling the occurrence and spread of radioactive surface contamination is important for a number of reasons. Unplanned and unwanted radioactive material could contaminate and interfere with or invalidate the results of experiments or other radiation measurements performed

as part of the utilization program. Unsuspected radioactivity in the restricted area could inadvertently be transported or "tracked" to the unrestricted area, and thereby constitute an uncontrolled release of radioactive material. Finally, removable or fixed surface contamination in the restricted area of sufficient source strength could measurably impact the radiological health and safety of people working there. The reviewer should evaluate how the applicant's program for contamination control meets all applicable requirements of the regulations and the facility ALARA program. The specific areas of review should include all of the items listed in the format and content guide. For existing programs, information about the effectiveness of the program should also be reviewed.

Acceptance Criteria

Acceptance criteria for the information on contamination control include the following:

- The scope of the program should demonstrate that the applicant understands the potential problems caused by radioactive contamination and recognizes that the best way to control it is to establish procedures to prevent it initially.
- The bases of procedures should show that routine monitoring of locations, equipment, and personnel for contamination will be established and maintained.
- The bases of procedures should show that no materials, equipment, or personnel will be permitted to leave an area known to be or suspected of being contaminated without being appropriately monitored.
- The contamination control program should include provisions to avoid, prevent, and remedy the occurrence and the spread of contamination.
- Explicit contamination control training should be established as part of comprehensive radiation protection and radioactive waste management training, as needed.
- The contamination control program should include provisions for recordkeeping in accordance with 10 CFR Part 20 regarding occurrence and spread of contamination, sufficient in content and retention to support cleanup of contamination, maintenance, and planning for eventual decommissioning of the facility.

Review Procedures

The reviewer should determine whether all acceptance criteria are reasonably addressed and met. The reviewer should evaluate whether the written plans and the bases of procedures for contamination control include, at a minimum, requirements for monitoring of personnel and property for contamination upon exit from established areas in which contamination could be present. The reviewer should evaluate whether appropriate controls are established to prevent the further spread of contamination if detected. The reviewer should evaluate plans for decontamination.

For material and equipment in areas that could be contaminated, the reviewer should verify that plans and the bases of procedures at the facility treat the material and equipment as radioactive or contaminated so that it could be released from areas where contamination could be present only to other areas with monitoring, control, and documentation in accordance with reviewed and approved procedures. The reviewer should examine the plans governing records for the release of potentially contaminated material and equipment to make sure the property and the results of the monitoring operation would be described in sufficient detail to avoid ambiguity.

The reviewer should examine the description of plans for contamination control at the facility to verify that the facility could comply with applicable requirements and regulations for controlling, identifying, monitoring, labeling, packaging, storing, releasing, transporting, and accounting for contaminated material and waste that is contaminated, and for releasing surface-contaminated material to controlled or uncontrolled areas.

Evaluation Findings

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

- The plans in the SAR for ensuring control of radioactive contamination for *(insert name of facility)* have been reviewed. This included review and evaluation of the following:
 - the depth and breadth of the plan and bases of procedures for anticipating, identifying, controlling further spread of, remedying, and recording information about occurrences of radioactive contaminating materials

- provisions for planning both reactor utilization and operation activities to avoid or prevent uncontrolled occurrence and spread of radioactive contamination
 - provisions for routine monitoring and access control to identify radioactive contamination and to assess and limit personnel exposures
 - the bases for technical specifications that control activities that have the potential to cause or spread contamination
- The staff examined recordkeeping for contamination and historical information about occurrences of radioactive contamination at the facility, which helps to confirm that the program is effective. The program for contamination control meets all regulatory requirements and ensures the control of radioactive contamination so that there is reasonable assurance that the health and safety of the facility staff, the environment, and the public will be protected.

11.1.7 Environmental Monitoring

Areas of Review

The reviewer should evaluate the environmental monitoring program, if one exists at the facility, to verify that the information submitted includes the following:

- compliance with any commitments made in environmental reports or other documents; standards the applicant used in the environmental monitoring program
- if a program has been established, the effectiveness of the program
- for new facilities not yet in operation, establishment of pre-operational baselines used to ascertain natural background so that the radiological impact of facility operation on the environment can be determined
- the facility policy, the bases for procedures implementing the facility policy, the overall program, and technical specifications or internal requirements of the applicant that promote compliance with environmental quality requirements
- the written plans and the bases of procedures for implementing the environmental monitoring operations, including the document control measures employed to ensure that the plans and procedures, including

changes, are reviewed for adequacy and approved by authorized personnel and are distributed to and used at the appropriate locations throughout the facility

- the environmental surveillance program, including information on the identification of possible and probable radioactive contaminants resulting from operation of the facility, selection of sampling materials and locations (include maps), sample collection methods and frequency, sampling and counting equipment, and sample analysis techniques, sensitivities, and detection limits.

Acceptance Criteria

Acceptance criteria for the environmental monitoring program should include the following:

- The documentation should discuss the environmental quality commitments that the program should address and the standards that were used in development of the program.
- The methods used to establish the preoperational baseline conditions for new facilities should be described.
- The methods and techniques to sample and analyze the radiological effect of facility operation should be complete, applicable, and of sufficient validity that the environmental impact can be unambiguously assessed. Results should be compared with preconstruction or preoperation environmental data.
- The environmental monitoring program should provide confidence that a significant radiological impact on the environment from the facility would be detected and the type and magnitude of the radiological impact would be determined.

Review Procedures

The reviewer should confirm that the information in the SAR addresses the issues included in the acceptance criteria and contains the information requested in the environmental monitoring section of the format and content guide. The reviewer should examine the plans and methods designed to assess changes in the environment related to utilization and operation of the reactor. The reviewer should also examine plans for verifying and documenting the results.

Evaluation Findings

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

- The staff has reviewed the description of the environmental monitoring program presented in the SAR for the (*insert name of facility*). This review verified that the environmental monitoring program described is appropriate to the facility and its projected impact. The staff examined the provisions of the program to ensure the safety of the public and protect the environment. This review demonstrates that required and sufficient plans are identified or exist to provide reasonable assurance that an environmental monitoring program can be effectively implemented and sustained during the day-to-day operation of the facility, and that any radiological impact on the environment will be accurately assessed.

11.2 Radioactive Waste Management

As noted earlier, the magnitude of the radioactive waste management function and the scope of a waste management program vary widely from one non-power reactor to another. In general, the amount of radioactive waste formed will be related to the power level of the reactor and to the amount and type of utilization. The reviewer, therefore, should be prepared to find and evaluate provisions for managing such wastes that are commensurate with these factors. Furthermore, as noted, radioactive waste management could be assigned as an auxiliary function to an operations or to a radiation protection organization and not have an organizational unit of its own. In any case, the reviewer should explore and evaluate if the applicant has provided for defining, assessing, and managing such wastes to the extent necessary to protect the facility staff, the environment, and the public from unacceptable exposure to radiation.

Insofar as radioactive waste can be treated as one of the many types of radiation sources at a facility, all of the foregoing guidance in this document is applicable. However, because there may be some differences in management and ultimate disposition of such sources, the following additional guidance is provided.

11.2.1 Radioactive Waste Management Program

Areas of Review

Whether or not the applicant has established an organizational unit dedicated to management of radioactive wastes, the SAR should discuss the program planned

to manage such wastes. The reviewer should expect that the factors addressed by the applicant should include the following:

- philosophy of and approach to management of the wastes
- organization of the management function
- program staffing and position descriptions, and program personnel responsibilities and qualifications as discussed in the format and content guide
- any review and audit committees related to radioactive waste management
- training for staff
- plans for shipping, disposal, and long-term storage,
- program documentation and records, including availability and retention
- audits of the effectiveness of the program
- bases of procedures
- bases of technical specifications

Acceptance Criteria

Acceptance criteria for the radioactive waste management program include the following:

- The SAR should contain a commitment to comply with applicable regulations and guidelines for managing radioactive wastes.
- The program should be designed to address all technical and administrative functions necessary to limit radiation hazards related to radioactive waste. Technical specifications should be proposed and justified if needed to control the program.
- The program should include audit, review, and self-evaluation provisions.
- The program should be sufficiently flexible to accommodate changing radioactive waste loads, changing regulatory requirements, and changing environmental factors, and should remain effective in protecting the health and safety of the facility staff and the public.

Review Procedures

The reviewer should understand and evaluate how the radioactive waste management program fits into the facility's overall management structure, how such wastes are identified and segregated effectively, how the management and radiation protection organization will ensure that radioactive wastes are continuously controlled from formation to ultimate safe disposal, and what organizational entities are assigned responsibilities in the radioactive waste management program. The reviewer should compare the program under review with programs at other similar facilities that have been approved by NRC.

Evaluation Findings

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

- The applicant has described in the SAR the design of the program to manage radioactive wastes in sufficient detail to conclude that
 - the applicant has developed the bases for a complete and effective program,
 - the program includes review, audit, and assessment provisions, and
 - the program complies with all applicable regulations.
- The description of the waste management program gives reasonable assurance that radioactive wastes will not escape the control of the facility and will not pose a risk of undue radiation exposure to the facility staff, the environment, or the public.

11.2.2 Radioactive Waste Controls

Areas of Review

The reviewer should evaluate the radioactive waste control plans at the facility to determine if the plans address all of the factors discussed in the format and content guide related to maintaining control of such wastes from initial formation to ultimate disposition. Acceptable control should include methods to decrease and eventually minimize the formation of radioactive wastes.

Acceptance Criteria

Acceptance criteria for information on radioactive waste controls should include the following:

- The applicant should describe how all processes and procedures that could produce radioactive waste material will be evaluated.
- The discussion should show that appropriate monitoring and sampling will be performed and sufficient analyses will be completed to assess the extent of the radiation exposure from waste products.
- The methods to avoid inadvertent exposure of personnel or uncontrolled escape of the radioactive materials should be described.
- Methods to define and maintain continuous control of radioactive materials that require treatment and management as waste should be included.
- Methods should be discussed by which the quantities of radioactive waste can be decreased.

Review Procedures

The reviewer should compare the plans to identify and maintain control of radioactive wastes with plans at other similar non-power reactor facilities that NRC has found acceptable. The reviewer should also compare the applicant's submittal against the acceptance criteria in this standard review plan.

Evaluation Findings

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

- The applicant has described in the SAR methods by which the waste products from all procedures and processes will be monitored or otherwise assessed for radioactive material contents.
- When appropriate, controls will be established on the waste streams and products designed to prevent uncontrolled exposures or escape of radioactive waste.
- The descriptions of the plans and procedures provide reasonable assurance that radioactive wastes will be controlled at all times in a manner that

protects the environment and the health and safety of the facility staff and the public.

- The applicant has described efforts to evaluate the creation of radioactive wastes at the facility to determine if actions to reduce the amount of waste produced are feasible.

11.2.3 Release of Radioactive Waste

Areas of Review

This topic is briefly treated separately here, even though it may have been addressed within the context of liquid and airborne radioactive effluents. This topic deals with the termination of control of radioactive material by the facility upon release of such effluents to the unrestricted environment or, in the case of solid waste, transfer to another party for disposal. Areas of review should include the methods of characterizing the possible effluents and referencing the applicable regulations that establish limits for release. Descriptions of the identities and amounts of radionuclides in the effluents, the release points, and the characteristics of the environment to which they are released should also be reviewed.

Acceptance Criteria

Acceptance criteria for information on the release of radioactive waste should be based on the following:

- The applicant should describe methods used to identify and characterize liquid and gaseous waste effluents that are released to the unrestricted area that could contain radioactive materials.
- The applicant should identify the radionuclides by quantities, other relevant characteristics, release points, and relevant environmental parameters.
- The applicant should show by appropriate calculations or references that all releases of radioactive effluents would be managed, controlled, and monitored so that limits in applicable regulations would not be exceeded. The applicant should show that procedures are in place for the transfer of solid waste to other parties in accordance with all applicable regulations.
- The applicant should discuss methods to verify that releases have not exceeded applicable regulations or guidelines.

Review Procedures

The reviewer should compare the discussions in the SAR with the regulations in 10 CFR Part 20, Subpart K, and with any applicable guidelines. Furthermore, comparisons should be made with acceptable provisions at other similar non-power reactors that NRC has found acceptable.

Evaluation Findings

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

- The applicant has described radioactive waste effluents expected to be released from the restricted to the unrestricted area. The discussion includes the type and quantities of radionuclides, methods and locations of release, methods of assessing the potential doses to people in the unrestricted area, and methods of comparing the consequences of releases with limits in applicable regulations. The applicant has also described the release of solid waste from the facility for disposal.
- The discussions provide reasonable assurances that releases of liquid and airborne effluents from the facility will not exceed applicable regulations and will not pose unacceptable radiation risks to the environment, or the public.

11.3 Bibliography

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

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1993.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.19, "Shipment and Receipt of Special Nuclear Material by Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1991.

Code of Federal Regulations, Title 10, "Energy," and Title 49, "Transportation," U.S. Government Printing Office, Washington, D.C., revised periodically.

U.S. Nuclear Regulatory Commission, NUREG-0851, "Nomograms for Evaluation of Doses from Finite Noble Gas Clouds," January 1983.

U.S. Nuclear Regulatory Commission, NUREG/CR-2260 "Technical Basis for RG 1.145," 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents."

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

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.9, proposed Revision 1, "Interpretation of Bioassay Measurements," Task DG-8009.

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, "Operating Philosophy for Maintaining Operational Exposures As Low As Is Reasonably Achievable."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.13, Revision 2, "Instructions Concerning Prenatal Radiation Exposure."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.29, "Instruction Concerning Risks From Occupational Radiation Exposure."

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.29, proposed Revision 1, "Instruction Concerning Risks From Occupational Radiation Exposure," Task DG-8012.

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.37, "ALARA Levels for Effluents From Materials Facilities."

12 CONDUCT OF OPERATIONS

This chapter contains guidance for evaluating the information on the conduct of operations at a non-power reactor facility. The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan, the quality assurance plan, the security plan, the reactor operator requalification plan, the startup plan, and environmental reports. The administrative aspects of facility operations are the facility organization, review and audit activities, organizational aspects of radiation safety, facility procedures, required actions in case of license or technical specifications violations, reporting requirements, and recordkeeping. These topics form the basis of Section 6 of the technical specifications.

12.1 Organization

Areas of Review

Areas of review should include the following:

- organizational structure
- responsibilities of individuals and groups
- staffing for reactor operations
- selection and training of personnel
- organizational aspects of radiation protection

The requirements for the organizational aspects of non-power reactor facilities are similar in the various reactor designs. The organization of non-power reactor facilities is discussed in Chapter 14, "Technical Specifications," of the format and content guide. Additional details on the areas of review are given in this chapter of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the non-power reactor organization include the following:

- The applicant should submit a multi-level organization chart showing the person or group with legal responsibility for the reactor license (the licensee) at the top of the organization and the reactor operations staff at the bottom with various levels in between. The applicant should describe the relationship between the line organization and the review and audit function, and the line organization and the radiation protection function.
- The applicant should describe the responsibilities of the groups or persons shown on the organization chart. The issue of who is responsible for the

day-to-day operation of the facility and for radiation protection should be specifically discussed.

- The applicant should discuss the staffing at the reactor facility for various reactor modes, especially when the reactor is not secure. At a minimum, the staffing requirements shall meet the requirements of 10 CFR 50.54 (I)-(m)(1).
- The applicant should discuss the selection of personnel, including the minimum requirements for the facility staff with day-to-day responsibility for reactor safety. For example, the minimum educational requirements for the facility director should be discussed and may constitute a requirement of the technical specifications. The requirements for the university dean, the provost, or the company president should be discussed in general terms and should not constitute a requirement of the technical specifications. The reviewer should consider the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4-1988 in determining the minimum acceptable qualifications.
- The applicant should discuss the training of personnel, should reference the operator training program and the operator requalification program, and should include a review of compliance with the requirements of 10 CFR Part 55. The applicant shall meet the requirements of 10 CFR Part 19 and should discuss training to meet the requirements of 10 CFR Part 19.
- The applicant should discuss the organization of the radiation safety function at the facility (additional details can be found in Chapter 11, "Radiation Protection Program and Waste Management" of this standard review plan). The NRC staff does not have a preference regarding whether the radiation safety function is part of the reactor facility or is provided as a service to the reactor facility by an outside group. In either case, the applicant should describe the ability of the radiation safety staff to raise safety issues with the review and audit committee or with university or corporate upper management. Also, in either case, the radiation safety staff should encompass the clear responsibility and ability to interdict or terminate licensed activities that it believes are unsafe. This does not mean that the radiation safety staff possesses absolute authority. If facility managers, the review and audit committee, and university or corporate upper management agree, the decision of the radiation safety staff could be overruled. However, the applicant should make it clear that this would be a very rare occurrence that would be carefully analyzed and considered.

Review Procedures

The reviewer should review the information against the format and content guide. The reviewer should also examine Chapter 14 of the format and content guide to ensure that the applicant has described the basis of the organizational technical specifications.

Evaluation Findings

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

- The applicant has presented an organizational structure that reflects the complete facility organization from the official license holder to the reactor operations staff. All organizational relationships important to safety have been shown, including that of the review and audit function and the radiation safety function. The organization meets the non-power reactor standards in ANSI/ANS 15.1-1990, "The Development of Technical Specifications for Research Reactors," and ANSI/ANS 15.4-1988, "Selection and Training of Personnel for Research Reactors."
- The applicant has described the responsibilities of the persons in the organizational structure. The responsibility for safe operation of the facility and for the protection of the health and safety of the staff and the public has been shown.
- The applicant has described facility staffing requirements that demonstrate its ability to safely operate the facility and protect the health and safety of the staff and the public. The staffing meets the requirements of the regulations.
- Facility staff will be selected that meet minimum qualifications acceptable for non-power reactors. Reactor operators will be trained in a program that meets the standards for non-power reactors and the requirements of the regulations. Radiation protection training and specialized training will be conducted at an acceptable level.
- The applicant has described a radiation safety organization that is acceptable to the staff. This organization has direct access to upper management and the review and audit committee to express concerns, if necessary. The radiation safety staff has the authority to interdict or terminate activities to ensure safety.

12.2 Review and Audit Activities

Areas of Review

Because strong, independent oversight is very important to the safe operation of the facility and the protection of the health and safety of the public, NRC expects review and audit programs to be viable and fully supported by the licensee. Independent review of certain activities by experts, strengthens the program. Independent audit allows the licensee to find and correct problems before NRC discovers them. Review and audit activities should focus on the following areas:

- composition and qualification of the committee members
- charter and rules of the committee
- conduct of the review function
- conduct of the audit function

The requirements for review and audit activities at non-power reactor facilities are similar in the various reactor designs. These activities are discussed in Chapter 14 of the format and content guide. Additional details on the areas of review can be found in this chapter of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on review and audit activities include the following:

- The applicant should discuss the composition of the review and audit committee. One committee can perform both functions, or each function can be assigned to each of two committees. The review committee may have the authority to approve submitted documents or may give advice to the facility director. If the review and audit functions are separate, a minimum of three persons should be on the review committee and one person on the audit committee. The committee members should represent a broad spectrum of expertise (e.g., nuclear engineering, electrical engineering, mechanical engineering, and radiation protection); the exact composition of the committee will vary from facility to facility. Committee members should be appointed by the highest level of upper management. It is also desirable to have members on the committee who are not employed by the reactor owners.
- The applicant should discuss the charter and rules that govern the operation of the committee. The committee should meet at least once a year [10 CFR 20.1101(c), for example, requires an annual review of the content and implementation of the radiation protection program]. A

quorum should be defined as not less than one-half of the committee membership where the operating staff of the reactor does not constitute a majority. Minutes of committee meetings should be approved and distributed within 3 months after the meeting. Voting may be conducted at the meeting or by polling members. A majority of the committee members must vote for a measure before it passes.

- The applicant should give the details of the review function. The minimum list of items to be reviewed should be those given in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide. The reviews should include 10 CFR 50.59 safety reviews.
- The applicant should give the details of the audit function. The minimum list of items to be audited should be that given in ANSI/ANS 15.1-1990, with the addition of plans such as the quality assurance plan, if the facility has one, and the physical security plan. The audit of facility operations should include items such as organization and responsibilities, training, reactor operations, procedures, logs and records, experiments, health physics, technical specification compliance, and surveillances.

Review Procedures

The reviewer should compare the information with the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide. The reviewer should confirm that the SAR provides a basis for the technical specifications requirements for the review and audit function.

Evaluation Findings

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

- The applicant has proposed a review and audit function for the reactor facility. The committee members appear to be well qualified, with a wide spectrum of expertise. The committee membership includes persons from outside the university (*or corporation*). The staff has determined that the committee membership is acceptable.
- The review and audit committee has proposed a charter and rules that describe the number of times the committee meets, the way the committee conducts business, the requirements for a quorum when voting, and the way the committee distributes its reports and reviews to the applicant. The

staff has determined that the charter and rules for the committee are acceptable.

- The applicant has proposed a list of items that the committee will review. The staff has determined that this list is comprehensive and acceptable.
- The applicant has proposed a list of items that the committee will audit. The staff has determined that this list is complete and acceptable.

12.3 Procedures

Areas of Review

Areas of review should include the following:

- the minimum topics for which procedures are required
- the process for the review and approval of procedures
- the process for making substantive, minor, and temporary changes to procedures

The requirements for procedures at non-power reactor facilities are similar in the various reactor designs. Procedures for non-power reactor facilities are discussed in Chapter 14 of the format and content guide. Additional details on the areas of review can be found in this chapter of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on procedures at non-power reactor facilities includes the following:

- The applicant should propose a minimum list of procedural topics as given in ANSI/ANS 15.1-1990. If byproduct material is used at the facility under the reactor license, the applicant should discuss the requirement for procedures governing the use of this material.
- The applicant should discuss the method for the review and approval of procedures. The method should involve staff from reactor operations, radiation protection, and reactor administration and the review committee, as appropriate to the procedure under review and approval.
- The applicant should propose a method for making changes to procedures. This method should cover minor changes with little or no safety

significance, substantive changes that are safety significant, and temporary deviations caused by operational needs. The applicant should consider the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide..

Review Procedures

The reviewer should compare the information with the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide. The reviewer should confirm that the SAR provides a basis for the technical specifications requirements for procedures. Special circumstances or unusual utilization programs may require procedural topics beyond those given in the various guidance documents.

Evaluation Findings

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

- The applicant has proposed a set of required procedures that is appropriate to operation of the facility as proposed in the SAR and is acceptable to the staff.
- The applicant has described the review and approval process for procedures and has also described the method for making minor and substantive changes to existing procedures and for the temporary deviation from procedures during operations. The staff has determined that the process and method described by the applicant will ensure proper management control and proper review of procedures.

12.4 Required Actions

Areas of Review

The reviewer should examine the applicant's definition of reportable events and the actions to be taken after a reportable event or a violation of the facility safety limits. Some events may occur that the applicant does not consider to be reportable. The applicant is still required to take whatever actions are necessary to protect the health and safety of the public, regardless of whether an event is considered reportable. One difference between reportable and non-reportable events involves the timing of informing NRC of the event. The applicant is expected to keep records of all events, reportable or not, that would be reviewed by NRC during routine inspections. The discussion on reportable events should include notification of management, specification of immediate corrective actions

to correct and prevent recurrence of the event, and reporting to management, the safety review committee, and NRC. Additional information on the areas of review can be found in this section of the format and content guide, in Chapter 14 of the format and content guide, and in ANSI/ANS 15.1-1990.

Acceptance Criteria

The acceptance criteria for the information on required actions at non-power reactor facilities include the following:

- The applicant should discuss events that are defined as reportable. The applicant should consider the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide.
- The applicant should discuss the actions to be taken if a reportable event happens or if a safety limit is violated. The applicant should consider the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide.

Review Procedures

The reviewer should compare the information the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide.. The reviewer should confirm that the SAR provides a basis for the technical specifications requirements for required actions.

Evaluation Findings

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

- The applicant has defined a group of incidents as reportable events and has described the required actions it will take if a reportable event occurs. The definition of reportable events gives reasonable assurance that safety-significant events will be reported by the applicant.
- The applicant has proposed actions to be taken if a safety limit is violated or a reportable event occurs. The staff has determined that the applicant will take whatever actions are necessary to protect the health and safety of the public.

12.5 Reports

Areas of Review

The reviewer should examine the content, timing, and distribution of reports. The main purpose of these reports is to provide timely information to NRC. The type of reports the reviewer should consider are annual reports and special reports that contain information on reportable events, violations of safety limits, and changes in key personnel at the facility or in the transient and accident analysis. The minimum content of reports is discussed in Chapter 14 of the format and content guide. In addition, new facilities or facilities returning to operation after major modifications may be required to submit startup reports, which are discussed in Section 12.11. Additional information on the areas of review is given in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on reports include the following:

- The applicant should discuss the reports applicable to the facility's situation, following the guidance in this section and Chapter 14 of the format and content guide.
- The applicant should discuss the content of reports, the time limits for submitting reports to NRC, and the distribution of the reports.

Review Procedures

The reviewer should compare the information with the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide. The reviewer should confirm that the SAR provides a basis for the technical specifications requirements for reports.

Evaluation Findings

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

- The applicant has described the content, the timing of the submittal, and the distribution of the reports to ensure that important information will be provided to NRC in a timely manner.

12.6 Records

Areas of Review

The reviewer should examine facility records and review the records to be retained and the period of retention.

Acceptance Criteria

The acceptance criteria for the information on records include the following:

- The applicant should propose the retention of facility records following the guidance in this section of the format and content guide.
- The retention time for records should be similar to that given in Chapter 14 of the format and content guide.

Review Procedures

The reviewer should compare the information with the guidance in ANSI/ANS 15.1-1990 and Chapter 14 of the format and content guide. The reviewer should confirm that the SAR provides a basis for the technical specification requirements for records.

Evaluation Findings

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

- The applicant has described the types of records that will be retained by the facility and the period of retention to ensure that important records will be retained for an appropriate time.

12.7 Emergency Planning

Emergency planning is a specialized area of review. The technical reviewer should forward emergency planning information and proposed emergency plans to the Radiation Protection and Emergency Preparedness Branch in the Office of Nuclear Reactor Regulation (NRR) for review. For the review and evaluation of emergency plans at non-power reactors, the emergency plan reviewer should use NUREG-0849 (appears as Appendix 12.2 in the format and content guide) which

covers the unique aspects of these reactors. The emergency plan reviewer should provide an evaluation for inclusion in the staff's safety evaluation report.

12.8 Security Planning

Security planning is a specialized area of review. The technical reviewer should forward security planning information and proposed security plans to the Safeguards Branch in NRR for review. Security planning information is considered either proprietary or safeguards information and must receive special handling. The technical reviewer should ensure that this information is handled properly. The reviewer should contact the Information Security Branch in the Division of Security regarding any questions on the proper handling of security information. The security plan reviewer should provide an evaluation for inclusion in the staff's safety evaluation report and a license condition for inclusion in the facility license.

12.9 Quality Assurance

Quality assurance is a specialized area of review. The technical reviewer should forward proposed quality assurance information to the Quality Assurance and Maintenance Branch in NRR for review. The technical reviewer should remind the quality assurance reviewer that this information is for a non-power reactor and that Regulatory Guide 2.5 (appears as Appendix 12.3 in the format and content guide) and ANSI/ANS-15.8 should be consulted. The quality assurance reviewer should provide an evaluation for inclusion in the staff's safety evaluation report.

12.10 Operator Training and Requalification

Areas of Review

The applicant should submit a reactor operator requalification plan within 3 months after the facility operating license is issued. Each reactor operator or senior reactor operator is required to successfully complete a requalification program developed by the licensee that has been approved by the Commission. Areas of review should include the following:

- requalification schedule
- lectures, reviews, and examinations
- on-the-job training
- emergency procedures
- inactive operators
- evaluation and retraining of operators

- requalification documentation and records
- requalification document review and audit

Additional information on the areas of review is given in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on reactor operator requalification plans include the following:

- The duration of the program should not exceed 24 months. The next requalification period should start immediately after the completion of the preceding period.
- Preplanned lectures should be given on a regular and continuing basis as required by 10 CFR 55.59(c)(2). The lecture topics should follow those given in 10 CFR 55.59(c)(2), at a minimum.
- On-the-job training should occur during the requalification period so that each operator (1) is involved in facility manipulations [see 10 CFR 55.59(c)(3)(i)], (2) understands the operation of apparatus and mechanisms associated with control manipulations and knows operating procedures [see 10 CFR 55.59(c)(3)(ii)], (3) is cognizant of changes in facility design, procedures, and license [see 10 CFR 55.59(c)(3)(iii)], and (4) reviews the contents of all abnormal and emergency procedures on a regular basis [see 10 CFR 55.59(c)(3)(iv)].
- The requalification program should include provisions for evaluating of operators [see 10 CFR 55.59(c)(4)] that include written examinations, observation and evaluation of operator performance, and simulation of emergency or abnormal conditions.
- Written examinations should be used to determine operators' knowledge of subjects covered in the requalification lectures and abnormal and emergency procedures. They should be used to determine areas where retraining is needed. The written examination should contain a sample of the items specified in 10 CFR 55.41 and 55.43 as appropriate to the facility and the operators.
- The operating test should require the operator to demonstrate an understanding of and the ability to perform the actions necessary for a sample of the items specified in 10 CFR 55.45(a)(2-13) as appropriate to

the facility and the operators. The operating test should be given annually and should be used to determine areas where retraining is needed.

- The requalification plan should include systematic observation and evaluation by supervisors of the performance and competency of operators as specified in 10 CFR 55.59(c)(4)(iii).
- The evaluation of operators should include simulation of emergency or abnormal conditions as specified in 10 CFR 55.59(c)(4)(iv).
- Acceptance criteria for acceptable operator performance on examinations and tests should be given. Operators must score at least 70 percent to pass written examinations. Minimum acceptable performance on operating tests should be determined and documented by the applicant before testing begins.
- The applicant should discuss provisions for accelerated requalification if performance evaluations indicate the need [see 10 CFR 55.59(c)(4)(v)]. This includes the status of the operator during retraining, the form retraining will take, and the acceptance criteria to complete retraining.
- The applicant should maintain records in accordance with the requirements in 10 CFR 55.59(c)(5).
- The applicant should discuss the requirements for operators to maintain active status. At a minimum, operators should actively perform the functions of a reactor operator or senior reactor operator for 4 hours per calendar-quarter.
- The applicant should discuss the requirements for inactive operators to return to active status. At a minimum, the requirements should meet those discussed in 10 CFR 55.53(f).
- The applicant should require audits of the plan by the facility review and audit committee at least every other year. This audit requirement should be given in the plan or in the technical specifications.

Review Procedures

The reviewer should confirm that the information in the reactor operator requalification plan includes that given in the areas of review. The plan should meet the requirements of 10 CFR Part 55 and the acceptance criteria above. The reviewer should note that 10 CFR 55.59(c)(7) recognizes that non-power reactors have specialized modes of operation and differences in control, equipment, and

operator skills and knowledge. The plan should generally conform with the requirements of 10 CFR 55.59(c)(1-6) but need not be identical. The reviewer should use judgment and knowledge of non-power reactors to determine the degree of conformity with the regulations.

Evaluation Findings

The reactor operator requalification plan should contain sufficient information to support the following type of conclusion, which will be included in the staff's safety evaluation report:

- The applicant has submitted a reactor operator requalification plan that contains the program for reactor operator and senior reactor operator requalification, the requirements for reactor operators and senior reactor operators to maintain active status, the steps to be taken to return an inactive operator to active status, and a requirement for audits of the plan records. The applicant's procedures for training operators and the operator requalification plan meet the requirements of 10 CFR Part 55 and are acceptable. The plan and procedures give reasonable assurance that the reactor facility will be operated by competent operators.

12.11 Startup Plan

Areas of Review

The applicant should submit a startup plan for a new reactor or, if significant modifications were being made, for a reactor that required confirmation of operability. Areas of review should include the following:

- the proposed tests the applicant will perform to demonstrate operability
- the contents and timing of the report to be submitted to NRC summarizing the startup tests

Additional information on the areas of review is given in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria should be tailored to the situation at the facility and should include the following:

- The applicant should have plans for receiving fuel, handling and performing quality assurance checks on the new fuel, and loading fuel using a critical experiment.
- The critical mass (number of fuel elements) should be approximately known and should be exactly determined by a systematic approach to a critical experiment.
- Neutron detectors of high sensitivity and reliability may be used to supplement the operational instrumentation during subcritical neutron multiplication measurements.
- Measurements should be planned to measure operational reactor physics parameters, such as shutdown reactivity (to confirm shutdown margin), reactivity feedback coefficients, differential and integral control rod worths, power level monitors, scram and interlock functions, fuel heat removal, and related thermal-hydraulic parameters.
- Measured and predicted reactor physics parameters should be compared and the results of the comparisons should be evaluated against preestablished acceptance criteria.
- The control rods should be calibrated and excess reactivity should be loaded systematically, in order to obtain accurate values.
- Thermal power of the reactor should be calibrated acceptably and accurately to ensure compliance with the licensed power level limits and any other license conditions, such as pulse characteristics.
- Area and effluent radiation surveys should be conducted to confirm predictions of the radiological status of the facility.
- All instruments and components should be tested before routine operations begin.
- Other systems discussed in the startup plan should be tested and found to be operational before routine operations begin.

Review Procedures

The reviewer should compare the startup plan submitted by the applicant with startup reports, technical specifications, other license conditions, startup plans, and final reports from other similar reactor facilities. The reviewer should verify that key facility parameters and conditions that are calculated and used in the

applicant's safety analysis will be measured and that acceptance criteria are specified for the tests.

Evaluation Findings

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

- The applicant has submitted a startup plan for bringing the reactor into routine operation. The staff has determined that implementation of the proposed startup plan will provide reasonable assurance that the reactor is operating as described and analyzed in the SAR.

12.12 Environmental Reports

Environmental reports are submitted by applicants to aid the NRC staff in preparing documentation to meet the requirements of the National Environmental Protection Act of 1969, as amended. Licensing actions for non-power reactors usually fall into one of three categories: those identified as categorical exclusions, those requiring the preparation of an environmental assessment, and those requiring the preparation of an environmental impact statement. To date, environmental impact statements have been prepared only for the initial licensing of test reactors. Environmental assessments have been prepared for the construction permit and initial licensing, license renewal, decommissioning plan orders, and license termination for research reactors. Environmental assessments have been prepared for licensing amendments that do not meet the criteria for categorical exclusions.

The environmental report submitted by the applicant should meet the requirements of 10 CFR 51.45. The report should follow the guidance in this section of the format and content guide. The reviewer should decide whether the applicant has provided enough information to allow NRC to prepare the required environmental documents for the licensing action being considered. The NRC preparer of the environmental documents may request additional information from the applicant concerning environmental aspects of the requested licensing action if the environmental report does not contain sufficient information.

Because the environmental impact of a non-power reactor is not significant in most cases, the staff has prepared a generic environmental assessment for the construction, initial licensing, and license renewal of research reactors and critical facilities licensed to operate at power levels of 2 MW(t) or less. A specific environmental assessment should be prepared for these facilities, but the generic

assessment may be used to supplement the specific assessment. Appendix 12.1 is a copy of this generic environmental assessment.

12.13 References

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, LaGrange Park, Illinois, 1988.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1976.

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

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research Reactors."

Appendix 12.1

Environmental Considerations Regarding the Licensing of Research Reactors and Critical Facilities

Environmental Considerations Regarding the Licensing of Research Reactors and Critical Facilities

Introduction

This discussion deals with research reactors and critical facilities which are designed to operate at low power levels, 2 MW(t) and lower, and are used primarily for basic research in neutron physics, neutron radiography, isotope production, experiments associated with nuclear engineering, training, and as part of a nuclear physics curriculum. Operation of such facilities will generally not exceed a 5-day week, 8-hour day, or about 2000 hours per year. Such reactors are located adjacent to technical service support facilities with convenient access for students and faculty.

Sited most frequently on the campuses of large universities, the reactors are usually housed in already existing structures, appropriately modified, or placed in new buildings that are designed and constructed to blend in with existing facilities. However, the environmental considerations discussed herein are not limited to those that are part of universities.

Facility

There are no exterior conduits, pipelines, electrical or mechanical structures, or transmission lines attached to or adjacent to the facility other than for utility services, which are similar to those required in other similar facilities, specifically laboratories. Heat dissipation, if required, is generally accomplished by use of a cooling tower located next to or on the roof of the building. These cooling towers typically are about 10 feet x 10 feet x 10 feet and are comparable to cooling towers associated with the air conditioning systems of large office buildings. Heat dissipation may also be accomplished by transfer through a heat exchanger to water flowing directly to a sewer or a chilled water system. Makeup for the cooling system is readily available and usually obtained from the local water supply.

Radioactive gaseous effluents are limited to argon-41 and the release of radioactive liquid effluents can be carefully monitored and controlled. Liquid wastes are collected in storage tanks to allow for decay and monitoring before dilution and release to the sanitary sewer system. This liquid waste may also be solidified and disposed of as solid waste. Solid radioactive wastes are packed and shipped off site for disposal or storage at NRC-approved sites. Such waste is transported in accordance with existing NRC-DOT [Department of Transportation] regulations in approved shipping containers.

Chemical and sanitary waste systems are similar to those existing at other similar laboratories and buildings.

Environmental Effects of Site Preparation and Facility Construction

Construction of such facilities invariably occurs in areas that have already been disturbed by other building construction and, in some cases, solely within an already existing building. Therefore, construction would not be expected to have any significant effects on the terrain, vegetation, wildlife, or nearby waters or aquatic life. The societal, economic, and aesthetic impacts of construction would be no greater than those associated with the construction of a large office building or similar research facility.

Environmental Effects of Facility Operation

Release of thermal effluents from a reactor of less than 2 MW(t) will not have a significant effect on the environment. This small amount of waste heat is generally rejected to the atmosphere by means of small cooling towers. Extensive drift and/or fog will not occur at this low power level. The small amount of waste heat released to sewers, in the case of heat exchanger secondary flow directly to the sewer, will not raise average water temperatures in the environment.

Release of routine gaseous effluents can be limited to argon-41, which is generated by neutron activation of air. In most cases, this will be kept as low as is practicable by using gases other than air for supporting experiments. Experiments that are supported by air are designed to minimize production of argon-41. Yearly doses to unrestricted areas will be at or below established guidelines in 10 CFR Part 20. Routine releases of radioactive liquid effluents can be carefully monitored and controlled in a manner that will ensure compliance with current standards. Solid radioactive wastes will be shipped to an authorized disposal site in approved containers. These wastes should not require more than a few shipping containers a year.

On the basis of experience with other research reactors, specifically TRIGA reactors operating in the 1- to 2- MW(t) range, the annual release of gaseous and liquid effluents to unrestricted areas should be less than 30 curies and 0.01 curie, respectively.

No release of potentially harmful chemical substances will occur during normal operation. Small amounts of nonradioactive chemicals and/or water with a high-solid content may be released from the facility through the sanitary sewer during periodic blowdown of the cooling tower or from laboratory experiments.

Other potential effects of the facility, such as esthetics, noise, societal, or impact on local flora and fauna, are expected to be too small to measure.

Environmental Effects of Accidents

Accidents ranging from the failure of experiments up to the largest core damage and fission product release considered possible would result in doses that were less than the guidelines in 10 CFR Part 20 and are considered negligible with respect to the environment.

Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation involve the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of these unavoidable effects.

Alternatives to Construction and Operation of the Facility

To achieve the objectives associated with research reactors, there are no suitable alternatives. Some of these objectives are training of students in the operation of reactors, production of radioisotopes, and use of neutron and gamma-ray beams to conduct experiments.

Long-Term Effects of Facility Construction and Operation

The long-term effects of research facilities are considered to be beneficial as a result of the contribution to scientific knowledge and training. Because of the relatively small amount of capital resources involved and the small impact on the environment, very little irreversible and irretrievable commitment is associated with such facilities.

Costs and Benefits of Facility Alternatives

The costs are about several millions of dollars with very little environmental impact. The benefits include, but are not limited to, some combination of the following: conduct of activation analyses, conduct of neutron radiography, training of operating personnel, and education of students. Some of these activities could be conducted using particle accelerators or radioactive sources, which would be more costly and less efficient. There is no reasonable alternative to a nuclear research reactor for conducting this spectrum of activities.

Conclusion

The staff concludes that no significant environmental impact will be associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 MW(t) or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities.

13 ACCIDENT ANALYSES

Other chapters of the SAR should contain discussions and analyses of the reactor facility as designed for normal operation. The discussions should include the considerations necessary to ensure safe operation and shutdown of the reactor to avoid undue risk to the health and safety of the public, the workers, and the environment. The analyses should include limits for operating ranges and reactor parameters within which safety could be ensured. The bases for the technical specifications should be developed in those chapters.

In this chapter the applicant should present a methodology for reviewing the systems and operating characteristics of the reactor facility that could affect its safe operation or shutdown. The methodology should be used to identify limiting accidents, analyze the evolution of the scenarios, and evaluate the consequences. The analyses should start with the assumed initiating event. The effects on designed barriers, protective systems, operator responses, and mitigating features should be examined. The endpoint should be a stable reactor. The potential radiological consequences to the public, the facility staff, and the environment should be analyzed. The information and analyses should show that facility system designs, safety limits, limiting safety system settings, and limiting conditions for operation were selected to ensure that the consequences of analyzed accidents do not exceed acceptable limits.

The applicant should also discuss and analyze a postulated accident scenario whose potential consequences are shown to exceed and bound all credible accidents. For non-power reactors, this accident is called the maximum hypothetical accident (MHA). Because the accident of greatest consequence at a non-power reactor would probably include the release of fission products, the MHA, in most cases, would be expected to contain such a scenario involving fuel or a fueled experiment and need not be entirely credible. The review and evaluation should concentrate on the evolution of the scenario and analyses of the consequences, rather than on the details of the assumed initiating event.

Because the consequences of the postulated MHA should exceed those of any credible accident at the facility, the accident is not likely to occur during the life of the facility. The MHA is used to demonstrate that the maximum consequences of operating the reactor at a specific site are within acceptable limits. The applicant may choose to perform sensitivity analysis of the assumptions of the MHA. For example, reactor operating time before accident initiation may be examined to determine the change in MHA outcome if a more realistic assumption is made. Assumptions made in the accident analysis may form the basis for technical specification limits on the operation of the facility. For example, if the accident analysis assumes that the reactor operates for 5 hours a day, 5 days a week, this may become a limiting condition for operation.

The information in this chapter should achieve the objectives stated in this chapter of the format and content guide by demonstrating that all potential accidents at the reactor facility have been considered and their consequences adequately evaluated. Each postulated accident should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor. The information for a particular reactor may show that some of the following categories are not applicable:

- MHA
- insertion of excess reactivity (ramp, step, startup, etc.)
- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

The applicant should systematically analyze and evaluate events in each group to identify the limiting event selected for detailed quantitative analysis. The limiting event in each category should have consequences that exceed all others in that group. The discussions may address the likelihood of occurrence, but quantitative analysis of probability is not expected or required. As noted above, the MHA analyzed should bound all credible potential accidents at the facility.

The applicant should demonstrate knowledge of the literature available for non-power reactor accident analyses. The Bibliography section at the end of this chapter lists documents categorized as follows: non-power reactors (in general), radiological consequences, and fuel types.

Area of Review

Area of review should include the following: systematic analysis and discussion of credible accidents for determining the limiting event in each category. The applicant may have to analyze several events in a particular accident category to determine the limiting event. This limiting event should be analyzed quantitatively. The steps suggested for the applicant to follow once the limiting event is determined for a category of accidents are given in this chapter of the format and content guide.

Acceptance Criteria

For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 criteria (10 CFR 20.1 through 20.602 and

appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results are less than 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above.

If the facility conforms to the definition of a test reactor, the results of the accident analysis should be compared with the criteria in 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses given in 10 CFR Part 100 are reference values and are not intended to imply that the dose numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, they are values that can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of exposure of the public to radiation.

For MHAs for research reactors, acceptable consequences may exceed 10 CFR Part 20 limits. The reviewer will evaluate this on a case-by-case basis. The applicant should discuss why the MHA is not likely to occur during the operating life of the facility.

Review Procedures

Information in the SAR should allow the reviewer to follow the sequence of events in the accident scenario from initiation to a stabilized condition. The reviewer should confirm the following:

- The credible accidents were categorized, and the most limiting accident in each group was chosen for detailed analyses.
- The reactor was assumed to be operating normally under applicable technical specifications before the initiating event. However, the reactor may be in the most limiting technical specification condition at the initiation of the event.
- Instruments, controls, and automatic protective systems were assumed to be operating normally or to be operable before the initiating event. Maximum acceptable nonconservative instrument error may be assumed to exist at accident initiation.

- The single malfunction that initiates the event was identified.
- Credit was taken during the scenario for normally operating reactor systems and protective actions and the initiation of engineered safety features.
- The sequence of events and the components and systems damaged during the accident scenario were clearly discussed.
- The mathematical models and analytical methods employed, including assumptions, approximations, validation, and uncertainties, were clearly stated.
- The radiation source terms were presented or referenced.
- The potential radiation consequences to the facility staff and the public were presented and compared with acceptable limits.

The reviewer should confirm that the facility design prevents loss of fuel integrity in the event of a credible loss-of-coolant accident (LOCA) or loss-of-flow accident. Emergency core cooling may be required for some non-power reactors to satisfy this condition.

Reactivity limits and the functional designs of control and safety-related systems should prevent loss of fuel integrity during credible accidents involving insertion of some fraction of excess reactivity. At a minimum, the amount of reactivity allowed for moveable or unsecured experiments should be analyzed. Applicable reactivity feedback coefficients and automatic protective actions, if applicable, should be included in the analyses.

Loss of fuel integrity should be prevented if normal electrical power is lost. Safe reactor shutdown should not be compromised or prevented by loss of normal electrical power.

Evaluation Findings

It is essential that all credible accidents at a non-power reactor be considered and evaluated during the design stage. Experience has shown that such facilities can be designed and operated so that the environment and the health and safety of the staff and the public can be protected. Because non-power reactors are designed to operate with primary coolant temperatures and pressures close to ambient, the margins for safety are usually large, and few, if any, credible accidents can be sufficiently damaging to release radioactive materials to the unrestricted area. For potential accidents and the MHA that could cause a release, the acceptance criteria

and review procedures discussed above are sufficiently comprehensive and will not be repeated for each postulated accident. However, the potential consequences, detailed analyses, evaluations, and conclusions are facility specific and accident specific. The findings for the nine major accident categories are presented below. These findings are examples only. The actual wording should be modified for the situation under review.

This section of the SAR should contain sufficient information to support the types of conclusions given below. Those conclusions will be included in the staff's safety evaluation report. The appropriate number for the reactor under evaluation should replace the notation "xx". The reviewer should modify these conclusions to conform to the reactor design under consideration.

Maximum Hypothetical Accident

- The applicant has considered the consequences to the public of all credible accidents at the reactor facility. A maximum hypothetical accident (MHA), an accident that would release fission products from a fuel element or from the failure of a fueled experiment and would have consequences greater than any credible accident, has been analyzed. The MHA, however, is not credible for a non-power reactor. *(The MHA is specific to the reactor design and power. The reviewer may have to evaluate an MHA that differs from the grouping of MHAs that follows. The reviewer should select from items a-e, if appropriate.)*

- (a) *(For TRIGA, PULSTAR, or SPERT fuel), xx (an agreed-upon number, normally one for TRIGA or SPERT fuel; although three has been accepted for PULSTAR, the number is determined on a case-by-case basis) fuel assemblies lose cladding integrity while suspended in air (or in the reactor pool) in the reactor confinement (or containment). All fission products in the gap are released rapidly. The fuel assembly has just been removed from the maximum neutron flux position in the core after long, continuous operation at full licensed power (or full fuel cycle).*
- (b) *[For low-powered (less than 2 MW) MTR fuel] An assembly is stripped of all cladding on one face of one fuel plate while suspended in air (or in the reactor pool) in the reactor confinement (or containment). All fission products escape rapidly by physically sound processes (e.g., conservative analysis, experimental data, or the combination of the two verify the release process). The fuel assembly has just been removed from the maximum neutron flux position in the core after long, continuous operation at full licensed power (or full fuel cycle).*

- (c) *(For high-powered reactors)* Fuel cooling is compromised or reactivity is added to the reactor so that a certain amount of fuel melts causing cladding failure. Fission products are released into the reactor coolant and then into the facility air on the basis of conservative analysis, empirical information, or the combination of analysis and data.
- (d) *(For reactors in which a fueled-experiment failure has greater consequences than fuel failure)* It is assumed that a fueled experiment fails in air (or water) in a reactor irradiation facility. *(Because failure could include melting, all available fission products, or that portion that is demonstrated by analysis, data, or a combination of the two)* Fission products are assumed to escape to the reactor confinement (or containment). The inventory of fissile material is the maximum allowed by technical specifications for a fueled experiment and is consistent with Regulatory Guide 2.2. The failure occurs after long, continuous operation at full licensed power.
- (e) *(For AGN-201 fuel)* It is assumed that fissionable material is inserted into an irradiation facility in the reactor. The added reactivity causes a power excursion. Fuel failure does not occur and the radiological consequence is limited to whole-body dose of xx mrem to the reactor staff.

The reviewer should modify the following paragraphs, as appropriate:

- The air handling and filtering systems (i.e., confinement or containment) are assumed to function as designed, and radioactive material is held up temporarily in the reactor room and then released from the building. Realistic methods are used to compute external radiation doses and dose commitments resulting from inhalation by the facility staff. Realistic but conservative methods are used to compute potential doses and dose commitments to the public in the unrestricted area. Methods of calculating doses from inhalation or ingestion (or both) and direct shine of gamma rays from dispersing plumes of airborne radioactive material are applicable and no less conservative than those developed in Chapter 11 of the SAR. The exposure time for the reactor staff is xx and for the public it is xx.
- The calculated maximum effective doses for the MHA scenario are the following:
 - external—(xx mrem) staff; (xx mrem) public
 - internal—(xx mrem) staff; (xx mrem) public

- These doses and dose commitments are within the acceptable limits (*state the limits*). Because the assumptions of the scenario are conservative, the postulated accident would not be likely to occur during the life of the facility. The applicant has examined more realistic assumptions about operating time and release fractions that decreased the source term by xx percent of the one calculated, lowering the maximum doses by that factor (*if applicable*). Thus, even for the MHA, whose consequences bound all credible accidents possible at the facility, the health and safety of the facility staff and the public are protected.

Insertion of Excess Reactivity

The reviewer should select one of the two findings that follow:

- (1) The applicant has discussed possible methods by which excess reactivity could be inserted accidentally into the reactor to cause an excursion. Rapid insertions were initiated by (*state the initiators analyzed, some examples follow*):

- dropping of a fuel assembly or a fueled experiment into a core vacancy
- removal or ejection of a control, safety, or transient rod
- sudden malfunction, movement, or failure of an experiment or experimental facility
- insertion of a surge of cold primary coolant
- malfunction of reflector components

Slow insertions were initiated by (*state the initiators analyzed, some examples follow*):

- insertion of a fuel assembly or fueled experiment into a core vacancy
- malfunction of a control or safety rod system
- operator error, especially at reactor startup (inadvertent criticality)
- malfunction of power level indicator, especially at reactor startup

- protracted malfunction, movement, or leakage of an experiment or experimental facility
- malfunction of reflector components

The applicant has discussed the scenario for the above events, presented a qualitative evaluation, and compared the likely consequences.

The SAR shows that physical limitations and technical specifications provide acceptable assurance that inadvertent removal or ejection of a control rod, a safety rod, or both, is prevented unless sufficient fuel has been removed, which would ensure subcriticality. Similar controls offer acceptable assurance that fuel or fueled-experiment handling above the core is prevented unless the control rods are in position to ensure subcriticality. Even with such controls, fuel or a fueled experiment could be handled while the reactor is in a critical state and while the core has a fuel vacancy at the core periphery. It is postulated that a fuel element or fueled experiment is inadvertently dropped into the vacancy, rapidly inserting reactivity equal to its worth at that position. The reactor enters a supercritical state by $xx\% \Delta k/k$, which induces a stable positive reactor period of xx msec. Reactor power increases so fast that safety rods are assumed not to move significantly during the transient, even though both the period scram and power level scram are tripped. The power level and fuel temperature are analyzed by validated and acceptable methods. The analyses show that the steam void formed in the core reduces reactivity sufficiently to terminate the excursion, or the prompt negative temperature coefficient of the fuel reduces reactivity sufficiently to terminate the excursion. The safety rods continue to insert within their required drop time, which stabilizes the subcritical reactor. During the transient, xx MW-sec/g of energy was deposited in the hottest point of a fuel element, raising its maximum temperature to xx °C. Because this temperature is lower than the safety limit temperature of the fuel cladding xx °C, fuel integrity would not be lost. Therefore, no fission products would be released from the primary barrier by this accident. *(This approach could also be used for experiment malfunction and other rapid additions of reactivity.)*

Because of the peak power level during the transient, the operator inserting the fuel or fueled experiment was exposed to a brief pulse of radiation. The integrated dose was computed not to exceed xx mrem, which is below acceptable limits for occupational exposures.

or

- (2) The SAR shows that physical and technical specification limitations give reasonable assurance that a rapid insertion of reactivity is not credible. However, malfunction of the control rod drive mechanism or operator error during reactor startup could cause an inadvertent withdrawal of the control rod and an unplanned increase in reactor power. The accident scenario assumes that the reactor has a maximum load of fuel (consistent with the technical specifications), the reactor is operating at full licensed power, and the control system malfunction withdraws the control rod of maximum reactivity worth at its maximum drive speed. Both the power level scram and reactor period scram are assumed to be operable. *(In some analysis it is assumed that the first scram that would terminate the reactivity addition fails and that the second scram terminates the event. In some cases, both scrams are assumed to have failed. If this is the case, the evaluation should be modified appropriately.)* The continuous removal of the rod causes a continuous decrease of reactor period and a continuous increase in reactor power. The analyses, including trip level uncertainties and rod-drop delays, show that the period scram terminates the power increase before the thermal reactor power reaches xx MW. The thermal-hydraulic analysis shows that the energy deposited and instantaneous power level would not raise the peak temperature in the hottest fuel element above xx °C. Because this temperature is lower than the safety limit temperature for fuel cladding (xx °C), fuel integrity would not be lost. *(This approach could be used for other slow additions of reactivity.)*

Loss of Coolant

- The applicant has discussed possible methods by which sufficient primary coolant would be lost rapidly to pose a risk to adequate removal of heat from the fuel. The credible accident with the worst potential consequences is initiated by the catastrophic failure of *(state the component that fails, usually a beam tube or primary coolant pipe)*, which would allow a coolant loss at xx liter/min initially. The scenario assumes that the reactor is operating at full licensed power and has been operating long enough for the fuel to contain fission products at equilibrium concentrations. Therefore, the maximum possible decay heat is available at the start of the event. The pool level scram shuts down the reactor when the coolant reaches the technical specification level. Coolant reaches the top of the core in xx min, and the bottom of the core in xx min. At this time, decay heat raises fuel temperatures. For the SAR analyses, the applicant used validated and acceptable methods to calculate fuel temperature changes.

The reviewer should select one of the following situations:

- With natural-convection air cooling, the analyses show that the peak fuel temperature will not exceed $xx^{\circ}\text{C}$ in xx hr, which is below the temperature necessary for fuel cladding to maintain fuel integrity.
- With the emergency core cooling system functioning as designed, the analyses show that the peak fuel temperature reaches no more than $xx^{\circ}\text{C}$, which is below the temperature necessary for fuel cladding to maintain fuel integrity.
- As the primary coolant escapes and the reactor core becomes uncovered, the decay fission products constitute an unshielded gamma-ray source near the bottom of the pool. This source could expose personnel above the pool to direct gamma radiations and personnel on the floor of the reactor room to scattered gamma radiations. The applicant has analyzed both locations, including the potential doses to facility staff and the public in unrestricted areas. The delay time while the water is escaping from the reactor pool allows the facility staff to take cover and avoid doses larger than xx mrem. The maximum potential dose rates in the unrestricted area would not exceed xx mrem/hr, which provides sufficient time for protective action, if required, so that no doses would exceed acceptable limits.
- To determine the maximum potential consequences for fuel integrity and personnel, the applicant has analyzed a loss-of-coolant scenario in which all primary coolant is lost instantaneously (*if applicable*). The other assumptions are the same as for the slower loss evaluated above. Although the assumptions for this scenario exceed those discussed above, fuel integrity should be ensured and personnel doses would not exceed acceptable limits.

Loss of Coolant Flow

The reviewer should select one of the two findings that follow:

- (1) The applicant has discussed possible methods by which coolant flow through one or more fuel channels could be interrupted while the reactor is operating. The postulated initiating events range from total loss of forced flow as a result of pump or normal electrical power failure to blockage of xx fuel channel(s) by a foreign object. The scenario assumes that the reactor has been operating at full power and fission product decay rates have reached equilibrium.

When the pump stops, a conservative assumption is that forced flow stops instantly (*pump coastdown can be used in the calculations if appropriate*). The coolant-flow scram shuts down the reactor within the technical

specification time limits for circuit delays and rod-drop times. The reactor is designed to change passively to natural-convection flow when forced flow ceases. However, during the changeover, there is a transient period before natural-convection flow can remove decay heat. The analyses account for this transient, showing that the peak fuel temperature does not reach an unacceptable value. Therefore, the maximum credible loss-of-flow accident would not cause loss of fuel integrity.

For blocked fuel cooling channels, the applicant has analyzed heat transfer around the area of the blockage. Appropriate assumptions have been made concerning the amount of time that passes without detection of the blockage. If the blockage is indicated by the reactor instrumentation, reactor operators take appropriate action. Thermal-hydraulic analysis shows that the peak fuel temperature in the area of the blockage will not reach an unacceptable value. Therefore, such blockage would not cause fuel integrity to be lost.

or

- (2) The applicant has shown that fuel cooling channel blockage could lead to fuel melting and fuel cladding failure. The analysis shows that this event is bounded by the fuel failure discussed in the section on the MHA. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and the public are protected.

(If the MHA is not a fuel failure accident, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that a blocked fuel-cooling channels event is not the MHA.)

Mishandling or Malfunction of Fuel

- The applicant has discussed initiating events that could damage fuel or accidentally release fission products from irradiated fuel in the core, in storage, or in between the core and the storage area. The events that would cause the worst radiological consequences have been analyzed by the applicant. This event is *(provide description)*.
- The analysis shows that this event is bounded by the fuel failure discussed under the MHA. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and the public are protected.

(If the MHA is not a fuel failure accident, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that this mishandling or malfunction of fuel is not the MHA.)

Experiment Malfunction

- The applicant has discussed the types of experiments that could be performed at the reactor within its license and technical specifications. The discussions include events that could initiate accidents such as *(list events, some examples are given below)*:
 - melting, leaking, detonation, or failure of the experimental material or its encapsulation, allowing radioactive material to escape into the reactor room or the air exhaust stream to the unrestricted environment
 - movement or misplacement of an experiment into a location of radiation intensity higher than that for which it was planned
 - movement, melting, or other failure of a neutron-absorbing experiment, which causes positive reactivity to be inserted inadvertently into the reactor
 - movement, failure, or leakage of an experimental facility, which causes positive reactivity to be inserted inadvertently into the reactor or radioactive material to be released by the malfunction
- The analysis shows that the technical specifications that limit experiment types and magnitudes of reactivities give reasonable assurance that the potential consequences of these initiating events would be less severe than those already evaluated in the section on the MHA or in fuel handling accident scenarios.

(If the MHA is not a release of radioactive material, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that experiment malfunction is not the MHA.)

Loss of Normal Electrical Power

- The applicant has discussed the events that could result from the sudden loss of normal electrical power. The reactor is designed so that the force of gravity automatically inserts safety or control rods (*or describe the system used that does not require electrical power*) and shuts down the fission reactions when power is lost. Furthermore, reactors with natural-convection cooling are not affected (*reactors with forced-convection cooling passively change to natural convection to remove decay heat when power is lost*).

The reviewer should modify the following statement to apply to the reactor under discussion.

- Most licensed non-power reactors have a large reserve of coolant in the pool that can absorb decay heat for hours, if necessary, without transfer of heat to the secondary system. In a few non-power reactors, emergency electrical power is eventually required to transfer heat to the secondary system. In some non-power reactors, emergency electrical power must be available for specified instrument and control functions. Emergency power design is discussed in Chapter 8 of the SAR. On the basis of these considerations, loss of normal electrical power at a non-power reactor would not pose undue risk to the health and safety of the public.

External Events

- The design to withstand external events and the potential associated accidents is discussed in Chapters 2 and 3 of the SAR. The reactor facility is designed to accommodate these events by shutting down, which would not pose undue risk to the health and safety of the public. For events that cause facility damage (*seismic events that damage the reactor facility or pool*), the damage is within the bounds discussed for other accidents in this chapter. Therefore, exposure to the staff and the public is within acceptable limits and external events do not pose an unacceptable risk to the health and safety of the public. (*An external event could be the MHA if enough damage is done to the facility to damage fuel. The conclusion above for the MHA would apply.*)

Mishandling or Malfunction of Equipment

Initiating events under this heading would require a case-by-case, reactor-specific discussion. If the SAR discusses additional events that fall outside the eight categories, the potential consequences should be compared with similar events already analyzed or with the MHA, as applicable.

Bibliography

Non-Power Reactors

American Nuclear Society (ANS), 5.1, "Decay Heat Power in Light Water Reactors," LaGrange Park, Illinois, 1978.

Atomic Energy Commission, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844, March 23, 1962.

Baker, L., Jr., and Just, L. C., "Studies of Metal-Water Reactions at High Temperatures, III, Experimental and Theoretical Studies of the Zirconium Water Reaction," ANL 6548, Argonne National Laboratory, 1962.

Baker, L., Jr., and Liimatakinen, R. C., "Chemical Reactions" in Volume 2 of The Technology of Nuclear Reactor Safety, Thompson and Beckerly (eds.), Cambridge, Massachusetts: The MIT Press, 1973, pp. 419-523.

Hunt, C. H., and DeBevee, C. J., "Effects of Pool Reactor Accidents," General Electric Technical Information Series, GEAP 3277, November 2, 1959.

Woodruff, W. L., "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," Nuclear Technology, 64, February 1984, pp. 196-206.

Radiological Consequences

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," Publication 30, Part 1, Chapter 8, Pergamon Press, 1978/1979.

Lahti, G. P., et al., "Assessment of Gamma-Ray Exposures Due to Finite Plumes," Health Physics, 41, 1981, p. 319.

U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

U.S. Nuclear Regulatory Commission, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, October 1977.

U.S. Nuclear Regulatory Commission, Nomograms for Evaluation of Doses From Finite Noble Gas Clouds, NUREG-0851, 1983.

Behavior of Zirconium-Hydride Fueled Reactors (TRIGA)

Atomic Energy Commission, "In the Matter of Trustees of Columbia University in the City of New York," Docket No. 50-208, Atomic Energy Commission Reports, Issued May 18, 1972.

Baldwin, N. L.; Foushee, F. C.; and Greenwood, J. S., "Fission Product Release From TRIGA-LEU Reactor Fuels," in Seventh Biennial U.S. TRIGA Users Conference, San Diego, CA, 1980.

Coffer, C. O.; Shoptaugh, J. R., Jr., and Whittemore, W. L., "Stability of the U-ZrH TRIGA Fuel Subjected to Large Reactivity Insertion," GA-6874, General Atomics Company, transmitted by letter dated July 25, 1967 (Docket No. 50-163), January 1966.

Foushee, F. C., and Peters, R. H., Summary of TRIGA Fuel Fission Product Release Experiments, Gulf-EES-A10801, September 1971.

General Atomics Company, "Technical Foundations of TRIGA," GA-0471, August 1958.

Kessler, W. E., et al., "Zirconium-Hydride Fuel Behavior in the SNAPTRAN Transient Tests," Transactions of the American Nuclear Society, 9, 1966, p. 155.

Lindgren, J. R., and Simnad, M. T., "Low-Enriched TRIGA Fuel Water-Quench Safety Tests," Transactions of the American Nuclear Society, 33, 1979, p. 276.

Shoptaugh, J. R., Jr., "Simulated Loss-of-Coolant Accident for TRIGA Reactors," GA-6596, Gulf General Atomic, August 18, 1965.

Simnad, M. T., "The U-ZrH₂ Alloy: Its Properties and Use in TRIGA Fuel," GA-4314, E-117-833, General Atomics Company, February 1980.

Simnad, M. T., and Dee, J. B., "Equilibrium Dissociation Pressures and Performance of Pulsed U-ZrH Fuels at Elevated Temperature," in Thermodynamics of Nuclear Materials. Proceedings of a Symposium, Vienna, September 4-8, 1967, International Atomic Energy Agency, 1968.

Simnad, M. T.; Foushee, F. C.; and West, G. B., "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, 28, 1976, pp. 31-56.

U.S. Nuclear Regulatory Commission, Generic Credible Accident Analysis for TRIGA Fueled Reactors, NUREG/CR-2387, Pacific Northwest Laboratory, 1982.

U.S. Nuclear Regulatory Commission, Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors, NUREG-1282, 1987.

West, G. B., et al., "Kinetic Behavior of TRIGA Reactors," GA-7882, General Atomics Company, March 1967.

Behavior of Plate Fueled Reactors (MTR)

Bullock, J. B., "Calculation of Maximum Fuel Cladding Temperatures for Two Megawatt Operation of the Ford Nuclear Reactor," Memorandum Report No. 1, Memorial Phoenix Project, Michigan, June 1962.

Forbes, S. G., et al., Instability in the SPERT I Reactor. Preliminary Report, IDO-16309, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, October 1956.

Knexevich, M., et al., "Loss of Water at the Livermore Pool Type Reactor," Health Physics II, 1965, pp. 481-487.

Miller, R. N., et al., Report of the SPERT I Destructive Test Program on an Aluminum Plate-Type Water-Moderated Reactor, IDO-16883, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, June 1964.

Nyer, W. E., et al., Experimental Investigations of Reactor Transients, IDO-16285, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, April 1956.

Shibata, T., et al., "Release of Fission Products from Irradiated Aluminide Fuel at High Temperatures," Nuclear Science and Engineering, 87, 1984, pp. 405-417.

Sims, T. M., and Tabor, W. H., Report on Fuel-Plate Melting at the Oak Ridge Research Reactor, July 1, 1963, ORNL-TM-627, Oak Ridge National Laboratory, October 1964.

Wett, J. F., Jr., Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air, ORNL-2892, Oak Ridge National Laboratory, April 16, 1960.

U.S. Nuclear Regulatory Commission, "Analysis of Credible Accidents for Argonaut Reactors," NUREG/CR-2079, Pacific Northwest Laboratories, April 1981.

U.S. Nuclear Regulatory Commission, Rubenstein, L. S., memo to Tedesco, R. L., "Design Basis Event for the University of Michigan Reactor," June 17, 1981.

U.S. Nuclear Regulatory Commission, "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA)," NUREG/CR-2198, Los Alamos National Laboratory, February 1981.

U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors, NUREG-1313, 1988.

Hunt, C. H., and C. J. DeBevee, "Effects of Pool Reactor Accidents," General Electric Technical Information Series, GEAP 3277, Pleasanton, CA, November 2, 1959.

Woodruff, W. L., "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," *Nuclear Technology*, 64, February 1984, pp. 196-206.

Radiological Consequences

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," Publication 30, Part 1, Chapter 8, Pergamon Press, Oxford, NY, 1978/1979.

Lahti, G. P., et al., "Assessment of Gamma-Ray Exposures due to Finite Plumes," *Health Physics*, 41, 1981, p. 319.

U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

14 TECHNICAL SPECIFICATIONS

This chapter provides guidance for reviewing and evaluating the technical specifications submitted to NRC by applicants for non-power reactor licenses.

The format for the acceptance criteria follows the format of American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-1990. In addition to providing the information specified in ANSI/ANS 15.1, the technical specifications shall be consistent with 10 CFR 50.34 and shall address all applicable paragraphs of 10 CFR 50.36. The content of technical specifications for non-power reactors is considerably simpler than that for power reactors, consistent with the difference in size and complexity between non-power reactors and power reactors. Maintaining system performance should provide the basis for the technical specifications of non-power reactors. By addressing limiting or enveloping conditions of design and operation, emphasis is placed on ensuring the safety of the public, the facility staff, and the environment. Because the performance-based concept is used for non-power reactors, standardization is possible across the entire set of technical specification parameters, even for the diverse types of non-power reactors.

ANSI/ANS 15.1 provides the parameters and operating characteristics of a non-power reactor that should be included in the technical specifications. Because of the wide diversity of non-power reactor designs and operating characteristics, some items may not be applicable to all facilities. The reviewer should review the proposed technical specifications considering the design and utilization of the reactor under review. In addition, experience has shown that some of the factors included in ANSI/ANS 15.1 should be explained and supplemented. The NRC staff discusses these factors in the format and content guide. In this standard review plan, guidance is provided for the NRC staff who review non-power reactor technical specifications against the requirements of ANSI/ANS 15.1 and the format and content guide.

Areas of Review

Under 10 CFR 50.36, every operating license for a nuclear reactor is required to include technical specifications that state the limits, operating conditions, and other requirements for facility operation. These specifications are designed to protect the environment and preserve the health and safety of the public.

For non-power reactors, the reviewer should ensure that the technical specifications conform to ANSI/ANS 15.1 and the format and content guide.

Acceptance Criteria

Acceptance criteria for the technical specifications for non-power reactors should include the following:

- Technical specifications shall satisfy 10 CFR 50.34 and 10 CFR 50.36 if they adequately address the issues and parameters of ANSI/ANS 15.1 as supplemented in Appendix 14.1 of the format and content guide for non-power reactors.
- Technical specifications should be standardized and consistent in format with the guidance in Appendix 14.1.
- All conditions that provide reasonable assurance that the facility will function as analyzed in the SAR should be in the technical specifications.
- In conjunction with the findings in other chapters of the staff's safety evaluation report, the limits for the facility design, construction, and operation should provide reasonable assurance that the facility can be operated without endangering the environment or the health and safety of the public and the facility staff.

Review Procedures

The reviewer should compare the proposed technical specifications with ANSI/ANS 15.1, as supplemented in the format and content guide, with previously accepted technical specifications of similar design, operating characteristics, site and environmental conditions, and use, and with the facility SAR.

The technical specifications and basis should be determined from the analysis in the SAR. The reviewer should confirm that the technical specifications are complete and follow the correct format. This review is part of the process to review and approve the technical specifications. The reviewer should confirm that each technical specification is supported by appropriate reference to SAR analysis and statements. NRC review of those SAR chapters should support the finding that each of the technical specifications is acceptable to NRC.

Evaluation Findings

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

- The staff has evaluated the applicant's (*or licensee's*) technical specifications in this licensing action. These technical specifications define certain features, characteristics, and conditions governing the operation of the (*insert name*) facility and are explicitly included in the (renewal) license as Appendix A. The staff has reviewed the format and contents of the technical specifications using the guidance of ANSI/ANS 15.1-1990, "Development of Technical Specifications for Research Reactors," and NUREG-1537 (Part 1), "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," February 1996.

The staff finds the technical specifications acceptable and concludes that normal plant operation within the limits of the technical specifications will not result in offsite radiation exposures in excess of 10 CFR Part 20 guidelines and reasonably ensures that the facility will function as analyzed in the safety analysis report. Furthermore, adherence to the technical specifications will limit the likelihood of malfunctions and mitigate the consequences to the public of off-normal or accident events.

References

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

U.S. Nuclear Regulatory Commission, NUREG-1537 (Part 1), "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," February 1996.

15 FINANCIAL QUALIFICATIONS

This chapter provides review and acceptance criteria for financial information submitted to NRC by applicants for non-power reactor licenses. This information is used to establish that the applicant is financially qualified to own, construct, operate, and decommission a non-power reactor. This information is usually submitted along with the application for a construction permit and an initial operating license or along with the application for license renewal. Financial qualifications cover three areas:

- financial ability to construct the non-power reactor facility authorized by the construction permit
- financial ability to safely operate the facility
- financial ability to safely decommission the facility so that the NRC can terminate the facility license at the end of the facility's use

Financial information to be submitted by the applicant is discussed in 10 CFR 50.33(f) and (k). The cover letter for the construction permit and operating license application, or for the license renewal application, can refer to this chapter of the SAR for complete financial information. If the applicant considers its financial information to be proprietary, an affidavit in accordance with 10 CFR 2.790 shall be submitted to request withholding of the information from the public. If possible, a non-proprietary version of the financial information should also be submitted.

Because the review of this information requires specialized knowledge that the non-power reactor technical reviewer usually does not possess, this financial information is submitted to expert financial reviewers (located in the License Renewal and Environmental Review Project Directorate in the Office of Nuclear Reactor Regulation at the time of this writing). The financial reviewer may ask the technical reviewer to comment on the reasonableness of the cost estimates submitted by an applicant.

15.1 Financial Ability To Construct a Non-Power Reactor

Areas of Review

In this section, the reviewer should evaluate information which demonstrates that the applicant possesses, or has reasonable assurance of obtaining, the funds necessary to cover estimated construction costs and related fuel cycle costs.

Areas of review should include the following:

- estimates of construction costs
- estimates of fuel cycle costs
- sources of funds to cover these costs

Additional information about the areas of review for construction financial assurance can be found in this section of the format and content guide. If a substantial percentage of the funding sources is not firm enough to provide reasonable assurance that the facility will be constructed as described in the application, the staff may either require the applicant to provide firm sources of funding before NRC issues the construction permit or may make the construction permit conditional upon the applicant ultimately securing the funds needed.

Acceptance Criteria

The acceptance criteria for the information on financial assurance for construction should include the following:

- The applicant should estimate the construction cost and give a reliable basis for the estimate, such as the calculations of the design architect or the engineering company, construction bids, or comparison to other completed, similar non-power reactors.
- The applicant should estimate the fuel cycle cost and give a reliable basis for the estimate, such as a calculation of fuel use and estimates of fuel vendors and other contractors or the experience of licensees of similar facilities.
- The applicant should discuss the sources of funding for construction, the amount of funding that is committed, and the amount that is potentially available; the applicant should confirm committed sources and discuss the conditions under which potential sources of funding would become committed.

Review Procedures

The reviewer should submit information provided by the applicant concerning reasonable assurance of construction financing to the License Renewal and Environmental Review Project Directorate for expert financial review. The financial reviewer may ask the technical reviewer to comment on the reasonableness of the estimates of construction and fuel cycle costs, based on the technical reviewer's knowledge of costs to construct other non-power reactors.

Evaluation Findings

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

- The applicant has supplied financial information for construction and fuel cycle costs. The staff reviewed the financial ability of the applicant to construct the proposed facility and to cover fuel cycle costs. The staff concludes that there is reasonable assurance that funds will be made available to construct and cover fuel cycle costs for the facility and that the financial status of the applicant regarding construction and fuel cycle costs is in accordance with the requirements of 10 CFR 50.33(f). Therefore, the staff concludes that the applicant's financial qualifications for construction of the facility and fuel cycle costs associated with the facility are acceptable.

15.2 Financial Ability To Operate a Non-Power Reactor

Areas of Review

In this section the reviewer should evaluate information which demonstrates that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operating costs for the duration of the license.

Areas of review should include the following:

- estimates of facility operational costs, focusing on the first 5 years of operation
- sources of funds to cover operational costs, focusing on the first 5 years of operation
- percentage of the cost of the operation of the facility devoted to commercial activities

Additional information about the areas of review for financial assurance for operations can be found in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on financial assurance for operations should include the following:

- The applicant should estimate the first 5 years of operating cost and give a reliable basis for the estimate, such as past operating costs or costs of operating similar facilities.
- The applicant should discuss the sources of funding for operating costs, the amount of funding that is committed, and the amount that is potentially available; the applicant should confirm committed sources and discuss conditions under which potential sources of funding would become committed and how the facility can be safely operated if some potential sources of funding are not realized.
- If the facility is to be licensed as a non-commercial, non-power reactor, the applicant should show that less than 50 percent of the cost of operating the facility is devoted to commercial activities.

Review Procedures

The reviewer should submit information provided by the applicant concerning reasonable assurance of operating costs to the License Renewal and Environmental Review Project Directorate for expert financial review. The financial reviewer may ask the technical reviewer to comment on the reasonableness of the estimates of operating costs, based on the technical reviewer's knowledge of costs to operate other non-power reactors. The financial reviewer may also ask the technical reviewer to comment on the activities considered to be commercial activities at a non-power reactor.

Evaluation Findings

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

- The applicant has supplied financial information for operating costs. The staff reviewed the financial ability of the applicant to operate the proposed facility. The staff concludes that funds will be made available to operate the facility and that the financial status of the applicant regarding operating costs is in accordance with the requirements of 10 CFR 50.33(f). Therefore, the staff concludes that the financial qualifications of the applicant for operating the facility are acceptable. The staff has also reviewed the proposed conduct of commercial activities at the facility. Because the cost of conducting commercial activities at the facility is less than 50 percent of the total cost of operating the facility, the staff concludes that the facility is not a commercial non-power reactor.

15.3 Financial Ability To Decommission the Facility

Areas of Review

In this section the reviewer should evaluate information indicating how reasonable assurance will be provided that funds will be available to decommission the facility.

The reviewer should examine the following:

- the cost estimate for decommissioning the facility
- the method or methods of providing funds for decommissioning
- how the cost estimate for decommissioning the facility is adjusted periodically over the life of the facility

Additional information about the areas of review for financial assurance for decommissioning can be found in this section of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on financial assurance for decommissioning should include the following:

- The applicant should discuss the decommissioning method to be used for the facility in sufficient detail to permit cost estimates to be developed.
- The applicant should estimate decommissioning costs and take into account the decommissioning method to be used. The decommissioning cost estimates are comparable to those for similar reactor designs reviewed by the staff or are comparable to the actual costs for the decommissioning of similar designs.
- The applicant should propose a method of funding decommissioning costs that meets the requirements of the regulations.
- The applicant should propose a method of periodically adjusting the decommissioning cost based on actual cost changes or changes in cost indices. Funding should be adjusted on the basis of cost changes.

Review Procedures

The reviewer should submit information provided by the applicant concerning reasonable assurance of decommissioning costs to the License Renewal and

Environmental Review Project Directorate for expert financial review. The financial reviewer may ask the technical reviewer to comment on the reasonableness of the estimates of the decommissioning costs, based on the technical reviewer's knowledge of decommissioning costs at other non-power reactors. The cost of high-level radioactive waste and spent fuel disposal is discussed in Section 1.7 of this document. The applicant shall have entered into an agreement with DOE for disposal of this material.

Evaluation Findings

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

- The applicant has supplied financial information for decommissioning costs. The staff reviewed the financial ability of the applicant to decommission the proposed facility. The staff finds the applicant's justification for the estimated cost of decommissioning the facility acceptable. The applicant has provided financial assurance for decommissioning the facility by the *(state the method used here)* method. The applicant has proposed a method for periodically adjusting the cost estimate and funding of decommissioning that the staff finds acceptable. The staff concludes that funds will be made available to decommission the facility and that the financial status of the applicant regarding decommissioning costs is in accordance with the requirements of 10 CFR 50.33(k) and 50.75. Therefore, the staff concludes that the financial qualifications of the applicant for decommissioning of the facility are acceptable.

16 OTHER LICENSE CONSIDERATIONS

This chapter contains guidance for evaluating license considerations that do not belong elsewhere in the SAR. One of these considerations is prior use of reactor components. A recent consideration discussed in this chapter is medical use of non-power reactors. There may be other topics that should appear in this chapter, but the applicant should determine these on a case-by-case basis.

16.1 Prior Use of Reactor Components

Areas of Review

Prior use needs to be considered in the case of license renewal or in the case of the previous use of facility components and systems. Prior use need not have occurred at the reactor for which the applicant is seeking a license. At new facilities, components previously used at other reactor facilities may be used, or the Department of Energy may provide fuel that was used in another non-power reactor that was shut down. The applicant should consider how the component or system was used in the past.

Areas of review should include the following:

- selection of components and systems for consideration of prior use
- discussion of deterioration mechanisms for the items considered
- analyses, tests, and measurements used to gauge deterioration
- discussion of the preventive and corrective maintenance program and success of the program
- discussion and analysis of why components and systems for which prior use was considered are acceptable for continued operation during the requested license period

Acceptance Criteria

The acceptance criteria for the information on the prior use of reactor components should include the following:

- The applicant should consider facility components and systems for the effects of prior use. Components and systems that have been used before and are significant to safety, such as fuel cladding, reactivity control

system, engineered safety features, and radiation monitoring systems, should be identified.

- The applicant should take into account the various deterioration mechanisms for the components and systems under consideration and note which mechanisms are applicable for those components and systems.
- The applicant should determine and justify acceptable levels of deterioration for the components and systems under consideration.
- Analysis should show that unacceptable levels of deterioration will not be reached during the license period. If analysis cannot show this, tests and measurements to gauge deterioration should be discussed. For components and systems that must be tested or measured, the applicant should propose technical specifications that state the frequency of the test or measurement and give performance standards for the component or system under consideration.
- The facility maintenance program should be an organized, systematic approach considering the issue of prior use of components and systems and should be based on analyses, tests, measurements, or manufacturer's recommendations to carry out maintenance.
- The applicant should show that components significant to the safety of the non-power reactor will function satisfactorily for the license period.

Review Procedures

The reviewer should study Chapter 13, "Accident Analyses," of the SAR to determine if the applicant has chosen proper components and systems for consideration. The reviewer can consider the performance of similar components in reactors or environments comparable to the facility under consideration. The reviewer should confirm that tests, measurements, and performance standards for important components and systems appear in the technical specifications.

Evaluation Findings

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

- The applicant has considered prior use of facility components and systems that perform a safety function. The components and systems considered

were (*list*). The prominent deterioration mechanisms considered by the applicant for these components and systems were (*list*).

- For (*list components and systems*), deterioration during the license period could be a concern. For these components and systems, the applicant has either (1) proposed and justified tests and measurements with acceptance criteria that help ensure that components and systems will be replaced or repaired before they perform unacceptably or (2) determined analytically that deterioration will be within acceptable limits. Failures that have occurred appear to be isolated events and do not represent a significant deterioration of the facility or a weakness in the maintenance program.
- Components or systems that perform a safety function have not significantly deteriorated and facility management will continue to maintain the facility so that there is no significant increase in the radiological risk to the facility staff or the public from component or system failures.

16.2 Medical Use of Non-Power Reactors

Areas of Review

A renewed interest has been expressed in the use of non-power reactors for the production of neutrons for medical therapy. Section 104a of the Atomic Energy Act of 1954, as amended (AEA), allows licenses to be issued for the use of utilization facilities in medical therapy.

The use of neutron beams to treat the cancer glioblastoma multiforme and other brain tumors was identified in the 1950s as a potential therapeutic use for non-power reactors. This treatment is called boron neutron capture therapy (BNCT) and consists of a pre-irradiation administration of a boron compound to the patient and the concentration of the boron compound in the tumor. The patient is then exposed to a neutron beam from the non-power reactor, causing the boron to fission into lithium and an alpha particle, which are heavy charged particles. In theory, these particles cause secondary ionization that kills the tumor cells.

Limited trials of BNCT were conducted in the late 1950s and early 1960s at Brookhaven National Laboratory and the Massachusetts Institute of Technology. These trials did not seem to hold much promise, and human irradiations for the conduct of BNCT stopped. Improvements in the drugs that concentrate boron in tumor cells and improvements in beam technology toward the use of epithermal beams have convinced researchers to reconsider BNCT and the medical use of non-power reactors.

Medical therapy at non-power reactors comprises two regulatory components. The first component involves the medical use licensee, who is responsible for the patient, preparation of a treatment plan and written directive (as defined in 10 CFR Part 35), administration of the boron pharmaceuticals and the neutron beam to the patient, supervision of the setup and irradiation of the patient, and control of the byproduct material formed in the patient's body as a result of the treatment. The regulatory requirements for the medical use licensee are not discussed in this chapter other than to state that the license must list the non-power reactor medical therapy treatment facility as an additional place of use. For further details on this aspect of BNCT, contact the Medical, Academic, and Commercial Use Safety Branch of the Division of Industrial and Medical Nuclear Safety in the Office of Nuclear Material Safety and Safeguards at NRC.

The second regulatory component involves the non-power reactor, which provides the treatment (neutrons) under the 10 CFR Part 50 reactor license. The non-power reactor component comprises the medical therapy treatment facility, production of the neutron beam, physical characterization of the beam at its interface with the patient, all health physics considerations associated with the beam, radioactive contamination and activation of the medical therapy treatment room and its contents, and adherence to a quality management program for the conduct of human therapy. The regulatory approach for the non-power reactor is modeled after the approach found in 10 CFR Part 35, Subpart I, concerning teletherapy.

Licensees may also have to meet Food and Drug Administration (FDA) requirements and receive approval from FDA to conduct medical therapy. These requirements are beyond the scope of this document.

Licensees who want to irradiate patients shall hold a Class 104a license from NRC. The determination if a construction permit (CP) is needed for an existing facility to receive a Class 104a license depends on the amount of modification and construction required at the facility to initiate BNCT. In general, if major modifications do not need to be made to the reactor core or structure, a CP is not required. However, the applicant should recognize that failure to address safety issues at the start of a modification could result in delays or additional modifications at a later point. Medical therapy treatment facilities can be installed, filters can be added to existing beam tubes, and in most cases, additional beam tubes can be installed in the reactor without a CP.

Areas of review should include the following:

- administrative requirements associated with patient irradiations, including quality assurance

- design of the medical therapy treatment facility to ensure radiation safety
- beam design to ensure radiation safety
- radiation monitoring to ensure radiation safety of staff and the patient
- beam monitoring to ensure that correct neutron beam characteristics are present
- surveillance requirements for the medical therapy treatment facility and the neutron beam
- calibration requirements for radiation and beam monitors

Additional information concerning the areas of review for medical use of non-power reactors can be found in this chapter of the format and content guide.

Acceptance Criteria

The acceptance criteria for the information on the medical use of non-power reactors include the following:

- An applicant proposing major modifications to the reactor core or structure should submit an application for a construction permit.
- The radiation shielding design of the medical therapy treatment room shall conform with the requirements of 10 CFR Part 20 for dose rates outside the room when the reactor is in operation and inside the room when medical therapy is not in progress and people are present. The design should also be in agreement with the guidance of the non-power reactor facility program to keep exposure as low as is reasonably achievable (ALARA), as required by 10 CFR Part 20.
- The medical therapy treatment room should be designed to minimize neutron activation of the room's walls, floor, and roof and the equipment used in the room. These aspects of the design should conform with the requirements of 10 CFR Part 20 and the guidelines of the non-power reactor facility ALARA program.
- The design of the medical therapy treatment room should prevent entry during treatment. The entry area should have a shielded door, labyrinth with unshielded door, or some other method to prevent entry during

treatment. The door or entrance should be interlocked with the systems used to cut off the beam if someone enters.

- The medical therapy treatment facility should be designed with two independent, redundant systems that have the capability to cut off the neutron beam within a short time. Beam controls and method of motion should be as described in the format and content guide. Interlocks between the systems and the medical therapy treatment room door or entrance should be based on the design guidance in the format and content guide.
- The medical therapy treatment room should have a method for viewing the patient during treatment. If a direct viewing method such as through a lead glass window is not used, redundant methods should be provided. Emergency lighting should be provided.
- The medical therapy treatment facility personnel should be able to scram the reactor from the medical therapy treatment facility control area and from within the medical therapy treatment room. They should also be able to communicate with the reactor control room and the inside of the medical therapy treatment room.
- The medical therapy treatment room should contain a radiation monitoring system that confirms that the beam is off by measuring radiation fields inside the room. The monitor should have alarms as described in the format and content guide and should have an emergency power supply.
- The medical therapy treatment facility should have a beam monitoring system that determines the dose given to the patient.
- The responsibilities of the non-power reactor licensee and the physician authorized user should be stated. This includes the method of referring patients for treatment to the non-power reactor and the rules for starting and stopping treatment.
- Procedures should exist for removing the patient from the medical therapy treatment room in the case of medical complications or equipment failures.
- The applicant should propose surveillance requirements for equipment important to the safety of the staff who work in the medical therapy treatment facility and patients who are treated there. These surveillances may be postponed if therapy is not being conducted.

- The radiation monitor in the medical therapy treatment room should be calibrated at regular intervals. The neutron beam monitors should also be calibrated at regular intervals. In one case, the NRC staff accepted a 2-year interval for the neutron beam monitor.
- Calibration checks of the beam, functional checks of the beam monitors, and characterization of the beam should be performed at regular intervals. In one case the NRC staff accepted a weekly interval for calibration checks of the beam and functional checks of the beam monitors and a 6-month interval for characterization. If no patient is being treated, these checks need not be done. Also, if the beam is modified or maintenance is done, the applicant should ensure that the beam characteristics have not changed.
- The applicant should discuss the limitations on the amount the actual radiation fluence given to the patient can exceed the prescribed amount. The NRC staff has accepted total fluence for a treatment that does not exceed the patient treatment plan by more than 20 percent. If the treatment is given in multiple fractions, a difference of 30 percent is allowed for any given week. If the treatment consists of three or fewer fractions, the total radiation fluence cannot exceed the prescribed amount by more than 10 percent.
- The applicant should define recordable events. In one case the NRC staff accepted the following events as recordable events:
 - a radiation treatment that was given without a written directive
 - not reporting in writing to the medical use licensee the fluence given within 24 hours of treatment, if a written directive is required
 - a treatment during which the administered radiation fluence for a given fraction was 15 percent or more greater than that prescribed
- The applicant should define misadministrations. In one case the NRC staff accepted the following events as misadministrations:
 - a treatment involving the wrong patient, mode of treatment, or treatment site
 - a treatment during which the fluence was greater than the limitations on fluence discussed above

- Requirements for reporting misadministrations to NRC should be discussed. The NRC staff has accepted 24-hour oral notice with a written report within 15 days. Records of recordable events should be maintained for 5 years.
- The applicant should discuss control of repair, maintenance, and modification activities.
- The applicant should discuss training requirements as outlined in the format and content guide.
- The applicant should submit a quality management program that covers the topics outlined in the format and content guide.
- The applicant should discuss other procedures pertaining to treatment that involve the reactor. This includes measurements of boron content in blood and tissue. Requirements for calibration and surveillance of equipment to ensure proper operability should be stated.

If the primary scram method is out of service in the medical therapy treatment facility control area, the applicant may propose the temporary use of the communications link with the control room to ask the operator to scram the reactor. If the primary method to indicate the status of the systems used to cut off the beam is out of service, the applicant may propose a temporary alternative. If the radiation monitor in the medical therapy treatment room is out of service, the applicant may propose a temporary alternative method of monitoring radiation in the room. In these cases, the use of the temporary alternative should be limited to a short period of time (e.g., no more than 10 working days).

Review Procedures

If the applicant needs to make major modifications to the reactor facility that require the issuance of a construction permit, the reviewer should use other applicable chapters of this standard review plan to evaluate those changes. For example, changes to the reactor core would be reviewed using Chapter 4, "Reactor Description." The reviewer should consult the Medical, Academic, and Commercial Use Safety Branch in the Office of Nuclear Material Safety and Safeguards during the review of the application, particularly for the quality management program and aspects of the application that deal with medical issues (e.g., definitions of recordable events and misadministrations). The appendix to this chapter in the format and content guide is a copy of the license amendment for conducting BNCT at the Massachusetts Institute of Technology.

Evaluation Findings

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

- The design of the medical therapy treatment facility is in compliance with the requirements of 10 CFR Part 20 and in agreement with the guidelines of the non-power reactor facility ALARA program.
- The design of the medical therapy treatment facility controls and systems used to cut off the neutron beam provides reasonable assurance that the neutron beam can be controlled by the applicant. Interlocks exist to prevent accidental exposure to the neutron beam. The radiation monitor in the medical therapy treatment room indicates the beam status.
- Review of the monitoring of the neutron beam shows that reasonable assurance exists that the applicant can determine neutron fluence and the neutron spectrum of the beam. Requirements pertaining to the accuracy of the fluence delivered to the patient are in effect.
- The administrative requirements for treatment clearly delineate the responsibility of the applicant in accepting patients, the aspects of the treatment the applicant is in charge of, and the conditions for initiating and discontinuing treatment.
- The requirements pertaining to the qualification and training of personnel associated with the medical therapy treatment facility are acceptable.
- The proposed surveillances associated with the medical therapy treatment facility that encompass safety of staff and patients are acceptable.
- The applicant has described a quality management program for the medical therapy treatment facility that is acceptable.

17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

This chapter contains guidance for evaluating decommissioning plans (DPs), requests for license termination, and applications for possession-only license amendments for non-power reactors. These applications are submitted by non-power reactor licensees who wish to terminate operations and decommission their facilities.

During the past 54 years, many non-power reactors, critical facilities, and special-purpose reactors have been dismantled, decontaminated, decommissioned, and released for unrestricted use. In the absence of detailed regulations governing decommissioning, NRC has developed a systematic approach for licensee and NRC actions to terminate facility licenses [American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-1981 and NRC "Guidance and Discussion of Requirements for an Application To Terminate a Non-Power Reactor Facility Operating License"]. Among these actions are issuance of possession-only license amendments and orders authorizing facility dismantlement.

On June 27, 1988, NRC published a notice of rulemaking (53 FR 24018) amending its regulations. These amendments affected the decommissioning of licensed reactor facilities. The amended regulations broaden the requirements leading to license termination, but do not change the characteristics of possession-only license amendments. The reviewer should be aware that additional changes have been proposed to the decommissioning regulations (see 60 FR 37374). Therefore, the following guidance is offered in the interim to facilitate the review of decommissioning activities during this period.

Because not all licensees request a possession-only license amendment before submitting a DP and a request for terminating the license, DPs and applications for a possession-only license amendment are treated separately in this review plan.

17.1 Decommissioning

17.1.1 Preliminary Decommissioning Plan

In addition to the DP required by 10 CFR 50.82, 10 CFR 50.75(f) requires each licensee to submit a preliminary DP. The preliminary DP shall be submitted at or about 5 years before the projected end of operation. The DP shall contain an estimate of the cost of decommissioning and an up-to-date assessment of the major technical factors that could affect planning for decommissioning. The licensee shall consider the following factors when assembling this information:

- the decommissioning alternative or method to be used
- major technical activities necessary to carry out decommissioning safely
- the current situation for disposal of high- and low-level radioactive waste
- residual radioactivity criteria
- other site-specific factors that could affect decommissioning planning and cost

The preliminary DP should show that the licensee is aware of the technical and administrative complexity of decommissioning. The reviewer may compare the licensee's plan to other plans for the same reactor type that have been reviewed.

17.1.2 Decommissioning Plan

As amended, 10 CFR 50.82(a) requires that a licensee who plans to cease operating a reactor facility apply both for authorization to dismantle and decontaminate the facility and for NRC to terminate the license. The application for termination of the license shall be accompanied, or preceded, by a proposed DP that describes how the licensee plans to dismantle the plant and reduce the residual radioactivity at the site.

The schedule for decommissioning is discussed in 10 CFR 50.82(b)(1)(ii), which requires that a non-power reactor be decommissioned without significant delay, except when concern for public health and safety makes delay necessary. The factors to be considered in evaluating an alternative that delays the completion of decommissioning are discussed in 10 CFR 50.82(b)(1)(iii). Among the factors that may warrant a delay in decommissioning are unavailability of waste disposal facilities, another non-power reactor on the same site, and other site-specific factors.

Under some circumstances, the licensee can apply for a possession-only license amendment in accordance with 10 CFR 50.90 after operations have ended and before decommissioning starts. This amendment normally does not extend the term of the facility license. Generally, the amendment should be based on the stated intent of the licensee to develop and submit a DP and an application to dismantle and decommission the facility and terminate the license. The possession-only license amendment grants the licensee regulatory relief from the requirements for an operating non-power reactor and permits the licensee to retain the reactor facility and related radioactive byproduct material and, in some cases, special nuclear material, pending approval of the DP.

Decommissioning an NRC-licensed facility entails (1) safe termination of facility operation, (2) reduction of residual reactor-related radioactivity to a level that permits release of the property, and (3) termination of the license. The first two activities are the responsibility of licensee, and the third is the responsibility of NRC at the request of the licensee.

17.1.3 Review of the Decommissioning Plan

In 1989 NRC issued draft Regulatory Guide DG-1005, which gave guidance on the format and content of DPs for nuclear reactors. This guide followed the informal guidance already used at NRC and suggested small changes in the format for licensee-initiated DPs. To provide guidance specifically for decommissioning non-power reactors, NRC has developed a format and content guide for applications for the licensing of non-power reactors (NUREG-1537, Part 1). Chapter 17 of that document is an update of "Guidance and Discussion of Requirements for an Application To Terminate a Non-Power Reactor Facility Operating License."

NRC must approve the DP before the licensee can begin any decommissioning activities.

Appendix 17.1 contains guidance on reviewing the format and content of decommissioning plans. Each section corresponds to the numbered section of the DP.

Areas of Review

Areas of review should include the following:

- management and organizational structure required to successfully complete the dismantlement and decommissioning
- methods, techniques, and equipment to assess radioactivity
- methods, techniques, and equipment for removing radioactively contaminated materials and components
- methods of controlling radioactivity for acceptable disposal
- methods for assessing and limiting personnel exposures to radiation, i.e., monitoring and protecting personnel from undue external exposures to the whole body or internal doses from ingestion, inhalation, or skin contact

- methods, techniques, and equipment for accurately measuring low exposure rates and low concentrations of reactor-related radioactivity
- approach and format for reporting radiological information for the residual facility when the licensee requests termination of the license and release of the facility.

Acceptance Criteria

The acceptance criteria for the information on the DP should include the following:

- The technical content of the DP should be detailed and accurate to demonstrate, with reasonable assurance, that the licensee has planned adequately and will have the funds and personnel to complete the decommissioning successfully on the proposed schedule.
- The DP should show that the licensee understands the actions necessary to assess and remove enough reactor-related radioactivity to support the termination of all applicable licenses

Review Procedures

The reviewer should evaluate the DP for completeness and technical accuracy. The reviewer should compare the analyses and discussions in the DP with appropriate document guidance and, as applicable, with NRC-approved DPs for similarly decommissioned facilities.

Evaluation Findings

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

- The DP shows that the licensee will comply with the technical and regulatory requirements to dismantle the reactor facility safely.
- The licensee has described in the DP methods to dispose of radioactivity from reactor operation that are acceptable.
- The DP shows that the licensee has the resources to complete the decommissioning so that NRC can consider release of the facility.

See Appendix 17.1 for details on how the reviewer should review correspondingly numbered section of the DP.

17.2 Possession-Only License Amendment

Possession-only license amendments are issued to remove the authorization to operate the reactor from the facility license. NRC generally issues a possession-only license amendment to a non-power reactor facility after the fuel is permanently removed from the core and shipped off site. No regulations require the issuance of a possession-only license amendment. Licensees of non-power reactors have gone directly from an operating license to an approved DP.

Before the new decommissioning regulations were promulgated in 1988 (53 FR 24018), possession-only license amendments were issued for the remaining term of the license, in some cases for many years. Without a regulatory requirement to complete decommissioning immediately, some amendments were renewed at least once. The new decommissioning regulations did not specifically address possession-only amendments, but 10 CFR 50.82(a) requires that an application for authorization to decommission be submitted within 2 years after reactor operations cease permanently. Possession-only amendments that were issued by the NRC staff before July 27, 1988, have been allowed to remain in effect for the term of the license. In some cases, if, because of factors similar to those in 10 CFR 50.82(b)(1)(iii), the licensee can show good cause for delaying the start of decommissioning, possession-only licenses issued before July 27, 1988, have been renewed. The purposes of a possession-only license amendment are discussed in this section of the format and content guide.

17.2.1 Review of the Application for a Possession-Only License Amendment

Areas of Review

The staff will review the possession-only amendment for the following technical and safety topics:

- changes made to the facility license, and if requested by the licensee, to the technical specifications and emergency, physical security, and operator requalification plans
- management and organizational structure required to possess the facility safely
- methods for limiting personnel exposures to radiation during the period of possession

Acceptance Criteria

To be found acceptable, the format of the possession-only amendment should contain the topics addressed in the format and The acceptance criteria for the information on the application for the possession-only license amendment include the following:

- The technical content of the possession-only license amendment application should be detailed and accurate to demonstrate, with reasonable assurance, that the health and safety of the facility staff and the public will be protected during the period of possession.
- The application should contain an explanation of the need for the possession-only license amendment, a request for a specific duration, a discussion of the activities to be performed, an outline of the schedule for the time the amendment is in effect, and an explanation of any factors beyond the control of the licensee that affect the schedules.
- The licensee should commit to a date to submit the applications for authorization to decommission and for license termination.
- The application need only include (1) a proposal from the licensee to modify or remove the paragraph in the license that authorizes the reactor to operate and (2) a safety analysis.
- The application may contain, at the discretion of the licensee, proposed additional license conditions; proposed technical specification, proposed physical security, emergency, and operator requalification plans; and additional safety analysis.
- The possession-only safety analysis should contain discussions, descriptions, and analyses of the shutdown facility.
- The safety analysis should state or refer to existing provisions needed to control reactor-related radioactivity and to protect the health and safety of the public.
- If the reactor was shut down unexpectedly because of factors beyond licensee's control, the application should describe the circumstances. These factors could include, for example, abrupt budgetary constraints or equipment failures.

Review Procedures

The reviewers should evaluate the possession-only license amendment application for completeness and technical accuracy. The reviewer should compare the analyses and discussions in the application with appropriate document guidance and, as applicable, with NRC-approved possession-only license amendments for similar facilities. For the review of physical security and emergency plans, the reviewer should adhere to the procedures discussed in Sections 6 and 7 of Appendix 17.1 of this review plan.

Evaluation Findings

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

- The application shows that the licensee has complied with the technical and regulatory requirements to possess the reactor facility safely.
- The licensee has proposed acceptable changes to the license (*if requested, to the technical specifications and emergency, physical security, and operator requalification plans*).

17.2.1.1 Facility License

The licensee should propose changes to the facility license consistent with the possession-only status. The authority to operate the facility should be removed from the facility license. Either the authorized maximum thermal operating power of the facility and the allowable pulse insertion, if applicable, should be reduced to zero or the paragraph concerning reactor operation should be removed from the license. If fuel has been removed from the site, the authorization to possess the reactor fuel should be removed from the paragraph authorizing the possession and use of special nuclear material (SNM). The authority to possess any other SNM or byproduct material that is specifically authorized by the facility license and that has been removed from the site or transferred to another license held by the decommissioning licensee or another licensee should be removed from the facility license. If the physical security plan is being amended, the reviewer may suggest that the paragraph in the license about this plan be amended to reflect the latest version. The expiration date of the license should not be changed.

17.2.1.2 Technical Specifications

If the licensee proposes changes to the technical specifications, they should be based on conditions analyzed in the possession-only safety analysis. The changes to the technical specifications should consist mainly of removing selected specifications rather than adding new specifications. The safety limits and limiting safety system settings may be removed from the possession-only license technical specifications because the reactor is no longer authorized to operate. The technical specifications should contain limiting conditions of possession analogous to the limiting conditions for operation for an operating reactor. Many of the limiting conditions for operation can be removed from the technical specifications or relaxed because the facility will not operate again. These specifications should include a list of radiation instruments necessary to monitor and provide minimum indications of migration or other unplanned changes in radiological conditions. Corrosion or degradation of equipment that confines radioactivity should be monitored. To limit corrosion, potentially radioactive liquids should be monitored for acceptable pH and electrical conductivity. There should be provisions for circulating coolant if coolant is required to remove decay heat or to control water quality. There should be provisions for detecting the loss of coolant if coolant is required for radiation shielding or for protection of the fuel.

If the fuel has not been removed from the facility, the technical specifications requirements for preventing criticality (core monitoring instrumentation and control rod inspection) and ensuring cladding integrity (fuel inspection and water quality) need to remain.

The technical specifications should contain a surveillance section that should include surveillance provisions for the limiting conditions for possession discussed above. The surveillance specifications should state the conditions or equipment monitored, the procedures, the instruments, the schedule, and the acceptable limits, unless they are given in the limiting operational conditions for possession.

The technical specifications should contain a section on facility description that should concisely specify the boundaries of the restricted area to which the possession-only license amendment will apply and other design features that are still applicable to the possession-only license.

The administrative section of the technical specifications would change little from that in the technical specifications for an operating reactor. This section should give the information necessary to ensure continued management of the facility and should describe personnel and programs for specified surveillance and maintenance activities.

Further detail on the content of technical specifications is given in this section of the format and content guide. The reviewer should also refer to Chapter 14 of the format and content guide.

17.2.1.3 Emergency, Physical Security, and Operator Requalification Plans

Amending an operating license to a possession-only license may allow major changes in emergency and physical security plans if the fuel is removed from the site. However, the regulations do not allow automatic elimination of the plans. See this section of the format and content guide and Sections 6 and 7 of Appendix 17.1 of this review plan for a complete discussion.

The reviewer should verify that the licensee has implemented for each plan one of the following:

- The licensee has not requested changes to the plan and will follow the requirements of the plan.
- The licensee has not made changes to the plan that decrease its effectiveness under the requirements of 10 CFR 50.54.
- For changes to the plan that decrease its effectiveness, the licensee has justified the changes and submitted them to NRC for review.
- The licensee has applied to NRC for an exemption from the regulatory requirements to maintain the plan.

If fuel has been removed from the facility and the possession of fuel is no longer authorized, licensed reactor operators are no longer required and the licensee may request that the requalification plan be eliminated. However, to move reactor fuel requires the presence of a qualified senior reactor operator or a licensed fuel handler.

17.2.1.4 Possession-Only License Amendment Safety Analysis

Even though the potential radiological hazards to the workers and the public are expected to be fewer for a permanently shutdown non-power reactor, the licensee should carefully consider them. These hazards should be discussed and analyzed in the possession-only amendment safety analysis. The primary purpose of this safety analysis is to show that the facility can be possessed in a way that protects the health and safety of the facility staff, the public, and the environment.

If the licensee has requested only to delete the authority to operate the facility without other changes in the license or technical specifications, the safety analysis

need only state that all requirements of the license and technical specifications remain in effect and that the reactor will not be operated again.

For other changes to the facility license and technical specifications, the major issue is ensuring that license and technical specifications requirements proposed for elimination or relaxation are justified. For example, the license may propose elimination of the requirement for periodic power calibration on the grounds that the reactor will never operate again and power calibration would be meaningless under these circumstances.

Work is allowed to characterize the facility for developing the DP. Proposed characterization activities may be described in the possession-only safety analysis and limited by technical specifications. Dismantling a reactor is not permitted by a possession-only license amendment, so the reactor facility described in the possession-only safety analysis should be similar to the operating reactor facility, and the SAR should be referred to as much as possible. However, many of the components, instruments, and systems need not remain operable in the facility. Therefore, the possession-only safety analysis should state which of the systems covered by the SAR will remain operable and which will be modified or deactivated.

The possession-only safety analysis should address potential accidents and provisions for limiting their consequences, if necessary. However, the acceptable accident scenarios for a possession-only safety analysis should be limited to ones credible for a shutdown reactor and for the residual radioactive material. To be found acceptable, postulated accidents should not subject the public or the workers to undue radiological exposure and should not exceed applicable regulatory limits (see Chapter 13 of this review plan).

This discussion also applies to a shutdown reactor with fuel on site. Credible accidents that involve the release of fission products from the fuel should be analyzed. Radiological exposure from these accidents should meet the guidelines in Chapter 13 of this review plan.

The experience and qualifications of the personnel required to manage a permanently shutdown reactor may be different from those required during the operating life of the reactor. The possession-only safety analysis should discuss the proposed changes in staff characteristics and other administrative requirements.

The reviewer should refer to the appropriate chapters of this review plan for specific guidance on reviewing such topics as radiation safety and accident analysis.

17.2.1.5 Changes to Facility Without License Amendment

Because 10 CFR 50.59 applies only to changes under a license authorizing operation, the licensee should propose a method for making operational or procedural changes under the possession-only license amendment. An acceptable method would be similar to the method allowed by 10 CFR 50.59 and discussed in Section 9 of Appendix 17.1 of this review plan.

Appendix 17.1

NRC Review of Decommissioning Plans for Non-Power Reactors

NRC Review of Decommissioning Plan for Non-Power Reactors

1 SUMMARY OF PLAN

In this section of the DP, the licensee should clearly present of the status of the facility, the overall objective of the decommissioning proposal, the approximate schedule for reaching significant milestones, and a summary of the important historical events that affect the decommissioning methods, approach, and schedules. An estimate of total cost and source of funds for decommissioning should be given (see Chapter 15 of this review plan). Although an estimate should already have been submitted by the licensee and reviewed by NRC, the reviewer should submit the financial information to the Inspection and Licensing Policy Branch in the Office of Nuclear Reactor Regulation for review. This section of the DP should also contain a brief discussion of quality assurance (QA) plans. QA activities should be described in detail in subsequent sections of the DP where dismantlement, decontamination, and decommissioning are discussed.

2 DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Alternative

The Commission's regulation in 10 CFR 50.82(b)(1) requires that the DP include a discussion of an alternative picked for decommissioning a reactor facility (DECON, SAFSTOR, or ENTOMB). In most cases, the requirements in 10 CFR 50.82(b)(1)(ii) and (b)(1)(iii) limit the choice for non-power reactors to the DECON alternative, which requires that decommissioning activities be completed without significant delay unless prevented by factors beyond the licensee's control. Therefore, the guidance in this appendix is based on the selection of the DECON option. If factors beyond the licensee's control, as noted in 10 CFR 50.82(b)(1)(ii), delay dismantlement and decontamination, a limited period of SAFSTOR would be acceptable. Some factors for delay of decommissioning that may preclude the DECON option are continued operation of another non-power reactor at the facility, unavailability of radioactive waste disposal facilities, and the need to protect the health and safety of the public.

If more than one non-power reactor is in a containment or a confinement or is located at a site so that systems are shared, the reviewer may find the SAFSTOR option acceptable for one reactor if the other reactor is to remain in operation. In this case, when the operating reactor ends operations, both reactors should be decommissioned without delay. It is possible that a particular licensee, because of facility location, may not have access to a disposal site for radioactive waste for some time. Again, the reviewer may find the SAFSTOR option acceptable if the

licensee continues to seek access to a waste disposal facility and decommissions the reactor without delay when this facility is found. Another reason to delay decommissioning may be that a radioactive material site characterization study shows that the potential exists for unacceptable radiation exposure to the public, workers, or the environment if dismantlement were to take place immediately. In this case, the licensee may justify to the NRC staff that a period of SAFSTOR is necessary to allow radioactive components to decay. Decommissioning should proceed without delay once enough time has passed to allow sufficient decay of components. It is anticipated that only test reactors would need substantial decay time.

If the licensee proposes temporary storage of radioactive material, it should present technical justification. Temporary storage may be acceptable if the justification clearly shows that the licensee has chosen such storage after substantial and careful review.

The reviewer shall not accept SAFSTOR if the licensee has chosen it in order to defer the planning and the decision process.

2.2 Facility Radiological Status

2.2.1 Facility Operating History

The licensee should discuss the operational history of the facility and explain how the history affects the inventory of radioactivity. Experience in decommissioning non-power reactors has shown that most of the radioactivity results from neutron irradiation of structural, shielding, experiment-related, or core-related components. Therefore, the licensee should use the operational history and characterization measurements to estimate the irradiation exposures and calculate the radionuclide content of principal materials and known impurities. The licensee should discuss operating power and time histories.

The licensee should discuss all radioactive spills, fuel failures, and component failures that led to a release of radioactive contamination or contaminated components. The reviewer should study current facility drawings and the SAR to determine the applicability and completeness of the discussion. The reviewer should also study annual reports and licensee event reports that are in the docket files to determine the completeness of the history presented by the licensee. The reviewer should ensure that this historical information is used in the development of the radiological status of the facility and the rest of the DP.

2.2.2 Current Radiological Status of the Facility

The licensee should discuss the types and quantities (curies) of radioactivity remaining at the facility at the time the DP was written. The reviewer should confirm that the information in this section is based on the facility history, surveys, calculations, and sampling performed to characterize the facility, including radioactive decay effects and any cleanup or decontamination tasks already completed. The licensee should discuss in detail the location of the radioactive materials and should explain the methods for analyzing the potential radiation exposure rates and doses to plant workers during surveillance, dismantlement, decontamination, packaging, and transportation.

For most materials, the thermal neutron fluences and absorption cross-sections should be used. However, the fast neutron exposures are important for some isotopes, and appropriate predictions should be included. In all cases, the individual isotopes and radionuclides of significance should be discussed. In some cases, these radionuclides may be beta emitters that have less of an impact on radiological safety than gamma emitters, but may be significant in terms of total curies of radioactive material. Because the origin of materials may influence the types and quantities of impurities, the reviewer should compare the discussion to DPs and decommissioning reports for similar reactors. Manion and LaGuardia (1980) discuss probable radionuclides.

2.2.3 Release Criteria

The licensee should state which release criteria will be applied to the decommissioning project. The reviewer should ensure that the criteria used conform with current guidance and should verify that the criteria have been factored into the planning of decommissioning activities and tasks as the minimum end point of decommissioning activities. In the termination survey, the licensee should show that the release criteria have been met.

2.3 Decommissioning Tasks

2.3.1 Activities and Tasks

The licensee should list the activities and tasks required to prepare the site and facility for dismantlement and decontamination. The reviewer should confirm that plans have been made for removing the following components:

- radioactive materials in storage or in laboratories
- control rod assemblies
- core support structures

- beam tubes and other experimental facilities
- tank or pool liner (after segmentation)
- thermal column structures
- primary coolant piping and systems (after segmentation)
- secondary systems
- concrete shielding
- other radioactive material

The activities should also include preparing and packaging the components for shipment, as necessary.

The licensee should give information about the expected radioactivity. The basis for this expectation should be supported from information in Section 2.2 of the DP. The plans for disposing of the radioactive material and the projected radiation exposures to personnel should also be stated.

The planning information in this section will be acceptable if all potentially radioactive components are discussed. The reviewer should refer to other decommissioning projects that have been successfully completed for guidance on dismantlement and decontamination methods that the staff has found acceptable for a particular situation. However, new approaches to a particular decommissioning problem may be acceptable if the licensee presents them in detail and demonstrates that the new approach can safely address the problem, taking into account regulatory requirements, release criteria, the ALARA (as low as is reasonably achievable) principle, and other DP requirements.

2.3.2 Schedule

The proposed schedule for accomplishing the tasks should be compatible with the quantity of radioactivity to be removed, taking into account its location and disposition. The schedule should include the methods and availability of equipment and systems required to extract, remove, and process materials. It should address the availability of staff and other resources to complete the schedule as proposed. The reviewer should compare documentation on schedules to that from similar dismantlement projects.

No regulatory requirement establishes a schedule for decommissioning a non-power reactor. However, 10 CFR 50.82(b)(1)(ii) and (b)(1)(iii) clearly require that decommissioning be completed without significant delay, unless delay is necessary to protect the health and safety of the public. Any delay in decommissioning should meet one of the criteria of 10 CFR 50.82(b)(1)(iii).

2.4 Decommissioning Organization and Responsibilities

This section should show a complete organization for implementing the DP, with points of interaction with the following:

- licensee management
- radiological protection organization
- facility safety committee
- any contractor organizations

The discussion should clearly show that the licensee will continue to be responsible for overall supervision, compliance with applicable regulations, and protection of the health and safety of the public. This section should also contain explicit information about the duties, responsibilities, and technical qualifications of persons in key positions in the decommissioning organization. At a minimum, the technical director of the organization should have an engineering or physical science degree and at least 2 years of combined experience in reactor operations or decommissioning, radiological protection, and radioactive waste management. Persons in key contractor positions should have comparable experience and qualifications.

The DP should show that the radiation protection organization has independent authority to stop work similar to the authority of the comparable organization during reactor operations.

The DP should show that the composition and authority of the decommissioning safety committee are similar to the reactor operations safety committee, but with increased expertise in radiological exposure and control. The DP should give details on the responsibilities of the committee, showing that it reviews and audits major decommissioning activities, proposed procedures, radiation exposure records, reportable occurrences, and changes to the DP. The discussion of the function and authority of the committee should show that it reports to high-level licensee management.

2.5 Training Program

The licensee should demonstrate its awareness that dismantlement and decontamination may be a new experience for its staff. Training is essential to prepare the staff to perform the decommissioning safely. The licensee should present a complete outline of training appropriate to the staff and contractor functions and positions, consistent with 10 CFR Part 19. The training program should also be outlined. The outline should contain the proposed technical

activities, the time allotted, and the depth of content of the training. The training program should contain provisions for evaluating the effectiveness of the training and associated recordkeeping. The training plan should present the qualifications and experience of the training personnel and should specify the methods for training, evaluating, and certifying contractor personnel.

2.6 Contractor Assistance

The licensee is permitted to use contractors to perform decommissioning activities. The DP should describe the role of contractors, the type of work to be performed, and the percentage of contractor effort in completing the process. The licensee should discuss any other non-power reactor decommissioning work or evaluation performed by the contractor and the results of that effort. The reviewer should review the decommissioning work that the contractor performed for NRC licensees and should study the inspection reports of those decommissioning projects to become familiar with contractor performance. The licensee should discuss the minimum technical and safety-related experience that the contract organization must demonstrate and the technical qualifications of the onsite director of contractor activities. Further, the licensee should identify the position in its decommissioning organization to which the contractor reports. The DP should explicitly state that the licensee is fully responsible for ensuring that all contractor activities are safely performed and comply with 10 CFR Part 20 and other applicable regulations, license conditions, and the decommissioning order and plan.

Contractor functions may include personnel protection services, demolition and dismantling, decontamination, post-disposal radiation surveys, transportation, training and recordkeeping. The licensee should discuss all functions to be performed by contractors and the method to oversee all of the contractors' activities.

2.7 Decontamination and Decommissioning Documents and Guides

The licensee should reference the applicable guides, standards, and reports used to develop the DP. Some of these documents are the following: International Atomic Energy Agency, 1986; NRC draft Regulatory Guide DG-1006; NRC IE Circular 81-07; NRC IE Information Notices 83-05 and 85-92; NRC letters dated March 17, 1981, and April 21, 1982; NUREG-0586; and NUREG/CR-1756 and Addendum 1.

The DP should show that the licensee will comply with 10 CFR Parts 20, 30, 50, 70, 71, and 73 (as they apply to decommissioning) and with applicable Department of Transportation (DOT) regulations, including 49 CFR Parts 170 through 189.

3 PROTECTION OF THE HEALTH AND SAFETY OF RADIATION WORKER AND THE PUBLIC

3.1 Radiation Protection

3.1.1 Ensuring As Low As Reasonably Achievable Radiation Exposures

The licensee should provide sufficient information to show that the facility managers have established a clear policy for all radiological activities to maintain exposures to the public, facility employees, and contractors as low as is reasonably achievable (ALARA). The licensee should show where the manager responsible for the ALARA program fits into the overall management structure and should explain how authority to implement the program is derived and applied. The ALARA program should include training, procedural reviews, walkthroughs, and criteria for minimizing occupational exposure and residual radioactivity. The reviewer should compare the DP ALARA program to the program that was in effect for reactor operations to help ensure that the DP ALARA program is complete.

3.1.2 Health Physics Program

To find the health physics program acceptable, the reviewer should confirm that the decommissioning radiation protection organization and program are consistent with those for facility operation because decommissioning is analogous to such activities that were performed during facility operation as major maintenance. Programs that have been decreased as a result of the granting of a possession-only license amendment would need provisions to be returned to a condition for active decommissioning. Because each non-power reactor licensee has developed a unique radiation protection program, the reviewer should study the SAR and DP and make a direct comparison. The program is found acceptable if the organization has comparable authority to operate independently, has access to top management, and is staffed with experienced and qualified health physicists. The program should include monitoring equipment, calibration schedules and procedures, and an effectively organized ALARA program. Operational plans for assessing and documenting radiation exposure and for controlling exposure during decommissioning activities should be described. This includes protection from

radiation fields, contamination, and airborne radiation hazards created during the decommissioning process.

3.1.3 Dose Estimates

For each major task in which radiation is a factor, the licensee should estimate the total cumulative and individual maximum dose equivalents to both radiation workers and the public. These estimates should take into account both external and internal exposures. Potential pathways and any doses to the public should be analyzed according to current practices. Methods to ensure compliance with 10 CFR Part 20 and the ALARA program should be explained. For each task identified by the licensee in Section 2.3.1 of the DP that involves radiation, a dose estimate should be given in this section of the DP. The reviewer should verify several representative dose determinations. The reviewer may gain insight into actual doses received during decommissioning activities by studying final reports from non-power reactors that have been decommissioned.

3.2 Radioactive Waste Management

3.2.1 Fuel Removal

Licensees are expected to properly remove all fuel under the reactor operating license before decommissioning begins. In this section of the DP, the licensee should state whether or not the fuel has been removed. If fuel will be removed as a part of decommissioning, the DP should give information about the removal. This information should include the defueling schedule, the methods of removing the fuel from the facility, the method of shipment, and the recipient of the fuel. If any fuel is still on site at the beginning of other dismantlement activities, the DP should contain a discussion of the license status and the way fuel integrity, physical protection, and emergency planning will be maintained. Note that decommissioning with fuel on site may require the analyses of new accidents or changes to previously analyzed accidents. The licensee should explicitly commit to compliance with all applicable regulations and guides, including 10 CFR Parts 20, 50, 71, and 73 and 49 CFR Parts 170 through 189.

3.2.2 Radioactive Waste Processing

The licensee should demonstrate that it has considered all of the radioactive sources discussed in Section 2.2.2 of the DP and has a plan for monitoring, sorting, packaging, and processing the wastes. If there are liquid radioactive wastes, methods for controlling and processing them, such as solidifying waste for shipment, should be discussed. If diluting waste for disposal into a sanitary waste system is proposed, the licensee shall discuss in detail assessment methods and

controls to comply with 10 CFR Part 20. Any gaseous or airborne radioactivity should be confined, filtered, and diluted for controlled release under 10 CFR Part 20. The licensee should discuss the use of controls, such as hoods, tents, face masks, or self-contained air packs. Control of radioactivity during demolition, segmentation, and compacting of solid radioactive wastes should be ensured during processing for disposal.

3.2.3 Radioactive Waste Disposal

The licensee should account for all of the radioactivity discussed in Section 2.2.2 of the DP. The licensee should state the regulations that are applicable to the various radiation sources. It should give details for the disposal site, types and quantities of radioactivity expected, methods of verifying and certifying the radioactive content, formal agreements required for transport and release of the radioactive waste to a designated licensee, and methods of transport. The DP should state that the licensee is responsible for properly packaging and labeling all radioactive waste for transport. The licensee shall include a commitment to comply with all applicable regulations (10 CFR Parts 20 and 71 and 49 CFR) and should list specific references.

The licensee should describe in detail, and with quantitative criteria, how material to be disposed of in unrestricted sites will be monitored, selected, and controlled. The only criterion acceptable to NRC at present is that no measurable radioactivity above background levels should be disposed of in unrestricted locations (NRC IE Circular 81-07 and NRC IE Information Notices 83-05 and 85-92). The licensee should give acceptable criteria for measuring and ensuring that condition, with limits defining sensitivity of instrumentation.

3.2.4 General Industrial Safety Program

Although NRC does not directly regulate or license nonradiological activities except as they would affect radiological health and safety, an agreement exists between the Occupational Safety and Health Administration (OSHA) and NRC concerning nonradiological activities. The DP should show that the licensee is aware of and responsible for controlling any nonradiological hazards or releases that ensue from decommissioning activities so that radiological work activities can be safely accomplished in accordance with the DP.

3.3 Radiological Accident Analyses

Fuel is not usually present on site when decommissioning begins. If this is the case, radiological accident scenarios need only address the residual radioactive material. The licensee should address potential accidental exposures during the

removal, processing, and packaging of decommissioning wastes, which may include liquid radioactive waste, airborne radioactivity, and solid parts, components, and other materials. Some examples of accidents are contaminated chemical solvent spills and fires, failures of filters and vacuum filter bags, and combustible radioactive material fires. The reviewer should judge the adequacy of the accident analyses on the basis of experience with the dismantlement of similar structures. The reviewer should assess potential radiation exposures using the same techniques as those used for assessing releases during normal operation and accident conditions in operating non-power reactors.

4 PROPOSED FINAL RADIATION SURVEY PLAN

After removing or reducing the reactor-related radioactivity, the licensee should prepare a detailed set of measurements to verify that the residual radioactivity can conform to the DP release criteria. The DP should include the proposed plan for performing these measurements, analyzing the results, interpreting the results, and relating those results to the release criteria for terminating the license. In the DP, the licensee should commit to submit these results in a comprehensive report to NRC when it requests the NRC inspection confirming that the criteria have been met. The reviewer should confirm that the final survey plan addresses the following:

- Summary of the location and types of all residual radioactivity.
- Scope of methods to measure and determine the relevant radiological parameters, including background radiation.
- Discussion of methods, statistical or otherwise, to ensure that essentially all residual sources of radiation are found and quantified.
- List of instruments and discussion of measurement techniques used.
- Procedures for calibrating instruments to ensure reliability and accuracy.
- Methods of documenting results that are available, retrievable, and comprehensible.
- References to publications that address the quantitative measurement of low exposure rates and low concentrations of radioactivity. (Note that NUREG/CR-2082 and its references are a reasonable starting point, but more recent publications might be applicable.)

The reviewer should verify that the final survey plan demonstrates that the surveys require techniques other than the usual ones used for personnel protection at the

facility. Such notations as "not detected" are not acceptable. Instead, the licensee should propose quantitative measurements and scientifically derived, valid limits of error. The reviewer should evaluate statistical inferences to locations not directly measured and should compare them to accepted methods.

At non-power reactors, radioactive material may exist in two forms. The material may be surface or absorbed contamination or neutron-irradiated material distributed within the body of a component. In the DP, the licensee should address reducing both sources of radiation. Guidance for surface contamination levels can be found in Regulatory Guide 1.86, Table 1. Current NRC criteria for release of non-power reactor facilities for unrestricted use are discussed in Section 17.1.4 of the format and content guide. Release criteria may change with time. The reviewer should verify that the guidance given in the format and content guide is current and that the licensee is using the most recent release criteria.

NRC regional staff or an NRC contractor will conduct an onsite survey to measure and verify the radiation exposure and contamination levels. When the requirements for release have been satisfied, NRC will issue an order that terminates the license and any further NRC jurisdiction over the facility.

5 TECHNICAL SPECIFICATIONS

After the fuel is shipped off site under the facility operating license, most of the technical specifications for the operating license should no longer apply. Therefore, the licensee should have requested relief by applying either for a possession-only license amendment or for authorization to dismantle and decommission the facility. In either case, the licensee should propose a modified set of technical specifications that contains safety controls and limitations that ensure continuous protection of the health and safety of the public and satisfy DP criteria. The format of the technical specifications for decommissioning should conform to the appropriate sections of ANSI/ANS 15.1-1990, as discussed in Chapter 14, "Technical Specifications," of the format and content guide, and should include the following:

- a section imposing limiting decommissioning conditions at the facility comparable to the limiting conditions for operation for required equipment and operational conditions
- a section providing for surveillance of the equipment and conditions
- a section describing the residual defueled facility and site to which the decommissioning order will apply

- an administrative section that outlines the management structure, provides for review and audit functions, provides for development and use of necessary procedures, and contains reporting and record-retention requirements

Technical specifications for decommissioning may also control worker access to high-radiation areas, release of effluents, and availability of appropriately calibrated radiation monitors. These technical specifications should require procedures to monitor materials and components before they are removed from controlled areas and procedures for defining and maintaining restricted areas. In addition, they should allow margins for flexibility or unanticipated events.

6 PHYSICAL SECURITY PLAN

If the fuel is shipped off site before or at the time decommissioning activities are authorized, the licensee may request relief from the physical security plan required by 10 CFR 50.54(p). However, appropriate access control measures will be required to limit radiation exposures to the workers and the public in accordance with 10 CFR Part 20. It is acceptable to include necessary access controls in the revised technical specifications. Security experts at the regional office or at headquarters should perform the review of this information in the DP. The reviewer should verify that the DP fully addresses this issue and contains a plan for implementing continuous and acceptable controls until the facility license is terminated by NRC.

7 EMERGENCY PLAN

The regulations require a facility to have an emergency plan as long as it has a license. However, if the fuel, which is the major source of radioactive material on site, is removed, accidents that involve enough radioactive material to reach the emergency action levels spelled out in the plan may be impossible. Fuel removal may allow a major reduction in the level of emergency preparedness required at the facility. Under 10 CFR 50.54(q), the licensee, without prior NRC approval, may make changes to the emergency plan that do not reduce its effectiveness. Changes that reduce the effectiveness of the plan cannot be made under 10 CFR 50.54(q) and shall be submitted to NRC for approval. The licensee may seek relief from 10 CFR 50.54(r) if the emergency plan no longer applies. To completely eliminate the requirement for the emergency plan, the licensee should apply for a specific exemption and should address all factors identified in 10 CFR 50.12. The reviewer should send requests for changes to the emergency plan to emergency planning specialists at headquarters for review.

8 ENVIRONMENTAL REPORT

In accordance with 10 CFR 51.21, the NRC staff shall prepare an environmental assessment for the proposed decommissioning of a non-power reactor and the termination of the facility license. The assessment shall be written in compliance with 10 CFR 51.30 and should be based on information submitted by the licensee in an environmental report (ER). The ER can be a section of the DP, or it may be a separate report submitted with the DP. To the extent possible, the content of the ER should follow 10 CFR 51.45. The ER should also address the items discussed in Section 8 of Appendix 17.1 of the format and content guide. Because most decommissioning activities and terminations for non-power reactors have the same basic environmental impact, the staff has developed a standard format for environmental assessments (EAs) and, if applicable, findings of no significant environmental impact for decommissioning and license termination. The reviewer should refer to recently issued orders for an example of the format. A separate EA and, if applicable, a finding of no significant environmental impact shall be prepared for the issuance of the decommissioning order and for the license termination order.

9 CHANGES TO THE DECOMMISSIONING PLAN

Because not all events and discoveries during decommissioning are predictable, the DP may need to be changed. The DP should contain procedures for making changes similar to those in 10 CFR 50.59 for operating reactors. The procedures should be clearly presented in the DP or technical specifications. More information can be found in License Amendment 85 for the Fort St. Vrain facility (Docket No. 50-267), dated November 23, 1992.

10 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1981.

Branagan, E. F., and F. G. Congel, "International Symposium on Decontamination and Decommissioning," Oak Ridge, Tennessee, 1985, papers by Paul G. Voilleque, R. A. Wynveen, et al

Code of Federal Regulations, Title 10, "Energy," and Title 49, "Transportation," U. S. Government Printing Office, Washington, D.C., revised periodically.

International Atomic Energy Agency, "Safety in Decommissioning Research Reactors," Safety Series No. 74, Vienna, Austria, 1986.

Manion, W. J., and T.S. LaGuardia, "Decommissioning Handbook," DOE/EV/10128-1, prepared for U.S. Department of Energy, November 1980.

U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1006, "Records Important for Decommissioning of Nuclear Reactors," September 1989.

U.S. Nuclear Regulatory Commission, IE Circular 81-07, "Control of Radioactively Contaminated Material," May 14, 1981.

U.S. Nuclear Regulatory Commission, IE Information Notice No. 83-05, "Obtaining Approval for Disposing of Very Low-Level Radioactive Waste 1—10 CFR 20.302," February 24, 1983.

U.S. Nuclear Regulatory Commission, IE Information Notice No. 85-92, "Surveys of Wastes Before Disposal From Nuclear Reactor Facilities," December 2, 1985.

U.S. Nuclear Regulatory Commission, letter from John F. Stolz to Dr. Roland A. Finston, Stanford University, "Use of the 10 mrem per year release criteria," March 17, 1981.

U.S. Nuclear Regulatory Commission, letter from James R. Miller to Dr. Roland A. Finston, Stanford University, "Use of the 10 mrem per year release criteria," April 21, 1982.

U.S. Nuclear Regulatory Commission, NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," August 1988.

U.S. Nuclear Regulatory Commission, NUREG/CR-1756, "Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors," March 1982, and Addendum 1, July 1983.

U.S. Nuclear Regulatory Commission, NUREG/CR-2082 "Monitoring for Compliance With Decommissioning Termination Survey Criteria," July 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," 1974.

18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

The conversion of a non-power reactor from the use of highly enriched uranium (HEU) to the use of low-enriched uranium (LEU) as a fuel was mandated by 10 CFR 50.64, which was published in the *Federal Register* (51 FR 6514) on February 25, 1986. The general basis for the regulation was that HEU posed a nuclear weapon proliferation risk through theft or diversion, and that limiting the availability of HEU would also limit this risk, promote the common defense and security, and protect the health and safety of the public.

The regulation stipulates that NRC will effect necessary amendments to the operating licenses of non-power reactors through an enforcement order. The use of an enforcement order places the responsibility to defend the order on NRC and not on the licensee. Therefore, NRC requires information and analyses from the licensee that support the conclusions that the health and safety of the public will continue to be protected during operation with LEU. Such a conclusion should be reached if no applicable acceptance criteria are exceeded by the conversion of the fuel from HEU to LEU. The following considerations may also influence the safety-related operating characteristics of the LEU-fueled reactor:

- A principal premise of 10 CFR 50.64 is that the use-related performance characteristics of the reactor are not to be degraded significantly by the fuel conversion. (See NRC Statement of Policy, No. 82-127, August 24, 1982; and NRC Proposed Rule, *Federal Register*, Vol. 49, No. 131, July 6, 1984, p. 27769.)
- Some design limitations may be caused by physical principles or other factors of the LEU-fueled reactor that are not subject to the licensee's control. For example, a generic LEU fuel plate design was adopted by joint concurrence of the U.S. Department of Energy (DOE), NRC, and licensees. Conversion to this fuel could require significant changes in the core geometry, core physics, and safety analyses of a particular reactor. Modifications to the reactor systems, procedures, or technical specifications required by the fuel conversion usually will be acceptable; however, the licensee should compare the safety-related characteristics of the HEU- and LEU-fueled reactors and should analyze all changes.
- Added features may be needed to ensure that the applicable acceptance criteria continue to be met.
- Modifications to the reactor not required by the conversion, but made to improve reactor capability beyond that of the HEU-fueled reactor, or done for reasons other than safety, will generally not be acceptable under the

NRC order. These changes should be requested by license amendment under 10 CFR 50.90, 10 CFR 50.91, 10 CFR 50.92, or 10 CFR 50.59.

Background material follows for the different reactor fuels used to date:

- For training reactor, isotope production, General Atomics (TRIGA) reactors, the fuel was developed and marketed by a sole source, General Atomics (GA), and since about 1962, the neutronics characteristics have been reasonably constant. The earlier 8.5 weight percent (w/o) uranium loading of 20-percent enriched uranium can be replaced by the modern 20 or 30 w/o, (20 % enriched) fuel, if changes in neutron flux and power peaking are adequately analyzed and considered. In NUREG-1282, NRC has reviewed and accepted LEU fuel for TRIGA reactors.
- For plate-type fuel [materials testing reactor (MTR) type], the designs of the original and replacement fuels were not standardized among the various facilities and various reactor designers. The variations were primarily related to the physical dimensions of the plates and water spaces, as well as to the uranium density of the fuel. As a result of these variations, core sizes (i.e., number of fuel elements) vary by a factor of almost 2.

After some discussion, DOE, NRC, and licensees selected a single generic LEU fuel plate made of U_3Si_2-Al . NRC generically evaluated and accepted an MTR fuel plate design in NUREG-1313.

This plate design could lead to extensive changes in the core design of some reactors. Therefore, the licensee should explain the factors involved in modifying the facility or its operating characteristics and provide applicable safety analyses. The characteristics of the approved fuel plate could be sufficiently different from the existing HEU fuel plate for a particular reactor that changes to the reactor would be required. The licensee should provide the technical reasons for the required changes and an analysis of the effect of the changes.

NRC has determined that the conversion to LEU should not result in a significant reduction in the nuclear capability of the reactor; nor should it increase the nuclear capabilities of the reactor with a significant decrease in safety margin. Therefore, conversion to LEU fuel should allow licensees of non-power reactors to maintain the current nuclear capabilities if no applicable acceptance criterion is exceeded and to state that the change is needed to accommodate the design differences for the LEU core and fuel.

Reference to a chapter other than this one (Chapter 18) alludes to another chapter in this standard review plan. Reference to a section alludes to a section in Appendix 18.1 to this chapter.

Additional guidance may also be found in Part 1 of this document as well as in other chapters of Part 2. Also, the licensee can consult such documents as the International Atomic Energy Agency (IAEA) documents IAEA-TECDOC-223, -324, and -643, for additional guidance.

The numbering system in Appendix 18.1 corresponds to the numbering system in the conversion SAR.

Appendix 18.1

Standard Review Plan and Acceptance Criteria for HEU to LEU Conversion

Standard Review Plan and Acceptance Criteria for HEU to LEU Conversion

1 GENERAL DESCRIPTION OF THE FACILITY

1.1 Introduction

The reviewer should verify that the introduction contains a brief overview of the plans for conversion of the reactor from HEU to LEU fuel, and that it summarizes the physical, nuclear, and operational characteristics of the facility with the LEU fuel. It should provide reference to where in the SAR changes are discussed. Additionally, it should provide comparisons and references to the existing reactor described in the HEU SAR. The reviewer should verify that Sections 1.1 through 1.5 establish an outline of the primary considerations for conversion to LEU fuel.

1.2 Summary and Conclusions of Principal Safety Considerations

The reviewer should confirm that this section briefly discusses all significant issues that were considered in the plans and designs for converting the reactor to LEU fuel. All issues with safety significance should be emphasized, especially if any margins in the safety analysis would be decreased.

Changes in the safety analyses should be discussed briefly, with references to the sections in which the detailed analyses are presented and in which the HEU- and LEU-fueled reactor operating characteristics are directly compared.

1.3 Summary of Reactor Facility Changes

The reviewer should confirm that this section summarizes any proposed changes to the site and the reactor facility for the conversion to LEU fuel. Both the HEU and the LEU fuel should be briefly described and compared. References should be provided for the detailed fuel design, performance discussions, and other significant documents. If the proposed LEU fuel has been reviewed and evaluated by NRC, the acceptance documents should be referenced (e.g., NUREG-1281, -1282, or -1313). The reviewer should verify that this brief discussion explains why the proposed LEU fuel is appropriate for the subject reactor and that the discussion addresses other proposed changes to the reactor and states why the changes are required by the fuel conversion.

1.4 Summary of Operating License, Technical Specifications, and Procedural Changes

The reviewer should verify that any changes, as a result of the fuel conversion, proposed to the facility operating license, the technical specifications, or operating procedures are briefly listed with a reference to the SAR sections in which they are discussed in more detail. Such specific changes may include (1) the uranium-235 possession limits for LEU fuel in the license, (2) license provisions to possess both HEU and LEU fuel at the same time if need be, and (3) technical specifications changes, including such LEU fuel design parameters as dimensions, materials, enrichments, number of plates and elements, heat fluxes and power densities, coolant flow rates, and reactivity coefficients.

1.5 Comparison With Similar Facilities Already Converted

The reviewer should ensure that the licensee has compared its proposed LEU reactor with similar non-power reactor facilities that have already undergone a conversion from HEU to LEU fuel, if there are any such similar reactors. The licensee should have identified (1) the similarities and differences between the facilities in design, construction, testing, and operation; (2) any problems that were identified and resolved at the other facilities; and (3) plans to address such problems at their facility.

2 SITE CHARACTERISTICS

Areas of Review

This section of the SAR should describe the environmental factors that could affect the suitability of the site for the location and operation of the non-power reactor. The areas of review should include the geography, demography, meteorology, geology, seismology, hydrology, and nearby transportation, industrial, and military facilities, and their interaction with the reactor. These interactions are of two general types:

- (1) the effects that the reactor could have on the environment and the population surrounding the reactor site
- (2) the effects that environmental factors could have on the reactor facility that affect its safe operation or shutdown

Generally, there should be no change to these topics with a conversion from HEU to LEU fuel. However, it is possible that new information has become available

since the last SAR revision. For example, it is possible that demography, and nearby transportation, industrial, and military facilities around the reactor facility have changed since the last SAR revision. If these changes are significant to the safety analysis of the reactor or if the licensee wants the SAR to be current, these new factors should be justified by the licensee and submitted to NRC as an update to the HEU SAR before the LEU SAR is submitted. Changes to the site characteristics due to the passage of time or due to the discovery of new information unrelated to the conversion cannot be approved by NRC using the conversion order. The reviewer should ensure that the conversion application does not result in the NRC approving changes to site characteristics by enforcement order.

Most non-power reactors release small quantities of radioactive argon-41 to the unrestricted area during normal operations. Any changes in this and other radioactive effluents resulting from the proposed conversion should be considered in this section and analyzed in Section 11. Comparisons, between the existing HEU-fueled reactor and the proposed LEU-fueled reactor, of potential doses to the public considering current population distributions should be made in Section 11 of this safety analysis, as well as comparisons with regulatory requirements to account for any changes since the most recent revision of the SAR.

Information should be provided on any environmental effects at the site that have the potential for causing damage, or other significant effects due to the conversion to the proposed LEU-fueled reactor. If changed, the information should include analyses of the impacts on reactor safety and response to such site-dependent factors as hydrology, geography, demography, geology, meteorology, seismology, and manmade accidental events. The analysis of these changes may be listed in this section and referenced to other sections of the SAR where the design or operating criteria of the specific systems, structures, or components are analyzed to address such changes.

Unless the information indicates significant changes in environmental impacts, a separate environmental report need not be provided.

Acceptance Criteria

The site characteristics should be specified to be considered in the facility design changes as documented in subsequent sections of the SAR. Additional general guidance may be found in ANSI/ANS 15.7. Since changes in site characteristics should have already been considered for the HEU reactor, there should not be any significant difference between HEU and LEU site characteristics.

Review Procedures

The reviewer should compare any changed environmental interactions between the converted LEU facility and the existing HEU facility. The reviewer should evaluate the discussions in the licensee's submittal about any changed interactions with characteristics of the site, either in kind or in magnitude, that results from the fuel conversion. The conversion of the reactor should not change the conclusions about the suitability of the site for operation of the reactor with the LEU fuel. Changes in the impact on site characteristics due to conversion may be compensated for by changes in facility design which should be listed and referenced to applicable sections of the SAR.

Evaluation Findings

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

- The site characteristics and associated environmental hazards will not be significantly changed by the reactor conversion, so there will be no decrease in the suitability of the present site for continued operation of the reactor. The conclusions of site suitability reached in the SAR and the safety evaluation report for the HEU operating license remain valid for the LEU-fueled reactor, as proposed.
- The impact on the site characteristics and associated environmental hazards due to conversion have been identified. Facility structures, systems, and components have been analyzed in subsequent sections of the safety analysis to demonstrate acceptable design to compensate for changes in site impact.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Areas of Review

This section of the SAR should contain information on the design bases of facility structures, systems, and components, and a discussion of any modifications in the design bases for the HEU to LEU conversion. This section should reference all significant standards and guides applicable to the conversion, including rules and regulations, staff reports (NUREGs), regulatory guides, and the American National Standards Institute/American Nuclear Society (ANSI/ANS) standards used to design all proposed modifications to the reactor facility. This section

should also describe how the facility is designed to accommodate potential wind, water, and seismic damage.

The information should note the subsequent sections in the SAR in which the modifications to design bases are used. It should consider the safety-related consequences of all of the natural and manmade external forces and events noted in Section 2 of the SAR, and how the structures, systems, and components limit the potential consequences of operation to the environment and the surrounding population. The emphasis should be to compare the design requirements from the proposed LEU-fueled reactor with the HEU-fueled reactor. The reviewer should evaluate any design changes from conversion-related changes in structures, systems, components, and reactor operating characteristics.

It is not expected that HEU to LEU conversions should raise the need for design bases changes to structures, systems, and components. If no changes are made to the design bases for structures, systems, and components, the SAR should so state and the reviewer should verify that there is no change.

Acceptance Criteria

Acceptance criteria should be based on the following considerations:

- The design bases for the facility structures, systems, and components have been compared and are the same for the HEU and LEU facilities.
- Changes in design bases for the LEU-fueled reactor facility structures, systems and components from those of the HEU facility were noted. The associated sections in which these changes in the design bases are used in the SAR is specified.

Review Procedures

The reviewer should evaluate design bases changes to structures, systems, and components required by conversion from HEU fuel. The reviewer should ensure that changes are compared between the HEU and LEU facilities, and should verify that reference is made to the associated safety analyses in subsequent sections.

Evaluation Findings

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

- The design bases changes in structures, systems, and components of the facility required by the conversion from HEU fuel are specified for use in the subsequent, applicable SAR sections.
- The LEU design bases for structures, systems, and components have not changed from those previously established for the HEU facility. This is acceptable for the LEU-fueled reactor facility as these design bases will be verified in subsequent, applicable SAR sections.

4 REACTOR DESCRIPTION

4.1 Reactor Facility

This summary description should provide the reviewer with a brief overview of the design and operating characteristics of the existing HEU-fueled reactor and comparisons with the proposed LEU-fueled reactor. Any significant safety-related modifications should be emphasized and the reasons they are required for the conversion should be summarized. Physical changes in the fuel characteristics and the balance of the reactor should be described briefly. This section should contain a discussion as to whether only the fuel will be changed and whether any significant operational characteristics or procedures will be changed for the proposed LEU-fueled reactor. The detailed discussions and analyses of these changes should be reviewed and evaluated in appropriate sections of this safety analysis.

4.2 Reactor Core

Areas of Review (Sections 4.2.1–4.2.6)

In converting a non-power reactor from HEU to LEU fuel, most of the physical and operational changes are expected to be associated with the reactor core. The core includes the fuel, the neutron moderator, the fuel coolant, the neutron-absorbing sections of the control elements, the neutron reflector, the neutron startup source, and any in-core experimental facilities.

Areas of review for the reactor core include an overview of the configuration of the proposed LEU-fueled core and comparison with the existing HEU-fueled reactor (e.g., critical masses and operational masses of uranium-235). Figures should show the locations and geometries of the components in the two cores and lists should provide important design parameters and operational characteristics. The design bases and functional characteristics of any core systems or components for which significant changes are proposed should be reviewed, as well as reasons that the changes are required to accommodate the conversion from HEU fuel. In

this section, the licensee should emphasize safety considerations for the overall core configurations proposed for the LEU-fueled reactor, including the initial LEU core, the equilibrium cores, and any interim cores used to transition between the initial and equilibrium cores.

Technical specifications and their bases in Section 14 of the SAR for all reactor core characteristics and components for the LEU-fueled reactor should be reviewed to compare them with the HEU-fueled reactors and the results of the safety analyses.

4.2.1 Fuel Elements

The areas of review for the fuel elements should include a detailed comparison of the HEU and LEU elements. All changes in enrichment should be discussed, including the effect on reactor operating characteristics and safety. The reviewer should compare the fuel plate and fuel element design bases, mechanical designs, construction materials, fuel element, including cladding, dimensions, metallurgical features, volume ratios of fuel to moderator and fuel to coolant, thermal capabilities, and capabilities of the fuel meat to sustain uranium burnup and to retain fission products. Both standard and special fuel element design, such as control-rod elements, should be reviewed, as applicable.

The reviewer should be familiar with the fuel evaluations previously completed by NRC (NUREG-1281, -1282, and -1313) as well as with other studies and tests of non-power reactor fuel performance referenced by the licensee or available in the general literature on non-power reactor fuel. The reviewer should evaluate the applicability of the reports to the proposed plans for conversion.

All fuel-related technical specifications and their bases in Section 14 of the SAR should be compared, including safety limits. Additional guidance may be found in ANSI/ANS 15.1 and 15.2.

4.2.2 Control Rods

Generally, control rods need not be modified or replaced in connection with an HEU to LEU conversion. However, any significant changes to the rods should be reviewed. Any change in design bases, mechanical design, absorber materials, or configuration in the core should be reviewed.

4.2.3 Neutron Reflector

The neutron reflector might need to be modified or replaced to accommodate the HEU to LEU conversion. The areas of review should include changes in the

design bases and functional characteristics, construction materials, thermal capabilities, effects of radiation, design life, and replacement capability.

4.2.4 Neutron Source and Holder

Any modifications to the startup neutron source or holder required to accommodate the HEU to LEU conversion should be reviewed. These should include mechanical and functional designs, construction materials, thermal and radiation capabilities, and the ability to be retracted or replaced.

4.2.5 In-Core Experimental Facilities

Any proposed changes to in-core experimental facilities to accommodate the HEU to LEU conversion should be reviewed. The design bases, design parameters, and functional characteristics should be compared for the HEU- and LEU-fueled reactors. The review should include the thermal-hydraulic characteristics, effects on core reactivity, and effects on nearby fuel elements and control rods. Potential accidents involving in-core experimental facilities should be addressed in the accident analyses in Section 13. Limitations on the use of experimental facilities should be addressed in Section 10 on experimental programs.

4.2.6 Reactor Materials

In the conversion from HEU to LEU, the fuel will be replaced and other core components may be modified or replaced. The reviewer should verify that the materials from which these components are made have been chosen for the LEU-fueled reactor core environment. Resistance to radiation damage, corrosion, erosion, or vibration should be reviewed. The licensee's safety-related comparison of the capabilities of the proposed materials between the HEU- and LEU-fueled reactors should be reviewed and evaluated.

Acceptance Criteria (Sections 4.2.1–4.2.6)

Acceptance criteria should be based on the following considerations: All proposed changes from the HEU-fueled reactor should be analyzed sufficiently to show that changes in design and operating characteristics would cause no significant decrease in safety margins and would not compromise safe shutdown of the reactor. Any proposed changes in core components other than the fuel must be made only to accommodate conversion to an available LEU fuel that is acceptable to NRC. The following criteria are specific to core components:

4.2.1 Fuel Elements

The design and procurement specifications for the LEU fuel should be consistent with the test fuels previously reviewed, evaluated, and accepted by NRC for non-power reactors. If an unapproved fuel type is proposed, sufficient information should be provided to support evaluations and conclusions comparable to NUREG-1281, -1282, or -1313, as applicable. The physical dimensions, uranium loading, thermal capabilities, uranium burnup capabilities, and safety limits should be consistent with the LEU fuels previously evaluated by NRC and the assumptions and inputs used for the thermal-hydraulic and safety analyses for the LEU-fueled reactor.

4.2.2 Control Rods

The control rods usually need not be modified or replaced for the HEU to LEU conversion. If no change to any control rod is required, the LEU safety analyses in subsequent sections should show that the existing rods can provide excess reactivity control and shutdown reactivities to maintain appropriate technical specification limits and not compromise safe reactor shutdown. If a change is required, the analyses should show that any changes in rod locations or in environmental effects on the rods from changes in reactor operating conditions (e.g., neutron fluence or spectrum changes) would not significantly decrease rod reliability or integrity. If changes in design materials or function of control rods are required to accommodate the reactor conversion, the analyses should show that the LEU-fueled reactor operating design bases, shutdown capability (to be analyzed in subsequent sections), and control rod integrity would not be reduced significantly below those for the existing HEU-fueled reactor.

4.2.3 Neutron Reflector

The reflector components or materials usually need not be modified or replaced in connection with HEU to LEU conversion. If the analyses show that reflector changes are required to accommodate the conversion (e.g., for reasons of changes in neutron flux levels or spectrum), the design bases, mechanical and functional designs, construction materials, thermal capabilities, design life, and replacement provisions should be analyzed to show that the safety-related capability will not be significantly reduced by the changes. If the reflector need not be changed in connection with the reactor conversion, analyses should show that any changes in environmental effects on the reflector from changes in reactor operating characteristics, such as neutron fluence and spectrum, would not significantly decrease reflector integrity or related safety analysis results.

4.2.4 Neutron Source and Holder

The neutron startup source or holder usually need not be modified or replaced in connection with HEU to LEU conversion. If it is, the design and operational characteristics of the LEU components should be comparable to the source in the HEU core. If the source or holder is not changed, the analyses should show that the existing component will perform in the LEU-fueled reactor core without significant decrease in its ability to function and maintain its integrity.

4.2.5 In-Core Experimental Facilities

The in-core experimental facilities or their intended uses might need to be changed to accommodate the LEU fuel characteristics. Analyses of proposed changes in experimental facilities and reactor operating conditions should show that these changes will not increase the risks of thermal or mechanical damage to the facility, its contents, or reactor fuel. The analyses also should show that any change in core reactivity conditions from experimental facilities in the LEU-fueled reactor will not exceed the bases for applicable technical specifications, or the envelope of analyzed and reviewed reactivity events for the reactor.

4.2.6 Reactor Materials

The design and operating specifications for core-related materials should show them to be no less resistant to damage or degradation by the environment (e.g., neutron fluence or spectrum) in the LEU core than in the HEU core. The SAR should show that any materials in the core to be changed in connection with conversion, including the fuel meat and cladding, have been designed so that aging effects (such as corrosion, erosion, radiation, temperature, metallurgical factors, or mutual compatibility) will not significantly decrease the expected life of safety-related components in the LEU core.

Review Procedures (Sections 4.2.1–4.2.6)

For all reactor core components, the emphasis should be to compare the design criteria and bases of the proposed LEU core with those of the existing HEU core. All features related to operations should be reviewed and compared. The reviewer should determine if the proposed LEU core in the converted reactor will continue to meet the design bases and operational objectives established for the HEU-fueled core. The reviewer should confirm that the licensee submitted all information requested in the format and content guide.

Evaluation Findings (Sections 4.2.1–4.2.6)

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

- The proposed conversion of the reactor from HEU fuel to an acceptable LEU fuel has been reviewed. The information and analyses provided were sufficient to ensure that the proposed LEU fuel will comply with design bases and specifications reviewed and evaluated in NUREG (*fill in number as appropriate*). The licensee has shown that the fuel characteristics will meet all applicable criteria for acceptability discussed in that document.
- The licensee has compared the proposed configuration and materials of the reactor using the LEU fuel with that of the existing HEU fuel. The effects of the core environments on the two types of fuel were shown to be similar with no significant decrease in safety margins to the LEU fuel. It was shown that the reliability and resistance to damage (*or other loss of integrity*) of the LEU is comparable to HEU fuel.
- The licensee has proposed to modify or replace the following core components in connection with the HEU to LEU fuel conversion: (*list components*). The licensee provided the technical and safety-related bases of the designs for these components and compared them with the design bases for the similar existing components. The operational performance is reasonably comparable, with no significant decrease in safety margin.

4.3 Reactor Tank and Biological Shield

Areas of Review

The reviewer should examine any proposed modifications to the pool (pool refers to either tank or pool in this section) or the biological shield that are required by the conversion from HEU to LEU fuel. Any structural, operational, or safety issues raised by the modifications should be reviewed. If the existing pool and biological shield are not modified, radiation protection factors should be reviewed for the LEU-fueled reactor operating characteristics and compared with requirements in Section 11 of the conversion SAR. It is not expected that fuel conversion will raise any safety issues related to the pool and shield systems.

Acceptance Criteria

Modifications to pool or biological shield should be acceptable if they will not decrease the designed integrity of the pool or primary coolant envelope against

mechanical failure or degradation related to neutron irradiation, corrosion, and erosion. Additionally, modifications to the biological shield or to reactor operating characteristics will be acceptable (1) if they do not decrease physical integrity of the shield against cracking or other structural malfunctions due design loads (e.g., natural or reactor-induced forces) or neutron irradiation and (2) if the protection of the radiation shielding does not decrease below acceptable requirements as analyzed in Chapter 11 of the staff's safety evaluation report.

Review Procedures

The reviewer should compare design changes against the as-built systems to confirm how proposed changes will affect the structural integrity, resistance to failure or degradation, or radiation protection factors of the pool and biological shield for normal operations or postulated accident conditions.

Evaluation Findings

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

- Any proposed changes to the pool or to the biological shield for the conversion are required to accommodate the conversion.
- Proposed modifications to the pool, shield, or reactor operating conditions will not significantly decrease resistance to failure, or cause new or accelerated degradation or loss of integrity.
- Any new conversion-related pool or shield safety issues are addressed by proposed modifications to the pool and shield systems.
- *(If no changes to the pool or biological shield are proposed)* The information submitted supports the conclusion that the existing systems will provide comparable protection to the public with the LEU fuel.

4.4 Core Support Structure

Areas of Review

The reviewer should evaluate the design features, construction materials, criteria for rigidity and weight support, and capability to maintain fuel and other core components in acceptable positions. The reviewer should also compare the weight and other relevant stresses imposed by the existing HEU core components to those of the proposed LEU core components.

Acceptance Criteria

Acceptance criteria should be based on the following considerations. If no modification is proposed, the LEU core operating conditions (e.g., neutron spectrum and flux levels) or components should not cause loads, stresses, or changes in material response that significantly change the core support function from those currently acceptable for the HEU core components. If there are proposed changes in the core support structures to accommodate the conversion, they should be specified and analyzed to ensure that LEU geometry will be maintained and loads will not be beyond design limits. Proposed changes in the design specifications and tolerances should not be significantly different from those for the current HEU core system. If major modifications or replacement of the core support structure are proposed, in addition to the above structural designs, the design should not decrease resistance to corrosion and erosion or cause a significant increase in long-term induced radioactivity above that for the existing HEU core support system. (This latter issue generally relates to maintenance and ultimate decommissioning and should not be an operational safety issue.)

Review Procedures

The reviewer should compare the design specifications for the proposed modifications to the core support systems with those systems used with the HEU core. The reviewer should compare changes in functional stresses between the two systems and the capabilities to withstand the stresses for the projected duration of the license with LEU core conditions (e.g., neutron spectrum and flux levels). The reviewer should compare construction materials or structure, reactor operating characteristics, and mechanisms of degradation. If the core support structures will not be changed, the reviewer should determine the change in weight between the HEU and LEU fuels and cores and its effect on the support structures and any changes in reactor operating characteristics that could affect the core support systems.

Evaluation Findings

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

- The proposed core support structure for the LEU-fueled reactor will not provide significantly less reliable support, rigidity, and spacings of core components than the existing system for HEU fuel.

- The materials of any proposed modifications are compatible with use in the reactor pool so that degradation should not be significantly faster or different in kind than for the structure in use with the HEU core.

(The reviewer also should discuss the analysis of the expected radioactivity, that will be induced, and its possible effect on maintenance or operations procedures in Section 12 and ultimate decommissioning in Section 15.3 of this appendix.)

4.5 Dynamic Design

Areas of Review (Sections 4.5.1–4.5.4)

In reviewing the dynamic design of the proposed LEU-fueled reactor core, the reviewer should analyze and compare the fuel system, control systems, and reactor core physics parameters of the LEU-fueled reactor with the existing HEU-fueled reactor. These characteristics should be used to confirm that the integrity of the proposed LEU-fuel will not be compromised and that the fuel design limits will not be exceeded during operation or during postulated accident scenarios that are discussed and analyzed in Section 13 of the SAR.

The dynamic design areas of review primarily cover normal reactor operation, which 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. This includes the following topics: reactor core design bases, functional design of reactivity control systems, and thermal-hydraulic characteristics. Postulated accident scenarios should be briefly reviewed, but the details, consequences, and safety significance should be analyzed and evaluated in Section 13 of this SAR.

Changes should be reviewed against existing technical specifications and their bases (e.g., reactivity change rates such as control rod speeds and experiment insertions). Analyses for the LEU-fueled reactor should reflect that changes are appropriate.

The reviewer should cover the nuclear design of the proposed LEU-fueled reactor under all of its allowed operating conditions, which include pulsing for some reactors. This includes deviations from nominal operating conditions under which the specific equipment is expected to operate and function in the safety analyses. Deviations should be described by the mechanistic analyses and may define the limiting conditions of operation. Examples of deviations include coolant flow transients caused by valve realignments, variable time delays in the response of instruments and controls, reactivity changes from experiments or control rod motion, and loss of normal electrical power. Additional guidance may be found in Shibata et al. (1984).

Calculations should demonstrate the principal differences in operating characteristics between the licensed HEU-fueled reactor and the proposed LEU-fueled reactor. The calculations for the HEU-fueled reactor should be compared with measured parameters and characteristics of that reactor to qualify the computational methods and assumptions. The same methods should be used by the licensee to predict the dynamic and static operation of the LEU-fueled reactor because it is expected that the design bases and licensed power levels will not be changed significantly by the conversion. The dynamic responses of the LEU-fueled reactor to anticipated and postulated changes in process variables and to reactivity changes should be reviewed, including inherent feedback mechanisms. The reviewer should compare the HEU core with the proposed initial LEU core, the equilibrium LEU cores, and any interim cores used to transition between the initial and equilibrium cores. If burnable poisons are included in either HEU or LEU fuel, their effects on reactor behavior over the fuel lifetime should be reviewed and compared.

4.5.1 Control Rod Worths and Excess Reactivity

Areas of review include the requirements and safety considerations for excess reactivity of the LEU-fueled reactor and a comparison with the HEU-fueled reactor. The following factors that affect reactivity and its control should be reviewed and compared with the HEU-fueled reactor: core and reactor configurations; differential and integral control rod worths; formation and decay of xenon-135, samarium-149, and other fission products; void coefficient and temperature coefficients of reactivity for fuel and moderator; burnup and buildup of fissile material; depletion of burnable poisons; movable and secured experiments; effects of other core components; and changes in neutron spectra and spatial flux density.

The bases for changes in applicable technical specifications should be developed in this section.

4.5.2 Shutdown Margin

Areas of review include the capability and requirements to ensure safe reactor shutdown based on the proposed excess reactivity and individual scrammable control and safety rod worths analyzed above. The discussion should include the technical and safety bases of a specified shutdown margin. The shutdown margin is an important parameter because it relates to the capability to shut down the reactor safely under any operating conditions. The derivation of this parameter and discussions of factors that affect the shutdown reactivity for the LEU-fueled reactor should be reviewed. This section should develop the bases for shutdown margin technical specifications for the LEU core and compare them with those stated for the HEU-fueled reactor. Guidance on acceptable shutdown margin

derivations is contained in Chapter 14, "Technical Specifications," of the format and content guide, which revises the ANSI/ANS 15.1 definition.

4.5.3 Other Core Physics Parameters

The various core physics parameters that affect reactor behavior should be compared for the HEU- and LEU-fueled reactors. Areas of review should include quantitative discussions and analyses of reactivity feedback coefficients (fuel temperature, moderator temperature, void, and the power defect); the magnitude and effect of prompt temperature coefficients on stability and safety of reactor operation, including pulsing if applicable; and changes in the delayed neutron fraction and the prompt neutron lifetime. The reviewer should compare the neutron flux densities, flux spatial distributions, neutron spectra, and power densities in fuel plates or rods for the HEU- and LEU-fueled reactors.

Because of the relatively low enrichment of uranium-235 in the LEU fuel, a new area for review is the buildup of plutonium-239 and its effect on reactor operating characteristics. Factors to be considered include the fractions of delayed neutrons from uranium-235 and plutonium-239 and the spatial distribution of the plutonium-239 in the core. Comparisons of the plutonium-239 effects on the reactor physics parameters between the HEU and proposed LEU-fueled reactor should be reviewed.

The information provided should compare the physics parameters for the existing HEU core, the initial LEU core, and subsequent LEU cores to the end of life.

4.5.4 Operating Conditions

The dynamic operating conditions for the LEU fuel designs should be compared to those for the HEU fuel. Core thermal power and neutron flux spatial distributions, both radial and axial, power peaking within individual fuel plates or rods, and related temperature distributions for the range of allowable operating conditions should be analyzed. This analysis should include calculations of both the HEU and LEU cores, giving a quantitative description of the main changes in operating characteristics. The values and characteristics calculated as operating conditions (e.g., maximum temperatures, power peaking factors) will be used or referenced in other areas, such as under accident analyses.

The dynamic response of the reactor to anticipated and postulated disturbances in the process variables and to changes in reactivity, including inherent feedback mechanisms and protective actions of the reactivity control elements, should be provided. For example, analyses for maximum allowable power, temperature and reactivity addition conditions should not result in exceeding LEU fuel design temperature limits. Further, any relevant changes in the technical specifications

should be discussed and justified here and referenced to the technical specifications discussion.

Operating parameters that verify proper core configuration and behavior and can be accurately measured, should be verified by measurements in the startup test program. A range of expected values with acceptance criteria and bases should be provided in this discussion and referenced to the startup test program discussion.

Acceptance Criteria (Sections 4.5.1–4.5.4)

Acceptance criteria for the dynamic design should be based on the following considerations: The comparison of the parameters for the existing HEU and proposed LEU cores that are significant factors in the stability and dynamic behavior of the reactor during steady-state or pulsed operation should not significantly change. The temperature coefficients and void coefficient of reactivity of the LEU-fueled reactor should remain sufficiently negative over the allowed ranges of operation that credible reactivity additions are compensated to prevent violation of facility design and safety limits. If the coefficients are not negative, the licensee should justify the values and discuss compensating safety features. If there are no major differences in the magnitudes or functions of individual parameters between the HEU and LEU cores, the reviewer should accept the licensee's use of the same analytical and computational methods to compare the performances of the two reactors. The analyses should confirm that the analytic results for the HEU core have been qualified by reasonable agreement with the experimentally determined parameters of that core and other test cores, such as those tested at the Special Power Excursion Reactor Test (SPERT) project (Nyer et al., 1956). The safety significance of any differences between calculated and measured reactor performances should be discussed and analyzed by the licensee to demonstrate that reactor safety is ensured. The proposed conversion should be acceptable if the analyses of the HEU and LEU cores confirm (a) no significant decrease in safety margin for the full range of operational conditions and (b) no loss of fuel integrity. The proposed startup test program plan should include verification of reactor core parameters as outlined in Section 12 that includes comparison to the predictions in this section. Postulated accident conditions should be evaluated in Section 13 of the SAR.

Review Procedures (Sections 4.5.1–4.5.4)

The reviewer should evaluate all reactor parameters that affect reactor operation, determine fuel integrity, and affect overall reactor safety. The reviewer should evaluate the methods and assumptions used by the licensee to analyze the operations of the HEU-fueled reactor. These analyses should be qualified by comparison with experimentally determined values for the HEU core. The reviewer should evaluate the licensee's analyses of the LEU core and compare

analytical results for the two cores. The reviewer should evaluate significant differences, including their effect on reactor safety, fuel integrity, and safe shutdown of the reactor. The reviewer should look for comparisons with similar reactors and for discussions of the full range of operation for the proposed LEU-fueled reactor. The reactor characteristics should be verified in the startup test discussed in Section 12, and postulated accident conditions are evaluated in Section 13 of the SAR.

Evaluation Findings (Sections 4.5.1–4.5.4)

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

- The licensee has used adequate methods and assumptions to analyze the static and dynamic behavior of the LEU-fueled reactor and has compared the analyses with the existing HEU-fueled reactor.
- The reactor physics parameters that the licensee has derived for the proposed LEU-fueled reactor are consistent with changes in the fuel and core design, and their safety significance has been discussed adequately.
- Safety analysis results for the proposed LEU-fueled reactor are not significantly different from the HEU-fueled reactor, and do not exceed acceptance criteria applicable to the LEU fuel. The differences result from acceptable changes in reactor design factors required by the fuel conversion.
- Inherent reactivity feedback coefficients are sufficiently negative for the proposed LEU-fueled reactor, and their effect on reactor behavior in both steady-state and transient operations has been analyzed, compared with the HEU-fueled reactor, and found acceptable.
- Any predicted changes in the dynamic behavior between the two reactors are explained and compensatory measures are proposed to achieve acceptable safety analysis results for the proposed LEU-fueled reactor.
- The fuel and core characteristics will be verified in the Section 12 startup tests, and are assumed in the Section 13 accident analysis. The conclusions reached about the acceptability of consequences for accidents analyzed in Section 13 for the existing HEU-fueled reactor are valid for the same postulated accidents in the proposed LEU-fueled reactor. The LEU accidents are discussed in detail in Section 13, with a summary of the fuel and core characteristics presented in Section 4.

- Acceptable changes to technical specifications and their bases have been proposed to accommodate the conversion from HEU fuel.

4.6 Functional Design of Reactivity Control Systems

Areas of Review

Although not likely to be affected by the conversion to LEU, the reviewer should evaluate the design features of the reactivity control systems to verify that they include redundancy, diversity, and automatic protective action. The reactivity control system should provide sufficient reactivity worths and response times for the full range of operation and Section 13 postulated accident conditions. The reviewer should examine rod drive speeds, reactivity change rates, and total reactivity worths. Instrumentation for control and position indication, safety wiring and circuitry for isolation and independence of control, and the functions of interlocks and prohibits should also be reviewed. The reviewer should compare the control functions for the HEU- and proposed LEU-fueled reactors. If differences in the operating characteristics of the two reactors require modifications to the reactivity control systems, the design bases and modifications should be reviewed and evaluated. Most conversions from HEU to LEU fuel should not require modifications to the systems, functional capability, or technical specifications for the reactivity control systems. However, any proposed changes in the technical specifications and their bases should be reviewed and verified.

This section should be reviewed in conjunction with Sections 4.2, 4.5, 7, and 13.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: The existing or modified reactivity control systems will be acceptable if no new safety or operational issues are raised by the fuel conversion, and analyses and discussions confirm that the LEU-fueled reactor is comparable to the HEU-fueled reactor in that it can be controlled and shut down with no significant reduction in reliability, response times, reactivity response, and redundancy, for the full range of operations and Section 13 postulated accident conditions. Additional general guidance may be found in ANSI/ANS 15.15 and ANSI/ANS 15.20 (draft). Selected reactivity control system functions and characteristics should be verified as outlined in Section 12 on startup testing.

Review Procedures

The reviewer should compare the reactivity conditions and the transient responses of the LEU- and HEU-fueled reactors. The reviewer should compare the redundancies and diversities of the reactivity control systems and their individual and total reactivity worths. The analyzed kinetic behavior of the HEU and LEU reactors should be reviewed, as should the capability of the reactivity control systems to provide acceptable margins of response. Reactor shutdown should be compared for steady-state and transient operations for the two reactors. Requirements for response of the reactivity control system to postulated accidents should be compared with scenarios in Section 13. Technical specifications and their bases should be compared for the control systems of the HEU- and LEU-fueled reactors.

Evaluation Findings

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

- The modified reactivity control systems or the functional capability of existing control systems in the LEU-fueled reactor will not significantly reduce the redundancy and operational reliability to react to anticipated changes in reactivity. They will provide stable and reproducible control during operation at licensed conditions, and will provide rapid and safe shutdown of the reactor under any operating or accident conditions.
- The proposed technical specifications and their bases for the LEU-fueled reactor give reasonable assurance that the reactivity control systems can satisfy the requirements for shutdown and excess reactivities with no significant change in the safety analyses results from that of the HEU-fueled reactor (i.e., the technical specification limits for shutdown margin and excess reactivity will be satisfied by the LEU reactor).

4.7 Thermal-Hydraulic Characteristics

In the HEU to LEU conversion process, several design parameters associated with the thermal-hydraulic characteristics of the reactor may change. Possibilities include the number and dimensions of coolant channels, number of fuel plates or rods, core dimensions, power density, fuel and cladding temperatures, surface heat flux, thermal conductivity of the fuel, and coolant flow rate during normal steady-state and transient conditions. Accident conditions are considered in Section 13.

Generally, conversion is not expected to require large changes in thermal-hydraulic parameters. The reviewer should consider the bases and the analyses of the proposed changes.

Areas of Review

The reviewer should compare HEU and LEU elements for heat transfer and heat capacity parameters, coolant channel dimensions, and designs to accommodate applicable hydraulic forces.

Areas of review for this section include validation of thermal-hydraulic calculational methods. The validation should compare measurements and calculations for the HEU-fueled reactor with calculations for the proposed LEU-fueled reactor. The analyses should compare interrelated safety parameters, such as fuel temperatures, thermal power densities, and coolant flow characteristics. Limiting parameters, such as departure from nucleate boiling ratio (DNBR), flow instability, and fuel safety limits, should be calculated and discussed. The information should compare the thermal-hydraulic parameters of the existing HEU core with the proposed initial, intermediate, and equilibrium LEU or mixed cores.

The reviewer should evaluate the material for consistency and acceptability of any proposed changes in the technical specifications, and their bases regarding the safety limits (SLs), limiting safety system settings (LSSSs), and limiting conditions for operations (LCOs).

Acceptance Criteria

Acceptance criteria are based on the following considerations:

- Qualified calculational methods that have been shown to agree with measured thermal-hydraulic parameters on the HEU core should be applied to calculate thermal-hydraulic characteristics of the proposed LEU core.
- The LEU calculations and comparisons should show no significant decrease in safety margin results from the HEU core for normal and transient operating conditions and for accident scenarios analyzed in Section 13.
- The discussion in the SAR should demonstrate that any changes in thermal-hydraulic characteristics, such as DNBR, coolant flow stability, and maximum fuel temperatures, that have a significant effect on safety limits and limiting safety systems settings, are a result of the conversion from HEU fuel. The analyses and discussions should show that safe reactor

operation and shutdown would not be compromised by changes in thermal-hydraulic characteristics.

- The bases of any proposed changes in the technical specifications should be developed to ensure that the reactor can be safely operated under normal and transient conditions under all of the allowed combinations of thermal-hydraulic parameters.

Review Procedures

The reviewer should verify that the design bases and safety criteria applicable to the HEU-fueled reactor have been analyzed and compared for the proposed LEU-fueled reactor. This review should also include the proposed changes in the technical specifications to ensure that they are acceptable in regard to SLs, LSSSs, and LCOs.

The reviewer should compare the thermal-hydraulic analysis of the HEU core to that of the LEU core. The reviewer should confirm that (1) the basic assumptions made in the analyses are appropriate, (2) the analysis methods are applicable by showing comparisons between measured values for the HEU core and the analytic results for that core, (3) there is not an unexplained or unwarranted interpolation or extrapolation of the HEU thermal-hydraulic design, (4) the thermal-hydraulic design meets the acceptance criteria, and (5) the conclusions drawn about the LEU-fueled reactor are acceptable. The reviewer should verify that any significant changes in thermal-hydraulic parameters are required by the conversion from HEU to LEU.

The proposed LEU startup test program should be reviewed to confirm the thermal-hydraulic analysis predictions will be verified during LEU core startup. It should be reviewed to ensure that sufficient information has been provided to identify clearly the test objectives, methods of testing, and acceptance criteria.

The proposed technical specifications that relate to the core thermal-hydraulic and the reactor coolant system should be evaluated. This evaluation should cover all SLs and bases that could affect the thermal-hydraulic performance of the core. The LSSSs should be reviewed to ascertain that acceptable margins exist between the values at which reactor trips occur automatically for each parameter and the SLs. The reviewer should confirm that the LSSSs and LCOs, as they relate to the reactor coolant system, do not permit operation with any expected combination of parameters that would not satisfy the acceptance criteria.

Evaluation Findings

This section of the SAR should contain sufficient information to support one of the following type of conclusions, which will be included in the staff's safety evaluation report:

- The licensee's analyses of the thermal-hydraulic characteristics for the proposed conversion from HEU to LEU fuel have been reviewed. Any changes in these characteristics are small and within calculational uncertainties and the changes have been accounted for in the safety analysis and in the technical specifications. These changes have been verified through measurements (on the HEU core), calculations, and comparisons. The safety significance of the changes has been accounted for with no significant decrease in safety margins from the HEU core during normal or transient operating conditions (i.e., design and safety limits are not exceeded). This conclusion is based on the methods and assumptions used to analyze the core thermal-hydraulic performance. The licensee has committed to an initial startup test program to measure and confirm the change in the thermal-hydraulic design characteristics.

or

- The licensee's analyses of the thermal-hydraulic characteristics for the conversion from HEU to LEU fuel have been reviewed. The methods and assumptions in the analyses were verified by comparisons with measurements on the HEU-fueled reactor. The same methods and assumptions were used to analyze the proposed LEU-fueled reactor. Some of the thermal-hydraulic parameters, namely power density in the fuel, reactor coolant system flow around the fuel, and fuel-plate temperatures, would be changed significantly in the converted core for normal operation and some postulated accident scenarios. These changes resulted from factors required to accommodate the conversion, as stated in 10 CFR 50.64. The use of the standardized fuel plates (rods) approved by NRC led to a higher uranium concentration than in the HEU core and, consequently, to a smaller reactor core with higher power density. These changes in core parameters were selected to avoid significantly degrading reactor performance characteristics to maintain the neutronic characteristics of experimental facilities. Even with the changed thermal-hydraulic parameters, the predicted DNBR for the LEU-fueled reactor should have a minimum value of 2.0 for licensed operations, which is conservative and acceptable to the NRC for non-power reactors.

5 REACTOR COOLANT SYSTEMS

Areas of Review

The reviewer should evaluate the design bases and requirements for cooling the licensed HEU- and the proposed LEU-fueled reactors. The conversion from HEU to LEU fuel should not require modification to the cooling systems, but operational conditions may need to be altered. If modifications to the cooling system are required by the conversion, the effects on reactor operation should be reviewed for the two reactors. The reactor coolant system analyses should be reviewed against proposed changes in reactor operational conditions, safety limits, and cooling system capabilities. Also, the reviewer should assess any proposed changes in coolant system technical specifications. The coolant systems should be reviewed in conjunction with Sections 4, 6, 7, and 13.

Acceptance Criteria

Modifications to the coolant system will be acceptable if in the resultant design of the coolant system performance is not decreased significantly in the LEU-fueled reactor for the full range of operation and postulated accident conditions. If the HEU to LEU conversion requires significant changes in the coolant system, the changes in components or operating characteristics would be acceptable if the safety analyses assumptions and technical specification requirements and bases for the LEU fuel would not be exceeded. For example, on forced-flow coolant systems, the transition from forced to natural thermal convection flow should still prevent loss of fuel integrity after the conversion from HEU to LEU fuel.

Review Procedures

The reviewer should compare the functional requirements of the cooling systems for the licensed HEU-fueled reactor and the proposed LEU-fueled reactor. The water quality, flow capability, heat exchange systems, and operating fuel temperatures should be compared. Any required changes in technical specifications and their bases should be reviewed.

Evaluation Findings

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

- The maximum temperature for the LEU-fueled reactor does not reach unacceptable temperature under any potential operating conditions.

- The cooling system for the proposed LEU-fueled reactor is designed to respond to the full range of operation and postulated accident conditions without a significant decrease in safety margin from the licensed HEU-fueled reactor (i.e., the acceptance criteria and design bases for the reactor coolant system are met for the LEU-fueled reactor).
- (*For forced-flow reactors*) The transition from forced convection flow to natural thermal convection flow will not be degraded by the conversion from HEU to LEU fuel, and fuel integrity will not be compromised.

6 ENGINEERED SAFETY FEATURES

Areas of Review

Engineered safety features (ESFs) mitigate the consequences of design-basis accidents, such as loss of coolant, even though the facility is designed so that the occurrence of these events is very unlikely.

The reviewer should evaluate any significant changes in operating characteristics of the reactor required by the HEU to LEU conversion that would affect the likelihood or the consequences of postulated accidents. The reviewer should cover all ESFs, their functional responses, and their capability to mitigate the consequences of the accident scenarios reviewed in Section 13. The conversion from HEU to LEU fuel should not result in changes to reactor operating conditions that would affect accident scenarios sufficiently to require modifications to ESF systems or their operating conditions. However, any proposed changes in the ESF systems or the addition or subtraction of ESF systems should be reviewed to ensure that they perform as assumed for postulated accident conditions.

Acceptance Criteria

Acceptance criteria are based on the following considerations: Any modifications to the reactor systems or operating characteristics required by conversion will be acceptable if the ESF can mitigate accident consequences sufficiently to prevent any consequences to the public, the environment, or to the reactor from significantly exceeding the consequences accepted for the existing HEU-fueled reactor accidents.

Review Procedures

The reviewer should examine the consequences of postulated accidents that are mitigated by ESFs. The consequences from postulated accidents for the proposed LEU-fueled reactor should be compared with the consequences from the licensed HEU-fueled reactor. Any modifications to the ESFs and their functions should be

evaluated and the capability to mitigate the consequences should be compared with those features and functions of the licensed HEU-fueled reactor. If new ESFs are added to the facility, they should be reviewed following Chapter 6 of this standard review plan.

Evaluation Findings

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

- Any changes in the ESFs or their operation will not significantly increase the consequences of postulated accidents in the LEU-fueled reactor from those in the licensed HEU-fueled reactor. In particular, the magnitude of ESF-mitigated consequences to the public for the maximum hypothetical accident or design-basis accident should not be significantly greater for the LEU-fueled reactor than for the existing HEU-fueled reactor.
- There is reasonable assurance that differences in operating characteristics of the LEU-fueled reactor will not lead to postulated accidents whose ESF-mitigated consequences significantly exceed those for similar accidents in the HEU-fueled reactor.

7 INSTRUMENTATION AND CONTROL SYSTEMS

The instrumentation and control (I&C) systems of non-power reactors are designed to provide (1) accurate, reproducible reactor operating conditions; (2) information about reactor operating parameters; (3) the capability to shut down the reactor from any operating condition assumed in the safety analysis; (4) the capability to protect against reactor transients and accidents; and (5) the capability, through interlocks, to limit the likelihood of accidents.

Areas of Review

When a reactor is converted from HEU fuel, the areas of review are proposed modifications to the I&C systems and proposed changes to methods of operation of the existing systems. The reviewer should compare the safety-related operating conditions of the HEU- and LEU-fueled reactors and the functional responses required of the I&C systems to continue to meet the applicable design bases. If the proposed conversion would lead to significant changes in reactor operating characteristics, proposed modifications to the I&C systems should be reviewed. Fuel conversion is not expected to cause significant changes in reactor operating characteristics, so I&C systems probably will not require significant changes.

Acceptance Criteria

Acceptance criteria for the information on I&C systems should be based on the following considerations: Changes in the I&C systems or operational characteristics of these systems should not significantly decrease functional capabilities, including speeds of response, independence of redundant and diverse systems, or shutdown capability for the LEU-fueled reactor, as compared with those of the licensed HEU-fueled reactor. Continued use of existing I&C systems should be acceptable if reactor operating characteristic changes that result from fuel conversion do not decrease the information provided or the reliability of the systems. Specific factors that should be compared are discussed in ANSI/ANS 15.15-1978 and ANSI/ANS 15.20 (draft). The licensee should propose only changes that are required for the conversion, but that do not decrease information, control of reactor performance, or safe shutdown capability.

Review Procedures

The reviewer should systematically evaluate all proposed modifications to I&C systems or changes to operation of existing systems to confirm that sensors and channels are acceptably responsive to all LEU-fueled reactor characteristics. The design bases and methods of calibration for any changed I&C systems should be reviewed. The reviewer should consider all factors of I&C systems discussed in ANSI/ANS 15.15 and 15.20.

Evaluation Findings

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

- The operating characteristics of the proposed LEU-fueled reactor that would be significantly different from those of the existing HEU-fueled reactor and the responses of the I&C components and systems that provide information or control parameters have been analyzed and compared with the comparable parameters of the HEU-fueled reactor. The power level of the two reactors, the maximum fuel plate (rod) temperatures, and the neutron flux per unit power at the nuclear instruments are within the operational range of existing sensors and channels. Any differences should be acceptably addressed by recalibration as part of the LEU-fueled reactor startup program in Section 12, and in subsequent routine surveillance activities.
- The response of the I&C systems to the full range of operation and postulated accidents has been addressed by the licensee, and will cause no

significant decrease in safety capability. Therefore, the I&C systems can provide reactor control without a reduction in safety of reactor operation or shutdown.

8 ELECTRICAL POWER SYSTEMS

HEU to LEU conversion probably should not require modifications to the electrical power systems at non-power reactors because the design bases and functional requirements for the electrical systems are generally independent of detailed operating characteristics of the reactor. Any required changes resulting from the conversion (e.g., reactor coolant system or ESF power needs) should be reviewed for consistency with current design bases and electrical power systems requirements. Additional guidance may be found in Chapter 8 of this standard review plan.

9 AUXILIARY SYSTEMS

9.1 Systems Summary

HEU to LEU conversion probably should not require modifications to the auxiliary systems at non-power reactors, with the possible exception of fuel handling and storage systems. The systems summary should address briefly any proposed changes in an auxiliary system.

9.2 Ventilation System

The areas of review are any proposed changes in design or functional characteristics of the ventilation and air exhaust systems required by the conversion from the use of HEU fuel. The reviewer should compare the design bases of these systems for the existing HEU-fueled reactor and the proposed LEU-fueled reactor. The effects of proposed changes should be discussed.

9.3 Heating and Air Conditioning Systems

The areas of review are any proposed changes in design or functional characteristics of the heating and air conditioning systems required by the conversion from the use of HEU fuel. The reviewer should compare the design bases of these systems for the existing HEU- and the proposed LEU-fueled reactor. The effects of proposed changes should be discussed.

9.4 Fuel Element Handling and Storage

Areas of Review

The reviewer should evaluate all issues related to fuel storage of new and irradiated fuel. These issues comprise physical protection, self-protection, criticality avoidance, radiation shielding, and heat removal.

If fuel handling and storage requirements should be different for the LEU fuel because of changed physical security or self-protection requirements, the licensee should have addressed the issues in the facility security plan and provided a non-safeguards summary in this section of the SAR. (In some cases, security requirements may be relaxed after HEU fuel is removed). The physical security requirements and plan should be reviewed along with this section, as necessary, and under the guidance of Section 12. For example, provisions for maintaining and assessing self-protection of fuel (100 rem/h at 3 ft) should be reviewed in this section if facility changes are required because LEU will not require related safeguard procedures. Capability and authorization to simultaneously possess and store both HEU and LEU fuel, if required, should be reviewed, including proposed changes to license conditions and technical specifications.

Analyses of any changed requirements or capability to maintain subcritical configurations and adequately cooled geometries during storage of new and used fuel should be reviewed.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: Any changes from the conversion of HEU to LEU in fuel storage systems or procedures should not significantly decrease safety margins for radiation shielding, maintaining subcritical configurations, storage conditions (e.g., temperature and water quality) or convection cooling (i.e., acceptance criteria should continue to be satisfied). Fuel handling systems should be appropriately designed for any changes in fuel element parameters, such as size, weight, or fissile material content. Continued interim storage of HEU must conform to applicable regulations, such as 10 CFR 73.6, 73.60, and 73.67, and should be summarized here and discussed in detail in Section 12, "Conduct of Operations," under the discussion of the physical security plan.

Review Procedures

The reviewer should confirm that the licensee has described and analyzed any changes in the facility or in procedures required by the conversion from HEU fuel. These changes might include interim increased storage capability and authorization

to possess both HEU and LEU simultaneously; changes in the facility that affect criticality considerations, radiation shielding, and heat removal from irradiated fuel; changes in procedures or plans related to handling, moving, or shipping HEU or LEU fuel.

Evaluation Findings

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

- The replacement of HEU fuel with LEU fuel would cause no significant long-term change in the inventories of new or spent fuel or in the requirements for possession of special nuclear material after the HEU has been shipped off site.
- The licensee has considered the safety aspects of the possible interim possession and storage of both HEU and LEU cores. Conservative criticality analyses show that the design of the fuel storage facility will limit the effective multiplication factor, if fully flooded, to a maximum of 0.90. Also, protection factors for radiation shielding will limit potential personnel exposures to acceptable values discussed in Section 11.
- The licensee's analyses of temperature increases resulting from decay heat in stored spent fuel show that heat removal and dissipation are acceptably addressed to ensure fuel integrity.
- The licensee's analyses of the storage of the HEU and LEU fuel acceptably consider environmental conditions (e.g., water quality) to ensure fuel integrity will be maintained.
- The licensee has discussed and provided requirements for maintaining the HEU and LEU fuel in accordance with 10 CFR Part 73.

9.5 Other Auxiliary Systems

Auxiliary systems, such as those discussed above, tend to be facility specific. The information from the licensee should include a brief summary of all auxiliary systems at the facility with enough discussion to identify those that could be affected by the conversion from the use of HEU fuel. These auxiliary systems should be described, analyzed, and modified to ensure that there would be no significant decrease in safety margins and no significant increase in radiological risk to the public, the facility staff, or the environment.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary

The experimental program is the principal reason for the existence of most non-power reactors. Because the program might be affected by the conversion of the reactor core from HEU to LEU fuel, it and the associated experimental facilities require review, especially the experimental facilities located within or adjacent to the core. The reviewer should also address procedures for use of these facilities to determine whether they require change to accommodate the conversion.

10.2 Experimental Irradiation Facilities

Areas of Review

HEU to LEU conversion probably should not require modifications to the experimental facilities. However, if the conversion requires changes in core size, core shape, or relocation of control elements, or because of changes in the neutron spectrum of the reactor, some experimental facilities could require modification such as beam ports or pneumatic "rabbit" systems. The areas of review should include designs for component modifications, changes in the radiation characteristics or environments of an experiment facility, changes in reactivity effects and worths of experimental facilities or the associated experiments, and changes in procedures for the experimental facilities in the LEU-fueled reactor. If these topics are analyzed and reviewed in detail in some other section of the safety analysis, the reviewer should refer to those sections.

Possible damage to the reactor itself or possible overexposure of personnel to radiation as a result of malfunction or inadequate design of an experimental facility (or its controls) are appropriate areas of review related to HEU to LEU conversion.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: Any changes in the physical or functional characteristics of experiment facilities, performance of experiments, or reactor operating characteristics required by the conversion from HEU to LEU should be acceptable if they will not (1) lead to unanalyzed damage to the reactor, fuel, or controls; (2) interfere with adequate reactor shutdown; (3) degrade reactor core cooling; or (4) lead to radiation exposure from operations to the facility staff or to the public larger than acceptable limits (e.g., 10 CFR

Part 20), with appropriate "as low as is reasonably achievable" (ALARA) considerations in all cases.

Review Procedures

The reviewer should verify that the licensee has addressed any changes to experimental facilities or controls required by or resulting from the HEU to LEU conversion. The design of any proposed modifications, the materials to be used, the proximity to the core, and the potential for causing damage to fuel or controls should be reviewed. The reviewer should compare whether any proposed changes to the experimental facilities or their operating characteristics, the procedures for use, or reactor operating characteristics will significantly increase risks to the reactor components, the facility staff, or the public.

Evaluation Findings

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

- The licensee has addressed all changes in design and operating conditions to the experiment facilities and their control systems proposed to accompany the core conversion.
- No designs or potential malfunctions of these facilities would lead to significantly increased risks to the reactor, the public, or the facility staff.
- No significant new experimental programs or changes in procedures for performing experiments are proposed, so there is continuing reasonable assurance that the proposed experimental program would not lead to undue radiological risk to the public, the facility staff or users, or the environment.

11 RADIATION PROTECTION PROGRAMS AND WASTE MANAGEMENT

The information in this section is organized in two parts. The first part relates to the facility radiation protection program. The second part relates to radioactive wastes, including personnel and environmental radiation protection.

11.1 Radiation Protection Program

Areas of Review

No significant changes in the radiation protection program are expected to result from changes in reactor systems, operating conditions, or procedures that were required by the conversion from HEU to LEU fuel.

The reviewer should evaluate

- the potential sources of radiation at a research reactor (including if appropriate criticality monitors to comply with 10 CFR 70.24 requirements for fuel storage)
- the provisions made by the licensee to detect and measure all potential reactor-related sources
- the provisions made to assess potential exposures and doses to the facility staff and the public
- measures being taken to limit doses to within applicable regulations
- commitment to an ALARA program for potential exposures
- ongoing assessment of the effectiveness of the radiation protection programs

If the conversion of the reactor from HEU to LEU fuel requires modifications to the facility or changes to operating conditions, limits, or procedures, the reviewer should confirm that the licensee has addressed the effects of these changes on the radiation protection program at the facility.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: Any proposed changes to the radiation protection program or its implementation should be related to the conversion of the reactor from HEU to LEU fuel. Any changes to the radiation protection program should not significantly decrease the level of protection provided to the public, the facility staff, or the environment. The licensee's analytical methods and assumptions should be applicable to non-power reactors, commensurate with the types and magnitudes of radiological risks at the facility, and consistent with the existing program for the HEU-fueled reactor. The program shall continue to comply with the requirements of 10 CFR Part 20 and the facility-specific ALARA program.

Review Procedures

The reviewer should examine any proposed changes to the facility's radiation protection program to ascertain if the changes are required by the differences in operational characteristics, systems, or procedures between the HEU- and LEU-fueled reactors. The reviewer should evaluate the proposed program changes and the effect on projected exposures to the facility staff, the public, and the environment. The reviewer should evaluate the staffing, program documentation and written procedures, sources of radiation, radiation detection instruments and effluent monitors, personnel dosimetry program, environmental monitoring, and record-keeping procedures. The licensee's methods for analyzing projected doses and the results should be compared with accepted methodology for non-power reactors, such as those found in the Regulatory Guides 1.25, and 1.109, NUREG-0851, and ANSI/ANS 15.7 and 15.11.

Evaluation Findings

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

- All proposed changes to the radiation protection program, components, and conditions were required by the conversion from HEU to LEU fuel.
- The proposed changes will not significantly decrease the effectiveness of the radiation protection program (i.e., the requirements of 10 CFR Part 20 will not be exceeded)
- The changes will not significantly increase the projected radiation exposures and doses to the facility staff, the environment, or the public.
- The methods and assumptions used by the licensee to assess potential doses from operation of the proposed LEU-fueled reactor are technically valid, physically reasonable, and conservative.

11.2 Radioactive Waste Management

Areas of Review

Other than a different form of encapsulated uranium fuel from that of the HEU, changes in facilities or procedures are not expected to cause significant changes in radioactive wastes resulting from the conversion to LEU fuel.

The areas of review relate to the formation, control, release, and disposition of radioactive waste materials generated by reactor operations. These wastes may be liquid, solid, or gaseous (airborne), and all three forms should be addressed. At non-power reactors, some of the radioactive wastes come from routine reactor operations and some come from the experimental programs. In general, the reviewer should examine the radiation sources and source terms for these wastes and methods for identification, quantification, control, and disposition. Potential or routine radiation exposure of personnel and the environment to these radioactive wastes also are areas of review.

In addition to the control and disposition of these wastes as a result of the full range of operations, if there is the possibility of an accident that could lead to an uncontrolled release of radioactive materials to the environment, such an event should be analyzed in the accident analyses in Section 13.

If the conversion of the reactor from HEU to LEU fuel requires modifications to the facility or changes to operational conditions, limits, or procedures, the licensee must address these changes and analyze their effects on the production and disposition of radioactive wastes at the facility.

Argon-41 is a radioactive gas produced and treated as radiological waste during routine operations at most non-power reactors. Because it could be the only significant radionuclide released to the unrestricted environment at many non-power reactors, any changes to its potential radiological effect should be analyzed.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: The licensee should show that any changes in radioactive wastes that result from conversion from HEU to LEU fuel will not result in a significant decrease in the control of these wastes. The licensee should also show that potential radiation exposures and doses to the facility staff and the public will not be significantly increased. Potential releases to the unrestricted environment and exposures of personnel shall remain below applicable limits of 10 CFR Part 20, including the facility ALARA program guidelines.

Review Procedures

The reviewer should look for planned changes in the facility components, systems, operating conditions, or procedures to be made in connection with HEU to LEU conversion. The reviewer should examine the licensee's analyses of (1) any changes in the production or control of radioactive wastes and (2) the radiological exposure on the facility staff, the environment, and the public caused by the conversion-related changes in the quantities or types of radioactive wastes. The

reviewer should verify the licensee's ALARA considerations and proposed methods for compliance with other applicable regulations, such as 10 CFR Part 71.

Evaluation Findings

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

- The licensee's proposed modifications to the facility or changes in reactor operating conditions or procedures are required by the HEU to LEU conversion.
- The licensee has analyzed any changes in the production or control of radioactive wastes and demonstrated that any radiological effects from these changes are not significantly larger than for the HEU-fueled reactor.
- The evaluation confirms that the licensee's ALARA considerations are consistent with applicable program requirements and guidelines and that potential radiation exposures to facility personnel, the public, and the environment from these wastes remain consistent with all applicable regulations.

12 CONDUCT OF OPERATIONS

Areas of Review

The reviewer should assess the planning and implementation of facility staff activities that are subject to regulation and relate to the conformance, quality, and effectiveness of operations. Although major changes are not expected, some of the following areas could be affected by the conversion of the reactor from the HEU to LEU fuel:

- Organization and staff qualifications—These factors should not change as a result of fuel conversion.
- Procedures—Changes may be needed to specify new core configurations or loading techniques. Also other system changes for the conversion may require procedure changes.
- Operator training and requalification—If conversion of a reactor results in different systems, different operating characteristics and procedures, different administrative or operating limits, or different technical

specifications, associated retraining of licensed operators should be considered.

- Emergency plan—If the conversion changes facility characteristics or systems for emergency response, associated changes in the emergency plan and implementing procedures should be considered.
- Physical security plan—If the conversion results in changed requirements for physical control of the special nuclear material, a modification of the physical security plan and related procedures should be considered. Both interim (i.e., while the HEU fuel is still at the facility) and permanent changes should be evaluated.
- Reactor reload and startup plan—The conversion of a reactor from HEU to LEU fuel could be implemented all at once or LEU fuel elements could replace HEU fuel elements a few at a time according to a detailed plan and schedule. In either case, to ensure that the licensee has sufficient information to operate the reactor safely, a reload and startup plan should be included with the licensee's submittal. The plan should provide for testing any newly installed equipment; a proposed fuel loading procedure and schedule; radiation surveys; a systematic set of subcritical measurements in the approach to critical with the new fuel; experiments and measurements that compare predicted and calculated reactor parameters; and verification of compliance with license conditions, including technical specifications, of the LEU-fueled reactor.

Acceptance Criteria

Acceptance criteria should be based on the following considerations:

- Organization and staff qualifications—No direct relationship is expected between the organization and staff qualifications and the enrichment of the fuel in the reactor. Therefore, the licensee should not propose any changes in these functions as a result of fuel conversion. Any proposed changes should be submitted to NRC as a license amendment request.
- Procedures—Procedural changes should account for LEU-fueled core and fuel configuration may include such issues as location of fuel elements and control, conduct of critical experiments, storage and shipment of fuel elements, as well as procedural changes to account for needed technical specifications changes. Specific changes in shipping requirements and procedures should be considered by the licensee (e.g., type B material, and associated quality assurance program and cask user registration

requirements). Also, if changes to other systems are required, appropriate changes to facility procedures should be incorporated.

- **Operator training and requalification**—The retraining of licensed operators could be a one-time supplement to the existing operator requalification plan. This supplement would be acceptable if it specifies the procedures, techniques, and on-the-job training that will be used to instruct licensed operators on all changes made to the reactor, its operating characteristics, written procedures, and license conditions related to the conversion. Changes to the operator training program should be acceptable if the new reactor operating characteristics are consistent with the existing program, 10 CFR 50.54, 10 CFR Part 55, NUREG-1478, and ANSI/ANS 15.4.
- **Emergency plan**—Changes in operating characteristics or procedures resulting from fuel conversion generally will not significantly affect emergency events or emergency planning. However, if significant reactor changes are required by the conversion, they could affect emergency planning. In such cases, the licensee should submit a revised emergency plan, whose content is consistent with 10 CFR 50.54, Regulatory Guide 2.6, ANSI/ANS 15.16, and NUREG-0849.
- **Physical security plan**—Removal of the HEU fuel from the facility and replacement with LEU fuel could change the category for possession of special nuclear material, as defined in 10 CFR 73.2, 73.6, and 73.67. Changes in the physical security plan and its implementing procedures should be acceptable if the requirements of the regulations are met for irradiated and unirradiated fuel at the facility and in transit. If it is anticipated that HEU fuel radiation levels will decrease so that "self-protection" would be lost, the licensee must propose actions to preclude violations of license and applicable regulations. Guidance is available in NUREG/CR-4203.

If the logistics of conversion require both the HEU fuel and LEU fuel to be at the facility simultaneously, the licensee must propose appropriate license conditions to allow that.

Additional guidance may be found in Chapter 12 of this document.

- **Reactor reload and startup plan**—The plan would be acceptable if it clearly provides for the following: (1) the licensee receives, handles and loads the LEU fuel as though the reactor is being fueled for the first time (e.g. monitoring, examinations and cleaning); (2) the critical mass (number of fuel elements) is approximately known and will be exactly determined by a systematic approach to critical; (3) neutron detectors of high sensitivity and

reliability supplement the operational instrumentation during subcritical neutron multiplication measurements; (4) measurements are planned to measure operational reactor parameters, such as shutdown reactivity (to confirm shutdown margin), reactivity feedback coefficients, differential and integral control rod worths, power level monitors, scram and interlock functions, fuel heat removal, and related thermal-hydraulic parameters; (5) measured and predicted LEU parameters should be compared and the results of the comparisons should be evaluated against preestablished acceptance criteria; (6) the control rods will be calibrated and excess reactivity will be loaded systematically, resulting in accurate values; (7) thermal power of the reactor will be calibrated acceptably and accurately to ensure compliance with the licensed power level limits and any other license conditions, such as pulse characteristics; (8) area and effluent radiation surveys will be performed to confirm status; and (9) all new, recalibrated, refurbished, or modified instruments and components will be tested before routine operations begin.

Review Procedures

The reviewer should verify the following:

- Organization and staff qualifications—The reviewer should verify that no changes to organization and staff qualifications would be required by the fuel conversion and none are proposed.
- Procedures—The reviewer should verify that applicable procedures were appropriately considered and changed to accommodate the new LEU facility.
- Operator training and requalification—The reviewer should verify that the licensee plans to train all operators in the significant changes in design, operational characteristics, and technical specifications for the LEU facility.
- Emergency plan—The reviewer should determine if changes proposed by the licensee in the emergency plan for the LEU facility are consistent with the changes required for conversion and with the provisions of NUREG-0849, Regulatory Guide 2.6, and ANSI/ANS 15.16.
- Physical security plan—Review procedures should verify if the changes in the security plan are consistent with the change to LEU fuel and the requirements of 10 CFR Part 73.

- **Reactor reload and startup plan**—Comparisons should be made with the initial HEU-fueled reactor load and startup reports, technical specifications, other license conditions, reload and startup plans, and final reports from other similar reactor facilities that have been converted from HEU to LEU fuel. The reviewer should verify that key facility parameters and conditions which are calculated and used in the licensee's safety analysis are to be measured and that acceptance criteria are specified for the tests.

Evaluation Findings

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

- Sufficient information and discussion have been provided in each of the topics of this section and the submitted plans comply with all applicable regulations and guides.
- Implementation of the proposed reactor reload and startup plan will provide reasonable assurance that the LEU-fueled reactor is operating as described and analyzed in the SAR.
- Any changes in the facility or its operating procedures resulting from the fuel conversion and all proposed changes in the conduct of operations will afford acceptable security and protection of the health and safety of the public.

13 ACCIDENT ANALYSIS

Accident evaluations are an important aspect of the conversion from HEU to LEU fuel. In this section, the licensee should compare the consequences of all accident scenarios postulated and analyzed for the existing HEU-fueled reactor with similar accidents for the proposed LEU-fueled reactor. Additionally, the discussions should analyze any new accidents not previously evaluated that could be introduced by the conversion.

The information should summarize the postulated initiating event and the scenario of each accident in the HEU-fueled reactor, showing how the course of the accident and the consequences depend on the physical parameters and geometry of the reactor core, operating characteristics, and license and technical specification requirements for the reactor. The discussions should compare the parameters of the HEU-fueled reactor that have a significant effect on the progression and the consequences of the accident with the comparable parameters of the proposed LEU-fueled reactor. The comparison of the consequences should show that the

design of the proposed LEU-fueled reactor is sufficient to provide reasonable assurance that conclusions previously reached in the SARs and the staff's safety evaluation reports about the consequences of the postulated accidents remain valid.

The reviewer, through comparison of design features, should conclude that if the fuel conversion introduces accidents not possible with the HEU-fueled reactor, compensating measures and conditions have been proposed or are present to prevent the potential consequences of new accidents from exceeding the acceptance criteria for the existing HEU maximum hypothetical or design-basis accidents.

New or revised analytical procedures may be provided if the information demonstrates that the conversion from HEU fuel could introduce new, unanalyzed accidents, could cause significantly different consequences from a previously postulated accident, or the previously used analytical methods are not appropriate to compare the accidents with acceptable validity. If significantly different analytical methods are used for the LEU-fueled reactor, they may have to be validated by reanalyzing some HEU-fueled reactor accidents. In any event, the analyses and evaluations should demonstrate that the LEU-fueled reactor is designed to withstand all postulated HEU-fueled reactor accidents and any new accidents so that the consequences would not exceed the acceptance criteria used for licensing the HEU-fueled reactor.

Guidance for analyzing potential accidents at non-power reactors suggests grouping them into categories with similar initiating events or similar consequences and performing detailed analyses on the accident in each category with the worst consequences. The guidance information that follows is organized according to that approach and addresses major types or categories of potential accidents. Additional guidance for specific accidents and conditions may be found in Bullock (1962), Hunt (1959), Gulf General Atomics report GA-6596 (1964), ORNL report TM-627 (1964), Thompson (1964), NUREG/CR-2079, NUREG/CR-2198, NUREG/CR-2387, and Regulatory Guide 1.25.

Accidents tend to be facility specific, so the information that follows shows the scope and method acceptable for accident evaluations for conversions. In general, initiating events for potential accidents at most non-power reactors lead to scenarios that fall within the following three generic categories: (1) a maximum hypothetical accident, (2) a reactivity excursion or transient, and (3) a reduction in fuel cooling. The examples that follow address these categories.

13.1 Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) is a term used for non-power reactors that are undergoing major licensing reviews, such as initial licensing, license renewal, or fuel conversion. The MHA is facility specific. It is the postulated accident with potential radiological consequences to the public health and safety that are greater than for any other postulated event at the facility. In contrast to other postulated accidents, the MHA need not be entirely credible, so the probability and the mechanistic details of the initiating event need not be analyzed. The evolving accident scenario, however, should be based on physical principles and assumptions amenable to logical analysis. In order to envelope the consequences of the worst credible accident and develop and illustrate the analytical approach, only the consequences of the MHA when the scenario has reached a stable condition need be analyzed.

In general, the most serious radiological risk in the unrestricted area from a non-power reactor accident would result from fission products released from the fuel or from a fueled experiment. Therefore, the MHA scenario is expected to consider this release.

Areas of Review

The reviewer should examine the information which compared the scenario for the MHA of the proposed LEU-fueled reactor with the scenario for the existing HEU-fueled reactor. If the two are not significantly different, the reviewer may note which fuel and core parameters (including fission product inventory) and which reactor operating characteristics and procedures have major effects on the course of the accident and the consequences; then the reviewer should compare these important parameters for the HEU- and LEU-fueled reactors and note which parameter changes could most affect the consequences. The analyses should show the consequences of the MHA for the two reactors. The associated discussion should indicate any changes required by fuel conversion to protect against the transport of fission products to the unrestricted environment. This discussion should include any mitigation of consequences by ESFs for the two reactors.

If the MHA scenarios for the HEU and LEU are shown to be significantly different, the bases for the changes should be given, and the fuel parameters (including fission product inventory), reactor operating characteristics, or procedures that are the principal contributors to the changed scenario should be identified. The discussion should show how these changes were required by the conversion. The course of the LEU MHA should be analyzed to identify the significant steps, component malfunctions or failures, and protective or mitigative actions on the transport of radioactive material from the core to the unrestricted area. If the analytical methods were not used in the previous analyses of the HEU

MHA, the bases for their use should be validated, for example, by referring to previous validations of the methods or by calculating some known HEU events or conditions. This validation should demonstrate that the new methods accurately model reactor accident conditions.

For either case, the consequences to the health and safety of the public, the reactor staff, and the environment should be compared for the two reactors and with the acceptance criteria for postulated MHAs at non-power reactors.

Acceptance Criteria

Acceptance criteria for the information on the MHA at non-power reactors should be based on the following considerations:

- The principal objectives of the MHA analysis are to compare the consequences from the proposed LEU-fueled reactor with the existing HEU-fueled reactor and to relate any changes in consequences with reactor parameters or procedures that were changed to accommodate the conversion. The performance of the reactor should not be significantly decreased to accommodate the conversion.
- The initiating event for the MHA need not be credible nor evaluated, but the progression of the scenario should be based on sound physical principles and assumptions amenable to logical analyses.
- The proposed LEU-fueled reactor and any ESFs should be designed to prevent the projected MHA consequences from significantly exceeding the consequences resulting from the existing HEU-fueled reactor.
- The radiological consequences of the LEU MHA should be shown to exceed and envelope the consequences of all credible accidents postulated and analyzed for that reactor.
- Projected dose rates and accumulative doses to members of the public in the unrestricted area should not be significantly higher for the proposed LEU-fueled reactor than for the existing HEU-fueled reactor and shall be within acceptable limits.

Additional guidance on assumptions for MHA analyses may be found in U.S. Atomic Energy Commission report TID-14844.

Review Procedures

The reviewer should determine if the scenarios and consequences of the MHA for the HEU-fueled reactor and the proposed LEU-fueled reactor are significantly different. If not different, the reviewer should follow the analysis by the licensee, with quantitative checks at key steps as the accident scenario evolves. The reviewer should compare the projected consequences of the MHA for the proposed LEU-fueled reactor with the consequences of all other postulated accidents to verify that those of the MHA exceed and envelope all others.

If the scenarios differ significantly, the reviewer should verify that the evolution of the accident scenario is analyzed by applicable methods and should compare the projected consequences with acceptance criteria. The reviewer should verify that the projected consequences continue to comply with the acceptance criteria for MHAs at the specific non-power reactor facility.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the staff's safety evaluation report. The reviewer should choose a conclusion from 1A, 1B, or 1C and conclusion 2:

- (1A) The licensee has discussed the changes in fuel parameters, operating characteristics, and procedures that result from conversion of the reactor from HEU to LEU fuel. The effects of these changes on the assumptions and analyses of the MHA postulated for the existing HEU-fueled reactor are small enough so that the same MHA scenario is applicable for the proposed LEU-fueled reactor.
- (1B) The information and analyses for the MHAs of the two reactors were compared. The steps and mechanisms for transport of fission products from fuel to the unrestricted area are not significantly different, and any change in the fission product inventory or other radioactive material between the two reactors has been included in the analyses.
- (1C) Changes to the reactor fuel, core, operating conditions and procedures required by the fuel conversion were shown to be significant. The effects on the previously postulated MHA for the HEU-fueled reactor were shown to be large enough that a different MHA scenario was required for the proposed LEU-fueled reactor. The licensee adopted a different analytical procedure that was demonstrated to be applicable by verification with the HEU MHA (or with other acceptable analytical techniques and references). The MHA for the LEU-fueled reactor was analyzed and appropriate, and

acceptable assumptions were used. The transport of fission products and other radioactive material from the core to the unrestricted area was analyzed, and potential dose rates and accumulative doses to the staff, members of the public, and the environment were calculated. All changes in the reactor parameters and ESFs were included in the analyses.

- (2) The radiological consequences to members of the public in the unrestricted area and to the environment resulting from the postulated MHA for the proposed LEU-fueled reactor would not significantly exceed the consequences previously calculated and found acceptable for the HEU-fueled reactor. Furthermore, the maximum projected dose rates and accumulative doses for the MHA exceed and envelope the radiological consequences of all other accidents postulated for this LEU-fueled reactor and are within the guidelines of that acceptable for the HEU MHA.

13.2 Rapid Addition of Reactivity Accident

At non-power reactors there may be the potential for the inadvertent or accidental addition of significant amounts of excess reactivity in an uncontrolled way.

Among the reasons for this excess reactivity are the need for frequent power changes (including startups and shutdowns), experimental programs, and the need for versatility to accomplish these readily. The design and operation of a non-power reactor requires systems, administrative controls, and procedures to limit the likelihood and magnitude of accidental reactivity additions.

Areas of Review

The reviewer should consider the initiating events and scenarios postulated for accidental reactivity addition for the existing HEU-fueled reactor and the bases and criteria for selecting the ones with the most serious consequences for detailed analysis. The reviewer should evaluate comparisons of these considerations with similar ones for the proposed LEU-fueled reactor. The comparisons should show the parameters of the reactor fuel, core, operating characteristics, and procedures that affect the type and magnitude of consequences for postulated reactivity accidents in both HEU- and LEU-fueled reactors. Accidents can be grouped as either rapid addition of reactivity or slow (ramp) addition. The rapid reactivity addition includes rapid control or safety rod withdrawal or ejection, dropping a fuel element into a core vacancy, failure or dislodging of a component in the core, including an unsecured experiment, or a sudden change in coolant temperature or flow. A ramp reactivity addition could include any of the above on a slow time scale, such as continuous withdrawal of a control or safety rod by malfunction of its normal drive system, melting or flowing of an experiment from its intended location and configuration, or a slower change in coolant conditions that allows

temperature and reactivity changes in the fuel or moderator. All credible potential accidents should be considered when planning the reactor conversion.

If the comparisons by the licensee of the existing HEU- and proposed LEU-fueled reactors show sufficient similarity between the potential reactivity accidents and the consequences for the two fuels, a comparison of the scenario analyses should be performed, with quantitative checks at key points. The checks should examine the effects on the course of the accident evolution caused by the reactor design or operational parameters changed to accommodate the fuel conversion.

If the comparisons of the existing HEU- and proposed LEU-fueled reactors show significant differences in the types or consequences of potential reactivity accidents, new independent analyses for the proposed LEU-fueled reactor may be required. If new or revised analytical methods or assumptions are employed, the licensee should validate them with an analysis similar to that used for HEU accident scenarios.

In all cases, the analyses should include the initial reactor conditions; the postulated initiating event; the evolution of the accident; assumed operational condition (including failures, malfunctions, or normal operational function) of any barriers, protective systems, or actions; and the effect of any ESFs that stabilize the reactor condition. The consequences during and after the postulated accidents should be compared for the existing HEU- and proposed LEU-fueled reactors.

Acceptance Criteria

Acceptance criteria for the consequences of proposed reactivity accidents should be based on the following considerations:

- The initiating events and scenarios for reactivity accidents should be consistent with those assumed for the HEU reactor, and should be based on physical principles and assumptions that are amenable to logical analyses.
- Any significant differences in the initiating events or evolution of reactivity accidents between the existing HEU- and proposed LEU-fueled reactor should result from changes in reactor design parameters or operating conditions required to accommodate the conversion of the reactor from the use of HEU fuel.
- The potential consequences of reactivity addition accidents for the proposed LEU-fueled reactor should be consistent with the acceptance criteria for the HEU-fueled reactor.

Review Procedures

The reviewer should confirm that the submitted information systematically considers all potential reactivity accidents analyzed for the existing HEU-fueled reactor and identifies the initiating event, and describes the function of key features and components that significantly affect the course of the accident and the consequences. The reviewer also should confirm that information compares the operating conditions and the changes in design for the proposed LEU-fueled reactor and identifies any factors that could change the evolution or consequences from outcomes for the existing HEU-fueled reactor. The reviewer should watch for consequences that could possibly melt fuel, challenge the integrity of the cladding, or degrade the system core cooling characteristics.

The descriptions of the consequences for potential reactivity accidents in the proposed LEU-fueled reactor should be compared with the acceptance criteria. The reviewer should compare the licensee-provided information with applicable accident analyses and events (e.g., Woodruff, 1984).

Evaluation Findings

This section of the SAR should contain sufficient information to support one of the three conclusions below, which should will included in the safety staff's evaluation report:

- The licensee has demonstrated that the conversion from HEU to LEU fuel does not introduce the potential of a new reactivity addition accident not previously analyzed for the HEU-fueled reactor or significantly increase the consequences beyond those for an existing HEU-fueled reactor accident. In this assessment, the licensee presented the basic neutronic, thermal-hydraulic, and physical similarity between the HEU and LEU cores, and an analysis showing that the conclusions in the latest version of the HEU SAR regarding the consequences of the maximum credible reactivity addition accident are still applicable to the proposed LEU-fueled reactor. Therefore, risk to the health and safety of reactor staff or the public does not increase above that previously found acceptable for the HEU core from the maximum reactivity addition accident.

or

- The licensee has demonstrated that, although the conversion from HEU to LEU fuel does introduce the potential for a new reactivity addition accident or significantly changes the results of an existing HEU scenario, the results from the new safety analysis give reasonable assurance that the proposed LEU-fueled reactor is designed to avoid unacceptable fuel damage from

the postulated reactivity accident. The postulated maximum reactivity addition accident scenario, the assumptions, calculational techniques, and consequences are found to be applicable to the LEU-fueled reactor and acceptable. The calculations predict that the maximum fuel and cladding temperatures will be below the threshold of any damage, no fission products are predicted to escape, and no radiological assessment is required other than a statement that the radiological consequences are enveloped by the licensee's MHA. The calculations are sufficiently conservative in the assumptions of initial operating conditions and the analytical models to provide reasonable assurance that the integrity of the fuel will be maintained. Thus, the conversion from HEU to LEU does not present any significant risk to the health and safety of the public as a result of the consequences of the maximum credible reactivity addition accident in the LEU core.

or

- The licensee has demonstrated that, although the conversion from HEU to LEU fuel does introduce the potential for a new reactivity addition accident or significantly changes the results of an existing HEU scenario, the results from the new safety analysis give reasonable assurance that the proposed LEU-fueled reactor is designed to avoid unacceptable fuel damage from the postulated reactivity accident. The postulated maximum reactivity addition accident scenario, the assumptions, calculational techniques, and consequences are found to be applicable to the LEU-fueled reactor and acceptable. The calculations predict the maximum fuel and cladding temperatures and correlate these predictions to conservatively established thresholds of fuel and cladding damage. On the basis of the calculations, fission product releases and radiation exposures for the event are established to predict the radiological consequences. The calculations are sufficiently conservative in the assumptions of initial operating conditions and the analytical models to provide reasonable assurance that the radiological consequences are within the acceptance criteria for the HEU-fueled reactor. Thus, the conversion from HEU to LEU does not present any significant risk to the health and safety of the public as a result of the consequences of the maximum credible reactivity addition accident in the LEU core.

13.3 Reduction in Cooling Accidents

In non-power reactors moderated and cooled by liquids, principally water, the two potential type of reduction in cooling accidents for the HEU/LEU safety analysis are the loss-of-coolant accident (LOCA) and the loss-of-flow accident (LOFA).

13.3.1 Loss-of-Coolant Accident

Areas of Review

LOCAs are postulated accidents that result from the loss of reactor coolant faster than the makeup system can replenish it. This accident could occur from structural failure of the reactor tank or tube rupture in the experimental facilities. Loss of significant quantities of reactor coolant would reduce the rate of heat removal from the reactor core, unless the coolant is replenished in a timely manner [e.g., emergency core cooling system (ECCS)]. Generally, the enveloping analysis assumes an instantaneous loss of all coolant while the reactor is operating at full power with no ECCS to prevent fuel damage. Other non-power reactors, usually those with higher power levels, have assumed more mechanistic initiating events and scenarios. For example, loss of coolant from the maximum size penetration to the reactor pool is calculated, and the associated increase in cladding and fuel temperature and potential fuel damage and radiological consequences are also calculated. Some of these larger reactors assume ECCS function and some do not.

In the conversion from HEU to LEU fuel, several thermal-hydraulic characteristics of the core may change. The areas of review should include the number and dimensions of coolant channels, number and dimensions of fuel plates or rods, core dimensions, fission product decay power density, fuel and cladding temperatures, surface heat flux, thermal conductivity of the fuel, and significant residual heat removal mechanisms, including ECCS, if applicable. The event initiating a LOCA should not be expected to change with conversion, but the time-dependent temperature evolution during a LOCA may be different in the proposed LEU core compared with the existing HEU core.

The areas of review should include the postulated initiating event, the assumed initial reactor operating conditions, the reactor history as a basis for determining the decay heat (fission product) power level as a function of time, the duration and mechanisms of coolant loss, disposition of the coolant, initiation or presence of heat removal mechanisms from the fuel, heat sinks, and thermal-hydraulic analyses that give the maximum fuel or cladding temperatures as a function of time until the accident conditions are stabilized. If the licensee can show that significant reactor parameters for the LOCA are sufficiently similar for the HEU- and proposed LEU-fueled reactors, a comparison of the scenario evolutions, with check calculations at key points may be used.

The information provided should also compare the radiation fields and potential radiation exposures for facility staff and the public resulting from the loss of coolant that provided shielding around the reactor cores.

Acceptance Criteria

Acceptance Criteria for the scenario and consequences of the postulated LOCA should be based on the following considerations: The initiating event and scenarios for the existing HEU- and the proposed LEU-fueled reactor should be similar, unless the conversion requires significant changes in facility design or operating characteristics. If the scenarios are similar, similar assumptions and methods of analysis should be used. The analysis should show that the LEU-fueled reactor systems and fuel are designed to ensure that fuel integrity is not lost and that potential risk of damage to fuel or systems is not significantly increased by the conversion.

If changes in reactor systems, fuel, or operating characteristics required by the conversion are significant, a new scenario may be required as well as a different analytical method. The licensee should validate any significant differences in these factors by comparison with the HEU-fueled reactor and any applicable analyses. If different analyses are required, they should show that the LEU-fueled reactor fuel and system designs are sufficient to show that any consequences of the LOCA are within acceptance criteria for the HEU-fueled reactor. The information should include reference to applicable technical specifications that ensure operability and availability of any required ECCS for which credit is taken in the analyses.

Review Procedures

The reviewer should compare the information provided to determine if there are significant differences in the postulated LOCA events for the HEU- and LEU-fueled reactors. The reviewer should find any differences and trace the scenarios and analyses from initiation of the event until conditions are stabilized. The information should justify changes in reactor systems or operating characteristics designed to mitigate consequences to fuel, the reactor, and the public.

The reviewer should compare the analyzed consequences for the HEU- and LEU-fueled reactors and should compare the LEU consequences with the acceptance criteria.

Evaluation Findings

This section of the SAR should contain sufficient information to support one of the following type of conclusions, which will be included in the staff's safety evaluation report:

- The licensee assumed the same loss-of-coolant initiating event and scenario for the proposed LEU-fueled reactor as previously postulated for the licensed HEU core (e.g., the instantaneous, total loss of coolant). The

methods and assumptions are applicable and acceptable. The safety analysis demonstrates that the fuel and cladding temperatures are within the acceptance criteria for the LEU fuel, fuel integrity will acceptably be ensured, and there will be no significant release of fission products. Further, there is no significant difference in the radiation levels from the uncovered LEU core following a postulated LOCA scenario. Therefore, the changes do not increase the health and safety risk to the public from the postulated LOCA.

or

- The licensee assumed an acceptable loss-of-coolant scenario for the LEU core, which is consistent with the LEU facility design. The licensee calculated the fuel and cladding temperatures, associated fission product retention capabilities, and radiological consequences. The analyses and assumptions are applicable to the proposed LEU-fueled reactor. The radiological consequences are within the acceptance criteria for the HEU reactor. Therefore, the changes do not increase the health and safety risk to the public from the postulated LOCA.

13.3.2 Loss-of-Flow Accident (LOFA)

Areas of Review

LOFAs are postulated accidents that result from the loss of reactor coolant flow, but not the loss of the coolant itself. For the case of forced coolant flow, LOFAs may result from a loss of electrical power to a coolant pump or from an obstruction in a coolant channel. For natural convection flow, a LOFA may result from low head or obstructions in the coolant flow path. If forced coolant flow were lost, the system should revert to natural convection, which might involve a reversal of flow direction and temporary degradation in core heat transfer to the coolant. Loss of natural convection flow also would result in the degradation of core heat transfer. An increase in fuel and cladding temperatures may lead to fuel damage if (1) specified acceptable temperature limits are exceeded during the loss of forced flow transient or after steady-state natural convective cooling is established or (2) coolant flow stops during a loss of natural convective flow transient. The continued operability of the ultimate heat sink should be an area for review, along with corrective actions.

In the HEU to LEU conversion, several parameters associated with the thermal-hydraulic characteristics of the core may change. The areas of review should include the number or dimensions of coolant channels and fuel plates or rods, core dimensions, power density, fuel and cladding temperatures, surface heat flux, and thermal conductivity of the fuel. As a consequence, the time-dependent

temperature evolution may be different in the LEU core from the HEU core during LOFA situations.

The objectives of this review are to (1) compare the consequences for fuel integrity resulting from a LOFA scenario in the LEU core if the HEU and LEU LOFA scenarios are similar or (2) evaluate the licensee's demonstration of compliance with acceptance criteria used in the HEU safety analysis if a more severe LOFA scenario results from the conversion. The required analyses should include independent calculations of the HEU and LEU fuel temperature evolutions as a function of time in a LOFA, along with qualitative and quantitative comparisons of any differences in these temperature evolutions between the cores.

To accomplish these objectives, the reviewer should examine the postulated core and reactor conditions that are pertinent to the LOFA comparison, the thermal-hydraulic methods and assumptions used, the analyzed sequence of events, and the comparison with the HEU accident calculations for its scenario.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: The information should demonstrate that the consequences to fuel integrity resulting from a LOFA scenario in the proposed LEU-fueled reactor are not significantly greater than those projected in the licensed HEU-fueled reactor for the same accident scenario during the transient until conditions are stabilized. However, if the licensee can demonstrate that some changes in characteristics of the core thermal-hydraulics were required, some decrease in safety margin may be acceptable. In any case, the results of the safety analysis should show that fuel cladding integrity would not be lost as a result of the LEU core LOFA.

Review Procedures

The reviewer should compare the information submitted to determine if there are significant differences in the postulated LOFA events for the HEU- and LEU-fueled reactors. The review should reveal any differences and should trace the scenarios and analyses from initiation of the event until conditions are stabilized. The information should justify changes in reactor systems or operating characteristics designed to mitigate consequences to fuel, the reactor, and the public.

The reviewer should compare the analyzed consequences for the HEU- and LEU-fueled reactors and should also compare the consequences from the LEU LOFA with the acceptance criteria for the HEU LOFA.

Evaluation Findings

This section of the SAR should contain sufficient information to support one of the following type of conclusions, which will be included in the staff's safety evaluation report:

- The licensee assumed the same loss-of-flow initiating event and scenario for the LEU-fueled reactor as previously postulated for the licensed HEU-fueled reactor. The methods and assumptions are applicable to and acceptable for the proposed LEU-fueled reactor. The safety analysis demonstrates that the fuel and cladding temperatures are not significantly increased to the point of potential fuel damage by the conversion of the core from HEU to LEU fuel. The conversion does not increase the likelihood that fuel element integrity will be lost, nor that fission products will be released, and, therefore, the health and safety risk to the public from the postulated loss of flow accident should not increase.

or

- The licensee assumed a loss-of-flow initiating event and scenario applicable to the LEU core and reactor. The methods and assumptions are acceptable. The safety analysis demonstrates that the consequences of the event meet the acceptance criteria for the HEU core and reactor. Therefore, there is reasonable assurance that no undue risk will result to the health and safety of the public from the conversion to LEU fuel as a result of the postulated LOFA.

13.4 Other Accidents

The accidents discussed above for non-power reactors are postulated as the limiting accidents. However, in certain reactors, other accidents may be at least as important. These other accidents might include leakage of coolant into the cladding with a resulting steam pressure failure of the cladding, failure of a fueled or non-fueled experiment, physical damage to the core, or mechanical displacement of fuel. Misplaced experiments, mishandling of fuel, loss of normal electrical power, external events, and mishandling or malfunction of specific equipment are other potential accidents which may need to be considered. If these accidents are discussed in the HEU SAR, or if their possibility exists in the LEU core, they must be addressed in the conversion safety analysis. The general methodology used to review the specific accidents discussed in this chapter will be useful in the review of these "other" accidents.

14 TECHNICAL SPECIFICATIONS

Areas of Review

10 CFR 50.36 requires that the non-power reactor operating licenses contain technical specifications for the specific facility as an appendix to the license. ANSI/ANS 15.1-1990 is a specific standard developed with NRC support to guide the format and content for non-power reactor technical specifications. Chapter 14 of the format and content guide contains additional guidance on technical specifications.

The areas of review for HEU to LEU conversion are the technical specifications and their bases that address parameters related to the fuel, its safety limits, and core reactor physics parameters dependent on fuel characteristics. If the replacement LEU fuel were fabricated exactly like the HEU fuel and from the same materials, the only differences in reactor parameters would result from the uranium-235 enrichments. The technical specifications should describe fuel parameters that are essential to the safety analyses for the licensed HEU-fueled reactor. The technical specifications should be reviewed for consistency with the proposed LEU-fueled reactor.

Acceptance Criteria

Acceptance criteria should be based on the following considerations: Revised technical specifications should state the minimum changes required by differences in characteristics of the HEU and LEU fuels. For example, the revised technical specifications should accurately specify the uranium enrichment, the chemical and material form of the uranium in the fuel, and cladding materials and dimensions. The revised technical specifications also should include the safety limits, limiting safety system settings, reactor physics parameters, and other safety-related parameters for the proposed LEU-fueled reactor. The bases for any changed technical specifications should be traced to the LEU conversion safety analysis discussions and analysis from which they are derived. The format and content of the LEU technical specifications should not be different from those of the HEU-fueled reactor, except as required by the fuel conversion. The format should follow ANSI/ANS 15.1-1990.

Review Procedures

The reviewer should compare the proposed technical specifications for the LEU-fueled reactor with those for the existing HEU. The reviewer should compare the proposed changes in technical specifications with the bases and with the reactor analyses in the SAR that employ or derive them. The reviewer should compare the proposed changes in specifications with other tabulations of fuel and core

characteristics. The reviewer should compare the format of the proposed technical specifications with Chapter 14 of the format and content guide, the content with 10 CFR 50.36, and the specifications with the technical specifications for similar non-power reactors that have been converted from the use of HEU fuel.

Evaluation Findings

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

- The proposed changes in technical specifications are the minimum required by the conversion from HEU to LEU.
- The proposed technical specifications for the LEU-fueled reactor are consistent with the dynamic and safety analyses that have been reviewed and accepted. They provide assurance that the risks to the health and safety of the public from the LEU-fueled reactor will not be significantly greater than from the existing HEU fueled reactor and are acceptable.

15 OTHER LICENSE CONSIDERATIONS

This section includes some considerations and issues that might not be completely covered in other chapters. The examples given below are typical topics that should be considered, but additional topics may be necessary on a case-by-case basis. The reviewer should verify that the information submitted is consistent with the guidance in Chapter 15 of the format and content guide, as supplemented below.

15.1 Prior Utilization of Reactor Components

Two issues worthy of additional comment follow:

- Safety limits and resistance to failure of the fuel may be based on assumptions concerning the softening temperatures, burnup, cladding thickness, and strength of materials. The licensee should provide assurance that these parameters have not been changed significantly by prior use or storage and that the alloy or composition specifications of the materials are consistent with the assumptions in the analyses.
- Thermal-hydraulic analyses require knowledge of material composition, thickness, and heat conductivity. For pulsing reactors, heat capacity and temperature coefficients of reactivity also are essential parameters. The licensee should provide assurance that prior use or storage has not changed

these parameters significantly, and that factors resulting from oxide formation on cladding, depletion of fissile material, or burnable poisons are considered.

15.2 License Conditions

If the licensee proposes to continue to possess HEU fuel after conversion, license possession limits for uranium may need to be changed as discussed in this section of Appendix 18.1 of the format and content guide.

15.3 Decommissioning

The effects of conversion to LEU should be evaluated. Additional guidance in this regard may be found in Chapter 17 of the format and content guide.

References

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 1990.

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

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, LaGrange Park, Illinois, 1988.

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

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1987.

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

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

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.19, "Shipment and Receipt of Special Nuclear Material (SNM) by Research Reactor Facilities," ANS, LaGrange Park, Illinois, 1991.

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

Bullock, J. B., "Calculation of Maximum Fuel Cladding Temperatures for Two Megawatt Operation of the Ford Nuclear Reactor," Memorandum Report No. 1, Memorial Phoenix Project, Michigan, June 1962.

Gulf General Atomics, GA 6596, J. R. Shoptaugh, Jr., "Simulated Loss-of-Coolant Accident for TRIGA Reactors," August 18, 1965.

Hunt, C. H., and C. J. DeBevee, GEAP 3277, "Effects of Pool Reactor Accidents," General Electric Technical Information Series, November 2, 1959.

International Atomic Energy Agency, IAEA-TECDOC-223, "Research Reactor Core Conversion From the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels—Guidebook," Vienna, Austria, 1980.

International Atomic Energy Agency, IAEA-TECDOC-324, "Research Reactor Core Conversion From the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels—Guidebook Addendum on Heavy-Water-Moderated Reactors," Vienna, Austria, 1985.

International Atomic Energy Agency, IAEA-TECDOC-643, "Research Reactor Core Conversion Guidebook," Vienna, Austria, 1992.

Knexevich, M., et al., "Loss of Water at the Livermore Pool Type Reactor," Health Physics II, pp. 481–487, 1965.

Nyer, W. E., et al., "Transient Experiments With SPERT-1 Reactor," *Nucleonics*, June 1956.

Oak Ridge National Laboratory, ORNL 2892, J. F. Wett, Jr., "Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air," April 16, 1960.

Oak Ridge National Laboratory, ORNL TM-627, T. M. Sims and W. H. Tabor, "Report on Fuel-Plate Melting at the Oak Ridge Research Reactor, July 1, 1963," October 1964.

Shibata, Toshikazu, et al., "Release of Fission Products From Irradiated Aluminide Fuel at High Temperatures," *Nuclear Science and Engineering* 87, pp. 405-417, 1984.

Thompson, T. J., "Accidents and Destructive Tests," *The Technology of Nuclear Reactor Safety*, MIT Press, 1964.

U.S. Atomic Energy Commission, TID 14844, J. J. DiNunno, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1961.

U.S. Nuclear Regulatory Commission, NUREG-0849, "Standard Review Plan for Research Reactor Emergency Plans," 1985.

U.S. Nuclear Regulatory Commission, NUREG-0851, "Nomograms for Evaluation of Doses From Finite Noble Gas Clouds," 1983.

U.S. Nuclear Regulatory Commission, NUREG-1281, "Evaluation of the Qualification of SPERT Fuel for Use in Non-Power Reactors," 1987.

U.S. Nuclear Regulatory Commission, NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," 1987.

U.S. Nuclear Regulatory Commission, NUREG-1313, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors, 1988.

U.S. Nuclear Regulatory Commission, NUREG-1478, "Non-Power Reactor Operator Licensing Examiner Standards," 1993.

U.S. Nuclear Regulatory Commission, NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors," April 1981.

U.S. Nuclear Regulatory Commission, NUREG/CR-2198, "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA)," June 1981.

U.S. Nuclear Regulatory Commission, NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-fueled Reactors," April 1982.

U.S. Nuclear Regulatory Commission, NUREG/CR-4203, "A Calculational Method for Determining Biological Dose Rates From Irradiated Research Reactor Fuel," April 1985.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 1972.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I," October 1977.

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

U.S. Nuclear Regulatory Commission, DG8005, "Assessing External Radiation Doses From Airborne Radioactive Materials," 1991.

U.S. Nuclear Regulatory Commission, Memorandum to R. L. Tedesco from L. S. Rubenstein, "Design Basis Event for The University of Michigan Reactor," June 17, 1981.

Woodruff, W. L., "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," *Nuclear Technology*, 64, pp. 196-206, February 1984.

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC. Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG-1537
Part 2

TITLE AND SUBTITLE

Guidelines for Preparing and Reviewing Applications for the
Licensing of Non-Power Reactors, Part 2, Standard Review
Plan and Acceptance Criteria

3. DATE REPORT PUBLISHED

MONTH | YEAR

February | 1996

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

NUREG-1537, Part 2 gives guidance on the conduct of licensing action reviews to NRC staff who review non-power reactor licensing applications. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, conversions from highly enriched uranium to low-enriched uranium, decommissioning, and license termination.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Non-Power Reactor
Research Reactor
Test Reactor
Standard Review Plan

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE