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Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application

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OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE US DEPARTMENT OF ENERGY

Reactor and Nuclear Systems Division

PROPOSED GUIDANCE FOR PREPARING AND REVIEWING A MOLTEN SALT NON-POWER REACTOR APPLICATION

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CONTENTS

ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AHR	aqueous homogeneous reactor
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC	Advanced Reactor Concepts
CFR	US Code of Federal Regulations
DOE	US Department of Energy
ESF	engineered safety feature
HEU	highly enriched uranium
HVAC	heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
ISG	interim staff guidance
LEU	low enriched uranium
LOCA	loss-of-coolant accident
LWR	light water reactor
MHA	maximum hypothetical accident
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
NEI	Nuclear Energy Institute
NPUF	non-power production or utilization facility
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
SAR	safety analysis report
SNM	special nuclear material
SRP	standard review plan
SSCs	structures, systems, and components
TRIGA	Training, Research, Isotopes, General Atomics
TS	technical specifications

1 OVERVIEW

Development of non-power molten salt reactors (MSRs) are under consideration to further establish an MSR experience base, support the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.43(e), and provide any additional analyses needed for development of a full-scale MSR. Guidance provided in this report is based on MSRs operating with liquid fuel (i.e., fuel dissolved within a molten salt). These reactors, unless owned by the DOE or DOD, will require licensing by the US Nuclear Regulatory Commission (NRC) staff. Standard review plan (SRP) guidance for large light water reactors (LWRs) is available in NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants; Light Water Reactor (LWR) Edition* [1]. However, NRC staff observed [2] that NUREG-0800 is very cumbersome to apply to non-power reactors "because of the great differences in complexity and hazards between non-power reactors and nuclear power plants." Therefore, a program to develop performance-based guidance applicable to non-power reactors was initiated.

In 1996, NUREG-1537, Parts 1 and 2, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,* was published [2,3]. Part 1, the format and content guide, suggests a uniform format for presenting information in non-power reactor applications that is acceptable to the NRC staff, but conformance with the format and content is not required. Part 2, the SRP, ensures the quality and uniformity of the staff review of an application. Unfortunately, the application guidelines and SRP do not provide adequate guidance for all advanced non-LWR technologies and applications. This discrepancy eventually led to the 2012 development of interim staff guidance (ISG) for NUREG-1537 [4,5], which includes criteria for describing and reviewing aqueous homogeneous reactors (AHRs). Specifically, NUREG-1537 ISG, 2012 expanded the original document to address three areas:

1. updated criteria for heterogeneous non-power reactors,
2. criteria for licensing AHRs, and
3. criteria for licensing a Part 50-licensed isotope production facility.

In 2015, the US Department of Energy (DOE) opted to build on the AHR NUREG-1537 ISG experience by performing a gap analysis of the guidance that would be used to license a non-power MSR. MSRs represent one of the advanced non-LWR technologies selected by DOE for development through a multiyear cost share award with Southern Company Services. Under this Advanced Reactor Concepts 2015 (DOE Advanced Reactor Concepts [ARC] 15) award program, the DOE tasked Oak Ridge National Laboratory (ORNL) to evaluate the guidance changes that the NRC may need to consider when licensing an MSR non-power reactor. ORNL staff, with support from Boston Government Services, LLC, focused on five system-related chapters in NUREG-1537 that were considered most relevant to inform the effort that would be required for a non-power MSR applicant. ORNL documented this review in a technical report, ORNL/TM-2018/834 [6], *Proposed Guidance for Preparing and Reviewing Molten Salt Non-Power Reactor License Applications* (NUREG-1537). The report was subsequently shared with industry and the NRC.

The 2018 review was limited in scope, focusing on key system chapters based on the expected significance of each chapter relative to expected differences in addressing advanced non-LWR technologies, specifically non-power MSRs, compared with heterogeneously fueled non-power reactors. In the ORNL report, proposed generic adaptations were suggested for the following NUREG-1537 chapters:

- Chapter 4, "Reactor Description"
- Chapter 5, "Reactor Coolant Systems"
- Chapter 6, "Engineered Safety Features"

- Chapter 9, "Auxiliary Systems"
- Chapter 11, "Radiation Protection Program and Waste Management"

The inclusion of Chapter 11 in the previous review effort was intended to provide guidance for categorizing the waste-handling process for an MSR operating with homogenous fuel. The introductions from Parts 1 and 2 of the 2012 NUREG-1537 ISG provide guidance for the application and review of production facilities. After a period of operation, non-power MSRs with homogenous fuel will include gaseous and soluble fission products. The gaseous fission products will be collected and held for decay in an off-gas system. There might also be an initiative to polish or filter the soluble fission products in the fuel salt by some mechanical or chemical means. The treatment and handling of fission products in the non-power MSR fuel salt and the description of this process in the safety analysis report (SAR) must be very precise to avoid the waste treatment facility being construed as a co-located special nuclear material (SNM) fuel cycle facility (see Section 2.3 of this report).

Subsequent to the release of ORNL/TM-2018/834, NRC staff expressed a desire to continue the regulatory gap analysis that was begun in that report. This would provide additional clarity and information addressed in certain sections of the original report, while also providing new guidance on certain topics not addressed in the original report. This revision would benefit the NRC staff reviewing applications involving non-power MSR designs and would help developers understand how the NRC staff might approach the review of such applications.

The focus of this report is to provide infrastructure support to the NRC staff for the regulatory review of non-power MSRs.

2 DESCRIPTION OF WORK

The NUREG-1537 parts included in the appendixes of this report were evaluated relative to (1) the MSR regulatory gap analysis performed under the DOE ARC 15 award, (2) NUREG-1537 ISG, 2012, and (3) a generic working knowledge of MSRs operating with homogenous fuel based on documentation of the Molten Salt Reactor Experiment (MSRE) operation at ORNL. Most non-power reactors contain heterogeneous fuel elements consisting of rods, plates, or pins, and the fuel cladding acts as the initial fission product barrier. In a liquid fueled MSR, some fission products are soluble in the salt and are retained, whereas for noble gases and other non-soluble fission products, the initial fission product barrier is the fuel salt system boundary and the interfacing system boundaries. These MSR elements affect some of the traditional discussions of fuel, coolant systems, accidents, and many of the auxiliary systems required.

2.1 PROJECT ASSUMPTIONS AND GUIDELINES

NUREG-1537 provides performance-based guidance for research reactors and testing facilities, including SAR format, content, and review guidance. It is written in technology-neutral detail, but until the implementation of the AHR ISG, the guidance assumes the use of a heterogeneous fuel form. There was compelling interest in expanding the MSR technology-specific guidance and requirements beyond what was originally intended by NUREG-1537. Therefore, the following assumptions and guidelines were applied to bound the task:

1. The proposed MSR guidance should provide the same level of detail provided in NUREG-1537 [2,3] as augmented by the ISG for Aqueous Homogeneous Reactors [4,5].
2. Only generic MSR technology insights will be provided because specific design details for any particular non-power MSR technology are unknown.

3. Consistent terminology will be used based on the definitions provided in the previous work; definitions will be added as necessary.
4. Only a high-level review of the codes and standards cited in each chapter will be performed regarding the applicability and adaptability to a non-power MSR. An in-depth review of codes and standards is not part of the task.
5. Generic safety-related structures, systems, and components will be identified where appropriate.
6. Unique MSR considerations will be identified where appropriate, even if detailed discussion is not possible. This will likely include topics such as isotope production and chemical separation.

In NUREG-1537 ISG, 2012, some chapters are marked for complete replacement relative to those in NUREG-1537, and others are simply outlined with a limited number of changes to make the document applicable to AHRs. The AHR ISG changes were applied to the 1996 version of NUREG-1537 materials prior to applying any proposed MSR changes identified in this report.

The introductory material for NUREG-1537 indicates that the guidance is for non-power MSRs. For clarity, the introduction notes that MSRs are a class of reactors in which a molten salt performs a significant function in the core. However, unless otherwise noted, the terms *MSR* and *reactor* as used in this guidance are understood to mean a liquid fueled non-power MSR.

2.2 EFFORT AND SCOPE

The NRC staff directed ORNL to revise ORNL/TM-2018/834 with attention to the list of topics included below. The direction indicates that the revised technical report should generally provide chapter-by-chapter SAR and review guidance recommendations based on NUREG-1537, which identifies and addresses potential technical and regulatory gaps for non-power MSRs, including consideration of the following:

- Relevant chapters and references from NUREG-1537 [2,3]:
    - *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*, NUREG-1537, Part 1, February 1996 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML042430055)
    - *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,* NUREG-1537, Part 2, February 1996 (ADAMS Accession No. ML042430048)
- Relevant chapters and references from the Interim Staff Guidance augmenting NUREG-1537 [4,5]:
    - "Final Interim Staff Guidance Augmenting NUREG-1537, Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012 (ADAMS Accession No. ML12156A069)
    - "Final Interim Staff Guidance Augmenting NUREG-1537, Part 2, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria*, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012 (ADAMS Accession No. ML12156A075)
    - The applicability or adaptability of existing codes and standards including industry standards (e.g., American National Standards Institute [ANSI]/ American Nuclear Society [ANS], Institute of Electrical and Electronics Engineers [IEEE]) and other NRC guidance (e.g., Regulatory Guides, NUREGs)

- The revised definitions of non-power reactor, research reactor, and testing facility included in the draft final rule, "Final Rulemaking: Non-Power Production and Utilization Facility License Renewal (RIN 3150-A196, NRC-2011-0087)", January 11, 2019 (ADAMS Accession No. ML19008A088)," subsequently released on June 17, 2019 as SECY-19-0062 [7]
- The selection of structures, systems, and components (SSCs), including appropriate references to the advanced reactor design criteria and designation of safety related SSCs
- Minimum regulation requirements imposed on facilities licensed under Subsection 104 (c) of the Atomic Energy Act of 1954, as amended
- Acceptable accident analysis techniques and representative accident sequences (e.g., maximum hypothetical accident technique or other alternatives to using probabilistic techniques)
- To the extent possible, any unique considerations the NRC staff should account for related to
    - Material control and accounting
    - Physical protection
    - Emergency preparedness
    - Conduct of operations, including operator training and requalification
    - Environmental reviews
    - Quality assurance program description
    - Facilities possessing Category I, II, or III quantities of material
    - Facilities with limited staffing capabilities and needs
- Technical specifications (TS) unique to MSRs
- Criticality control and chemical hazards, especially as they relate to online or batch fuel cleanup processes
- Differences in the level of detail needed for the review of a construction permit application versus an operating or combined license application
- Any necessary updates to guidance for production facilities to accommodate fuel cleanup processes at MSRs

ORNL staff participated in a Nuclear Energy Institute MSR technology working group meeting to gather feedback on the revision approach and topics to be included. Although no specific feedback items were provided for inclusion in a guidance document during the meeting, industry stakeholders noted that they are beginning to interact with the NRC on their respective MSR technologies. Several vendors noted that it was crucial to start laying the groundwork for MSR research reactors and testing facilities and that an MSR-specific guidance document is needed to support further discussion.

Each chapter was reviewed, and proposals were made to remove references to LWR designs or to assess the applicability of LWR-related statements to MSR designs. The unique characteristics of non-power MSRs were also considered when formulating performance-based criteria for these reactors. The chapter revisions were compared among the ORNL subject matter experts and iterated for better compatibility. Finally, all the proposed chapter adaptations were collected and reviewed to ensure adequate coverage and placement of non-power MSR design detail.

The review team included some limited MSR informational material in the proposed adaptations to serve as a basic reference to reviewers addressing the unique nature of MSR designs. This approach is like that taken in the development of NUREG-1537 ISG, 2012.

The proposed MSR guidance is intended to be generic enough to cover different design alternatives, many of which are included in the MSR discussion. It is understood that any given MSR design might not employ all the MSR design characteristics (structures, systems, and components) discussed in the guidance. The proposed MSR guidance also includes nomenclature that is intended to be generic. Specific vendor designs will likely include alternate system or component terminology. For example, systems supporting the fuel salt, the primary cooling system, and the heat dissipation system are likely to be referred to by design-specific system names. To that end, a glossary was developed and inserted into the introductory sections for parts 1 and 2 of the suggested guidance for reference.

2.2.1 Terms and Definitions

The introductory sections for parts 1 and 2 of the guidance define terms often used when discussing an MSR. The intention is not to set terminology for all possible MSR technologies. If the suggested guidance does not properly characterize a new technology, then vendors and applicants should introduce appropriate substitute terminology and provide definitions. The following terms are specific to MSR technologies or have a definition that is different from its LWR counterpart. The defined terms are:

Active Reactor Core: In an MSR, the vessel region occupied by the fuel salt, where the majority of prompt neutrons are generated and where most fissions occur. In an MSR, the active reactor core geometry might change with time as a result of changes in density and voiding of the solution.

Coating or Cladding: Intervening protective layer of material between the fuel salt and the structural container alloy. Also included are surface modifications of the structural alloy to enhance its chemical or mechanical performance by altering its microstructure or composition (e.g., carbiding, phosphiding, or nitriding the surface).

Control Element(s): Object(s) employed to adjust reactivity. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned.

Emergency Cooling System: A system that provides decay heat removal from the reactor fuel following an accident (e.g., a direct reactor auxiliary cooling system or a reactor vessel auxiliary cooling system). Similar cooling systems for fuel drain tanks are also included, as are systems such as in-floor heat pipes, to provide cooling to fuel located in outer containment layers in the event of fuel system breach type accidents.

Fuel Barrier: Portion of the fuel system boundary in contact with the liquid fuel after addition to the fuel circuit and prior to transfer to waste handling (principally the vessel, chemical processing system boundary, drain tank [if used], heat exchanger, cooling thimbles, Control element thimbles, instrumentation thimbles, piping, tanks, and valves).

Fuel System Boundary: The material that mitigates the release of radionuclides from the reactor fuel, including volatile fission products (e.g., krypton, xenon, iodine). For an MSR, this includes the vessel, drain tank (if used), cooling thimbles, heat exchangers, chemical processing system boundary, waste-handling tank, pumps, valves, and piping—it essentially includes the radionuclide barrier and the fuel barrier.

Gas Management System: The cover gas system provided to capture volatile fission products (e.g., krypton, xenon, iodine) until ultimate discharge and to provide venting of any pressure/density transients that could result in damage to the vessel or the fuel salt/primary cooling system salt heat exchanger, thus resulting in loss of the fuel system boundary.

Heat Dissipation System: A set of components or system(s) that interface with the primary cooling system to provide the principal means of transferring the heat from the active reactor core to an ultimate heat sink. The heat dissipation system might use a variety of coolants (e.g., salt, liquid metal, gas, or water) but does not contain fuel.

Neutron Moderator: In an MSR, materials in or near the active reactor core that consist of light elements (e.g., H, B, C). Moderators are generally solid form.

Primary Cooling System: The system that directly interfaces with the fuel system boundary at the fuel salt/primary cooling system salt heat exchanger(s) to provide the principal means of removing heat from the fuel salt during operation by transferring the heat to the heat dissipation system. The primary cooling system may employ a variety of coolants (e.g., nitrate salt, halide salt, etc.), but it does not contain fuel.

Radionuclide Barrier: The portion of the fuel system boundary that serves as the innermost low-leakage barrier to the radionuclides within the gas management system.

Reactor Fuel: In an MSR, fuel salt consisting of fissionable and possibly fertile halide salts, fission products, and generally solvent halide salt(s).

Vessel: For an MSR, the structure containing the active reactor core. In certain design configurations, other components such as heat exchangers might reside in the vessel but outside the active reactor core.

An equivalent set of terminology is not available in NUREG-1537, but a pre-defined set of terms helps clarify the discussion of MSR technology.

2.2.2 Chapter Summary

Each chapter was reviewed and adapted for MSR technology and terminology. The following section summarizes chapter content and provides a high-level summary of significant changes from the NUREG-1537 guidance.

Introduction

The current introductory material provides background information on the development of NUREG-1537 review guidance, the licensing process, differences between research reactors and testing facilities, the overall structure of the document, and the general requirements for an application. The introductory material also provides significant guidance on the printed format of the application document.

The introduction includes significant updates for MSR technology. The introduction provides basic MSR background that is relevant to the entire document. Some generic MSR terminology is defined here, as well, for reference. References to applications submitted prior to January 1, 1994, have been removed. In addition, the intricate guidance on the printed application format is removed and replaced with a link to electronic submission guidance.

1 – Facility

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

Minor terminology changes have been applied to account for MSR technology.

2 – Site Characteristics

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

English measurements have been included in brackets following the International System of Units to provide easier comparisons with other reactor siting guidance such as that found in RG 4.7 for power reactors.

3 – Design of SSC

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

Minor terminology changes have been applied to account for MSR technology, and the chapter references were revised.

4 – Molten Salt Reactor Description

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

In ORNL/TM-2018/834, the discussion was expanded to include the vessel, the active reactor core, enrichment, uranium loading, and the chemistry of the reactor fuel (including fresh and reloaded fuel composition), salt type, plutonium loading (if applicable), expected fissile density in solution at operational pressure, temperature and redox conditions, fissile material solubility, build-up of fission products and related decay daughters in the fuel salt, precipitates, control elements, neutron moderation (if applicable), nuclear design, thermal-hydraulic design, and the gas management system. The chapter layout has been revised to support MSR technology.

5 – Molten Salt Reactor Cooling Systems

Chapter 5 lists the design bases and describes the functions of the reactor coolant and associated systems at the facility, including the primary and secondary systems as applicable, as well as the coolant makeup and purification systems. Provisions for adequate heat removal are described for when the reactor is operating and when it is shut down.

In ORNL/TM-2018/834, the discussion has been expanded to include the reactor fuel as the initial coolant, the MSR fuel system boundary, the primary cooling system, and the heat dissipation system as the principal heat transfer mechanisms for removing heat from the core (Chapter 4) to the environment, heat transfer mechanisms from other inter-connected systems such as the drain tank system or the gas management system, and fuel salt makeup and purification systems. The chapter layout was revised to support MSR technology.

6 – Engineered Safety Functions

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

In ORNL/TM-2018/834, the discussion is revised from describing a fuel cladding breach to include a description of a release of fuel salt beyond the fuel system boundary or any additional barriers (i.e., guard pipe or guard vessel), a release of gaseous fission products from the vessel or inter-connected systems, and mitigation systems used to bound credible accidents in the accident analysis.

7 – Instrumentation and Controls

Chapter 7 lists the design bases and describes the functions of the instrumentation and control (I&C) systems and subsystems at the facility, emphasizing safety-related systems and safe reactor shutdown.

Typical MSR parameters were added for the reactor control system and the reactor protection system. The potential for SSC other than the control elements to be used for reactivity control was added to the discussion where appropriate. Minor terminology changes were applied to account for MSR technology. Specific references to pulsing operation of the reactor have been eliminated.

8 – Electrical Power Systems

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

Minor terminology changes have been applied to account for MSR technology.

9 – Auxiliary Systems

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

In ORNL/TM-2018/834, the discussion has been expanded to include information on SNM used for both new and irradiated reactor fuel, including components (tanks, valves, pumps, instrumentation, controls), related cooling systems, processes (chemical blending, SNM transfers, waste storage, preparation for shipment), criticality control and monitoring, vaults, shielding, and contamination control. Provisions for cooling the gas management system discussed in Chapter 4 have been included as an auxiliary system. The option to discuss cooling systems for the fuel salt drain tank, primary cooling system drain tank, gas management system cooling, chemical polishing system cooling, and any other MSR-related support systems that require cooling has been included.

10 – Experimental Facilities

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

In addition to traditional experimental purposes, it was noted in the discussion that a non-power MSR may be used to gather information and data that could be useful for the purposes of licensing future prototype facilities and power reactors. Furthermore, the reactor itself can be considered an experimental facility to demonstrate MSR technology for eventual prototype and commercial scale up. Minor terminology changes have been applied to account for MSR technology. References to pool water level have been removed.

11 – Radiation Protection and Waste Management

Chapter 11 lists the design bases and describes the functions of the radiation protection and the radioactive waste management programs at the facility. This chapter also describes the control of byproduct materials produced in the reactor and utilized under the 10 CFR Part 50 reactor operating license. The description of the radiation protection program should include health physics procedures, monitoring programs for personnel exposures and effluent releases, and assessment and control of radiation doses to both workers and the public. The program to maintain radiation exposures and releases as low as reasonably achievable (ALARA) includes the control and disposal of radiological waste from reactor operations and from experimental programs.

In ORNL/TM-2018/834, the discussion has been expanded to consider that fission products are released to the liquid salt fuel solution and contained by the fuel system boundary. Gaseous fission products will be gathered and processed within the radionuclide barrier. The fission gas may require holdup for decay or further treatment before being released to the environment or disposed as waste. Residue from any cleanup and polishing of the liquid fuel will be laden with fission products and will require treatment as radioactive waste.

Discussion has been added regarding the precision and clarity needed to distinguish between the process of fission product waste removal and the production of SNM. Undesired fission products are typically removed from the liquid fuel by waste treatment processes with no intention of producing SNM. However, the reactor license must be worded to ensure that “separation of isotopes” or the “separation of byproduct materials” is not interpreted to indicate conditions that prevent this cleanup process.

12 – Conduct of Operations

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

A paragraph referencing initial reactor operator training was added to Section 12.10. Additional discussion has also been added to the startup planning for an MSR in Section 12.11.

The former section on environmental reports was removed. Adequate information on environmental review is provided in other regulatory documents. References on this topic are provided in the document introduction.

13 – Accident Analyses

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

This section also notes that the class of MSRs considered in the accident analysis uses liquid fuel rather than solid heterogeneous fuel, and the resulting fission products—both liquid and gaseous—must be contained within the facility barriers rather than within heterogeneous fuel cladding. This reinforces the discussion initially provided in the Introduction.

Limiting phenomena for MSRs are noted as fuel salt precipitation, fission-product precipitation, fuel salt chemistry/physical properties, delayed neutron production, core voiding, and tritium production. Each phenomenon is discussed separately. The list of postulated events has been revised to be specific to MSRs, and the typical parameters to be tracked during postulated events are listed. The discussion of the MSR MHA is revised to be more applicable to MSR technology. A revised discussion is provided of each listed postulated event.

The list of references was updated to reflect homogenously fueled reactors.

14 – Technical Specifications

Chapter 14 presents an overview of technical specifications (TS), which state the operating limits, conditions, and other requirements for the facility to ensure adequate protection of the public's health and safety. Appendix 14.1 specifies the format and content of TS for MSRs.

Minor editorial changes have been made to the TS overview in the main body of the chapter. The definitions in Appendix 14.1 have been revised to address MSR technology. Various discussions of heterogeneous fuel types were eliminated from the safety limit discussion in Section 2.1. MSR operation is described in the discussion of process variables in Section 2.1.1. References to Training, Research, Isotopes, General Atomics (TRIGA) reactors and pulsing operation were removed throughout the appendix.

The reactor core parameter overview in Section 3.1 was revised to reflect the processes and parameters associated with the fuel salt, the fuel system boundary, and MSR operating parameters. This includes additional discussion on core geometry and configurations. A new subsection has been added to discuss monitoring expected fuel composition changes. The discussion on coolant chemistry requirements was revised to reflect the use of reactor fuel and fuel composition changes during operation instead of water.

The coolant system discussion in Section 3.3 has been revised to reflect the process for heat removal from the reactor fuel to the environment through the primary cooling system and the heat dissipation system. The discussion on coolant chemistry requirements was revised to reflect the use of a coolant salt instead of water.

The discussion of containment or confinement in Section 3.4 was revised to reflect the low operating pressure of MSRs and the barrier approach to fission product release.

The discussion of effluents in Section 3.7.2 has been revised, as there is no need for ^{41}Ar to be called out specifically because a broader spectrum of fission products must be considered for routine operations such as in the MSR off-gas, polishing, and sampling systems. In addition, air entrainment in the fuel salt will be limited by the gas management system, which will minimize production of ^{41}Ar.

The list of references has been updated to reflect homogenously fueled reactors.

15 – Financial Qualifications

Chapter 15 concerns financial qualifications of the non-power reactor applicant for initial construction, continuing operations, and decommissioning. Adequate information on financial qualification is provided in NUREG-1537. Therefore, this chapter is not used for the MSR guidance and a reference to NUREG-1537, chapter 15 is provided in the document introduction regarding applicant financial qualifications.

16 – Other Licensing Considerations

Chapter 16 is reserved to discuss other licensing considerations such as prior use of reactor components. This chapter is available for an applicant to address any other licensing considerations not discussed elsewhere in the SAR. The section on medical use of non-power reactors has been removed. No MSR-specific materials have been added.

17 – Decommissioning

Chapter 17 provides guidance on decommissioning. Adequate information on decommissioning is provided in NUREG-1537. Therefore, this chapter is not used for the MSR guidance and a reference to NUREG-1537, chapter 17 is provided in the document introduction regarding decommissioning.

18 – HEU-to-LEU Conversions

The original Chapter 18 discusses the conversion of the reactor from highly enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, and it includes topics covered in Chapters 1 to 17 regarding HEU-to-LEU conversions. HEU-to-LEU conversions are not considered for the MSR work. Therefore, this chapter is eliminated for the MSR guidance.

Appendix A – Applicability of Selected Regulations in Title 10, Chapter I of The Code of Federal Regulations to Non-Power Reactors

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

The task scope does not include updating this section. Review of the appendix will be covered under a more generic analysis of non-power reactors. Therefore, Appendix A is not included in the proposed guidance update.

2.3 PRODUCTION FACILITY CONSIDERATIONS

A *production facility* is defined in 10 CFR Part 50.2 as follows:

1. Any nuclear reactor designed or used primarily for the formation of plutonium or ^{233}U
2. Any facility designed or used for the separation of plutonium isotopes, except laboratory-scale facilities designed or used for experimental or analytical purposes only
3. Any facility designed or used for the processing of irradiated materials containing SNM, except
    a. laboratory-scale facilities designed or used for experimental or analytical purposes
    b. facilities in which the only SNMs contained in the irradiated material to be processed are uranium enriched with ^{235}U isotopes and plutonium produced by the irradiation, if the material processed contains not more than 10^{-6} grams of plutonium per gram of ^{235}U and has fission product activity not in excess of 0.25 mCi of fission products per gram of ^{235}U
    c. facilities in which processing is conducted pursuant to a license issued under parts 30 and 70 of this chapter, or equivalent regulations of an agreement state for the receipt, possession, use, and transfer of irradiated SNM, which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission products and limits the process batch to

not more than 100 grams of uranium enriched in ^{235}U and not more than 15 g of any other SNM

A *utilization facility* is defined in 10 CFR Part 50.2 as

1. any nuclear reactor other than one designed or used primarily for the formation of plutonium or ^{233}U, or
2. an accelerator-driven subcritical operating assembly used for the irradiation of materials containing SNM and described in the application-assigned docket number 50-608.

Once operated, MSRs will contain gaseous, soluble, and insoluble fission products in the fuel salt. The gaseous fission products must be removed from the core and cover gas to allow for continued operation of the non-power MSR. This gas removal is the function of the gas management system. In addition, some non-power MSR designs might include a chemical or mechanical polishing system to filter soluble fission products that act as neutron poisons. MSR vendors would intend these actions to be part of the routine waste handling of the non-power MSR operation. As such, these activities would be covered under NUREG 1537, Chapter 11, “Radiation Protection Program and Waste Management.” An applicant should be sensitive to the potential for the separation of fission products to be viewed as a co-located SNM fuel cycle facility. Therefore, some enhanced discussion of MSR waste handling is warranted and is included in the adaptation of Chapter 11 for MSRs. An application for a non-power MSR license should explain why the waste management process does not involve separation of SNM. Strong discussion should reinforce the understanding that no facility exists to separate actinides.

3 OBSERVATIONS ON SIGNIFICANT NON-POWER MSR ADAPTATIONS

The material in NUREG-1537 is based on more than 50 years of experience with commercial and non-power reactors using heterogeneous fuel, including extensive research conducted on topics pertinent to LWRs. NUREG-1537 documents a performance-based approach for establishing non-power reactor design criteria and ensuring they are met. This is a familiar, accepted approach for non-power LWR application reviews that is proven by decades of operating experience, research, and revisions to the review methodology when justified and necessary. However, a previous gap analysis[1] on NUREG-1537 Chapters 4, 5, 6, 9, and 11 revealed fundamental differences in the potential suitability of heterogeneous fuel-based technology applied to the assessment of reactor components for an MSR design.

Key principles of MSR technology have been demonstrated through limited operating experience. This experience is not as comprehensive as it is for LWR technologies, so a review of performance-based methods applied to MSR technology will prove more difficult initially. Individual NUREG-1537 chapters must be revised to explicitly refer to MSR technology to enhance the review process for these reactors. This observation was borne out, as revised MSR guidance was developed previously for selected NUREG-1537 chapters. Any adaptations for MSR technology must consider review processes that ensure the adequacy of proposed designs while recognizing the limited extent of operating experience with MSR designs. The proposed MSR guidance is in accordance with the *Atomic Energy Act* of 1954, as amended, for the licensing of non-power reactors. Highlights of the proposed MSR guidance with significant adaptations to the guidance found in NUREG-1537 follow.

If the suggested guidance does not properly characterize a new technology, then vendors and applicants should introduce appropriate design and regulatory arguments regarding the differences.

[1] R. J. Belles, G. F. Flanagan, and M. Voth, *Regulatory Gap Analysis of Guidelines for Preparing and Reviewing Applications for Licensing of Non-Power Reactors (NUREG-1537) for Applicability to Molten Salt Reactors*, ORNL/SR-2016/725, December 2016, Distribution controlled by DOE sponsor.

3.1 CHAPTER 4, "MOLTEN SALT REACTOR DESCRIPTION"

The most significant change in Chapter 4 is regarding the references to heterogeneous fuel elements consisting of rods, plates, or pins with fuel cladding acting as the initial fission product barrier. A liquid fueled MSR uses homogenous fuel with no cladding. Therefore, with the exception of soluble fission products, the initial fission product barrier in an MSR is the fuel system boundary. This requires modification of fuel discussions and descriptions throughout the chapter. References to fuel melting and integrity must be replaced with appropriate discussions of the fuel solution and the integrity of the fuel system boundary.

The discussion of reactor fuel chemistry in Section 4.2, "Reactor Core," was revised. The formation of gaseous, soluble, and insoluble fission products in the fuel salt will affect system chemistry. References to pH control were universally revised to redox in the acid-based fuel salt solution. Chemistry control, mitigation, and redox tactics must be identified by the applicant for short- and long-term changes in the chemistry of the fuel salt. In general, fuel qualification is a thermophysical and thermochemical process rather than a mechanical process, as it is for heterogeneous fuel. Additionally, fuel purification might play a role in a non-power MSR design; if so, it must be detailed by the applicant per the guidance in Chapter 5.

Neutron-absorbing control rods are the typical means for controlling reactivity in an LWR, but there are many more ways to control reactivity in an MSR. MSR designs can control reactivity through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. As a result, the term *control elements* is universally used in place of the term *control rods*. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned. References to *fully inserted* or *fully withdrawn* elements must be carefully considered regarding what condition relates to the most or the least reactivity state.

The core support structure discussion has new implications regarding fuel salt. The fuel-positioning function of a heterogeneous reactor core support structure is not applicable to an MSR. Since the active reactor core is fluid fuel salt, the MSR core support structure is the vessel. The core support structure discussion was revised to include the vessel and the reflector's vertical and lateral support structure, as well as support structures for the reactor control, instrumentation, cooling components, and any other components connected to the vessel.

The International Atomic Energy Agency (IAEA) categorizes research reactors into three common types [8]:

1. pool-type, in which "the core is a cluster of fuel elements sitting in a large open pool of [fluid]"
2. tank-type, in which "the core is contained in a vessel"
3. tank-in-pool type, in which "the core is located in a pool, but enclosed in a tank through which the coolant is pumped"

Although a tank-type description closely fits MSR technologies, for clarity, this terminology was revised to *use of a gas-tight vessel*. Likewise, pulsing-power MSRs are not under consideration, so discussion of this design feature was removed.

Some delayed neutrons will be produced outside the active reactor core because of the nature of the flowing fuel solution and its movement out of the active reactor core. Any sudden reduction in flow, such as a transition from forced flow to natural circulation flow, will result in a subsequent reactivity addition. Therefore, the ability to control the core during normal operation and core flow changes must be considered by the applicant.

Gaseous, soluble, and insoluble fission products are generated within the fuel salt as an MSR is operated. Gaseous fission products accumulate within the vessel or within a cover gas at a free surface boundary. Text was added in a new Section 4.7, "Gas Management System," that requires the applicant to describe the design of the system for removing fission product gases from the core and cover gas of the MSR. Any decay heat removal provided by the gas management system to the overall reactor cooling is summarized in Chapter 4 and detailed in Chapter 5. Heat removal from the gas management system by the auxiliary cooling system is addressed in Chapter 9.

Fuel burnup does not have the same limitations for a homogenous MSR core as it does for a heterogeneous core. The term *burnup* relative to the reactor fuel is universally replaced by the phrase *composition changes*. The fuel's composition changes over time, impacting core reactivity. This should be addressed by the applicant in Section 4.5, "Nuclear Design."

The reference to Regulatory Guide (RG) 2.1, "Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors," has been replaced by RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," issued May 2009. This is a generic non-power reactor recommendation that applies across all technologies.

3.2 CHAPTER 5, "MOLTEN SALT REACTOR COOLING SYSTEMS"

Liquid homogeneous MSR fuel salt dissipates heat through a heat exchanger to cooling systems that do not include fuel. Therefore, the former heterogeneous fuel-based primary coolant system no longer exists with the same function as provided for LWR fuel. For clarity, the proposed title for Chapter 5 was revised from "Reactor Coolant Systems" to "Molten Salt Reactor Cooling Systems."

Some brief background on generic MSR heat transfer is provided in Chapter 5. The heat transfer systems are somewhat more complex than those required to cool heterogeneous-fueled reactors.

Section 5.2 was retitled "Fuel Salt Heat Transport" since the discussion is no longer directed at the function of the coolant in direct contact with the heterogeneous fuel assemblies. This section was refocused for the applicant to address heat transport within the fuel system boundary to the fuel salt/primary cooling heat exchanger(s) and to include system drawings and operating parameters. The specifics for the fuel salt are discussed in Chapter 4, and general thermal-hydraulic properties are discussed in Section 4.6.

Section 5.3, "Secondary Coolant System," was retitled to simply "Cooling Systems" and was divided into two subsections. This section provides parameters for the applicant to describe how reactor heat is removed and transferred to the environment, typically through two or more cooling loops. The first subsection, "Primary Cooling System," provides parameters for the applicant to describe the salt system that directly interfaces with the fuel salt through the fuel salt/primary cooling heat exchanger(s). The second subsection, "Heat Dissipation System," interfaces with the primary cooling system subsection and provides the parameters for the applicant to describe all remaining heat transport loops required by the design to disperse the reactor heat to the environment. Heat dissipation systems can use a variety of coolants, including salt, water, and liquid metal.

Gaseous, soluble, and insoluble fission products build up in the fuel salt as an MSR is operated. Gaseous fission products accumulate within the vessel or within a cover gas at a free surface boundary. Because this is not an issue with the heterogeneous fuel discussed in the original document, a section was added to Chapter 4 that describes the gas management system and handling of gaseous fission product buildup. Section 5.4, "Fuel Salt Cleanup System," provides the complementary parameters for the applicant to describe any cleanup or salt polishing system included in the MSR design to handle the buildup of soluble and insoluble fission products.

Provisions are made for salt makeup in the various MSR salt loops. A section in Chapter 5 that addresses makeup was revised to provide for the addition of salt to the fuel salt and to the primary cooling system salt. Likewise, a drain tank is typically provided to allow for safe storage of the fuel salt in the event of a design-basis accident or for fuel system boundary maintenance. A new section in Chapter 5 was added to describe the parameters of a fuel salt drain tank, if applicable to any particular design.

3.3 CHAPTER 6, "ENGINEERED SAFETY FEATURES"

The discussion of confinement and containment is generally applicable to all non-power reactor designs. However, it is noted in the proposed chapter adaptation that multiple confinement or containment boundaries might be included for an MSR design, depending on the relative location of the gas management system and other cleanup systems to the fuel system boundary. Likewise, an MSR technology may employ multiple barriers to satisfy confinement or containment requirements (i.e., functional containment). Therefore, while the traditional function of confinement and containment continues to be applicable to MSRs, the concepts may be altered slightly.

The current version of Section 6.2.3, "Emergency Cooling System," assumes that heterogeneous fuel is used in the reactor design and that it must be continuously covered and cooled to maintain fuel integrity. Consequently, the issue of concern has been the continued heat removal from the fixed core to maintain fuel integrity and coolable core geometry. Loss-of-coolant accidents (LOCAs) figure prominently in the discussion of the current requirements because of the importance of cooling the heterogeneous fuel and preventing core melt, which can lead to uncoolable core geometry. However, the MSR LOCA concern is at the fuel system boundary, which must be protected from overheating, as it can result in a boundary material failure, which could have a disruptive effect on continued fuel salt cooling. Therefore, the chapter was revised to indicate that for the homogenous fuel in an MSR, maintaining a decay heat removal path for continued boundary integrity is the key safety issue. Decay heat removal can be provided by multiple means, including a drain tank with a separate cooling system, a direct reactor auxiliary cooling system that passively rejects heat to air via a salt system, or a reactor vessel auxiliary cooling system that passively rejects heat to outside air via a water or air system.

3.4 CHAPTER 7, "INSTRUMENTATION AND CONTROL SYSTEMS"

Potential control functions focus on parameters such as temperature, flow, and level instead of conductivity and pH. Also, operating parameters for the reactor coolant are replaced by operating parameters for the reactor fuel (fuel salt) with heat removal by the primary cooling system and the heat dissipation system. Parameters for the gas management system, fuel salt cleanup system, and the fuel handling (addition) system must also be considered by the applicant.

In an MSR, reactivity can be adjusted by changing flow, manipulating the control elements in their various forms, operation of the fuel salt cleanup system, or fuel or fuel salt addition to the vessel. Therefore, discussion of controlling reactor power was revised from *inserting or withdrawing control rods* to *manipulation of the control elements, adjustment of reactor flow, manipulation of valves, etc*. Note that the routine online addition of fuel will affect reactivity and will require an I&C response. However, adding fuel is not considered an I&C function.

3.5 CHAPTER 9, "AUXILIARY SYSTEMS"

Chapter 9 notes that auxiliary systems should include homogeneous MSR fuel handling and storage of SNM used for reactor fuel (both new and irradiated), including associated components (tanks, valves, pumps, instrumentation, controls), processes (chemical blending, purification, SNM transfers, waste storage, preparation for shipment), criticality monitoring, vaults, shielding, and contamination control. A revision to the section indicates that the fuel handling discussion should address the form of the fuel during storage and handling. For facilities designed to fabricate fuel on site, the discussion should include a description of the form in which the fissile material is received, how and where it is stored before use, and how it is blended into a useable liquid fuel, including criticality control measures and monitoring.

The use of liquid fuel adds numerous new issues pertaining to quantification of the quantity of byproduct, source, and SNM that differ from issues pertaining to the use of heterogeneous fuel elements. Two new issues regarding MSRs that should be addressed by an applicant are added to Section 9.5. First, an effective means should be defined for limiting the SNM at the reactor site to establish an envelope used for safety and security analyses. Consideration should be given to the fact that fuel isotope quantities in the fuel salt will change during normal operation as ^{235}U is depleted and ^{238}U converts to ^{239}Pu or as ^{232}Th is converted to ^{233}U. Second, the mission of a non-power MSR facility should be evaluated to determine the limits on the quantity of byproduct materials to be created by experiments and routine operation.

Section 9.6, "Gas Management System," was revised to focus on control of the cover gas and any auxiliary cooling system for the gas management system. The revised text indicates that an applicant should describe gas management auxiliary systems that cool, circulate, decontaminate, recover, store, monitor, and dispose of the cover gas. Section 4.7 provides the parameters for the applicant to describe the design of the gas management system for removing fission product gases from the MSR core and cover gas.

Section 9.7, "Cooling Systems," was added for the applicant to coordinate the necessary auxiliary cooling systems with the system discussions required by Chapter 5.

3.6 CHAPTER 11, "RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT"

Chapter 11 notes that waste management is significantly different for an MSR when compared to an LWR. In an MSR, soluble, non-soluble, and gaseous fission products are released to the liquid salt fuel solution and contained by the fuel barrier. However, in an LWR, fission products are released to the fuel rod gap space and contained by the fuel clad. MSR gaseous fission products will migrate directly to the gas space in the top of the vessel (or elsewhere) and enter the gas management system to be processed within the radionuclide barrier. The gaseous fission products may require holdup for decay or further treatment before being recycled (cover gas), released to the environment, or disposed as waste. If applicable, residue from mechanical cleanup or polishing of soluble fission products will require treatment as radioactive waste.

Part 1, Section 11.2.2, "Radioactive Waste Controls," was revised to emphasize vigilance in the use of terminology that could confuse the process of waste removal with the production of SNM. Some non-power reactor licenses do not allow the "separation of isotopes" or the "separation of byproduct materials" to enforce regulations dealing with the separation of plutonium or the enrichment of ^{233}U or ^{235}U so as to produce SNM. In the strictest sense, that could be interpreted as not allowing the removal of wastes from MSR liquid-fueled cores. Fission gases do this inherently by simply rising from the liquid. Other undesired fission products are removed from the liquid fuel by waste treatment processes with no intention of producing SNM. Therefore, waste treatment processes must be clearly defined using unambiguous terms.

Section 11.2.2 has also been revised to note that there is potential for criticality concerns in the waste treatment process for accident analysis and normal operation. If a criticality concern does exist, then the applicable provisions of 10 CFR Part 70.24 should be met.

Section 11.3, "Respiratory Protection Program," was added as part of NUREG-1537 ISG, 2012. It continues to be applicable to MSR facilities and is included as part of the recommended NUREG-1537 adaptation for MSRs.

In Part 2, Section 11.2.1, "Radioactive Waste Management Program," the reviewer is directed to ensure that the applicant has described the waste management program in a manner showing that processes effectively remove undesired materials from the liquid fuel without providing a means for fissile material separation and collection in the process.

Careful attention must be directed to the waste management process so that there is no indication that a non-power MSR is a production facility. Strong discussion should reinforce the understanding that no facility exists to separate actinides.

3.7 CHAPTER 12, "CONDUCT OF OPERATIONS"

The technology-neutral material is generally applicable to MSR technologies.

In Section 12.10, "Reactor Operator Training and Requalification," the requalification requirements are outlined in sufficient detail. A paragraph has been added to this section to include discussion of initial reactor operator or senior reactor operator qualification.

The startup plan detailed in Section 12.11 was revised to acknowledge loading liquid fuel into the reactor and any subsequent startup with the fuel in solid or liquid form. The section also notes that operation with SNM is subject to the requirements of 10 CFR 70.

3.8 CHAPTER 13, "ACCIDENT ANALYSES"

Overall, the accident analysis methodology is applicable to MSRs. The applicable dose requirements were updated to remove distinctions indicating whether requirements were established before or after January 1, 1994. Also, the information to be supplied in the SAR has been referenced to 10 CFR 50.34.

MSRs utilizing liquid fuel produce both liquid and gaseous fission products that must be contained within the facility barriers rather than within heterogeneous fuel cladding. This impacts consideration of the MHA. Limiting phenomena for MSRs include fuel salt precipitation, fission-product precipitation and plate out on fuel system boundary surfaces, fuel salt chemistry/physical properties, delayed neutron production, core voiding, and tritium production.

The list of postulated events was revised to be more applicable to MSRs, and typical parameters to be tracked during postulated events are listed. Although the MSR accident analysis guidance in Chapter 13 is considered by event categories, the detailed results of the May 2019 MSR licensing basis event workshop [9] were considered when defining categories. An applicant will be required to identify their own technology-specific scenarios as part of their SAR. An MSR MHA need not include a Reactor Core scenario, because many non-core MSR subsystems also contain highly radioactive materials. An MSR-specific discussion of each of the listed postulated events is provided. The limiting event discussion was revised to include typical MSR parameters and terminology.

The list of references was updated to reflect homogenously fueled reactors.

3.9 CHAPTER 14, "TECHNICAL SPECIFICATIONS"

Minor editorial changes were applied to the TS overview in the main body of Chapter 14. Subsequently, the definitions in Appendix 14.1 have been revised to address specific aspects of MSR technology. The definition of *core configuration* was revised for MSR terminology. The definition of *control rod* was replaced with a definition for *control elements* that includes many options for adjusting MSR reactivity. Definitions for *Coating/Cladding* and *functional containment* were added. Definitions for *rod regulating* and *rod transient* were removed.

Discussion of heterogeneous fuel type safety limits were eliminated in Section 2.1 and were replaced with a discussion of MSR safety limits. Discussion of important MSR process variables was provided in Section 2.1.1. References to TRIGA reactors and pulsing operation were removed throughout the appendix.

The reactor core parameter overview in Section 3.1 was revised to reflect the processes and parameters associated with MSR operation. This includes additional discussion on core geometry and configurations. A new subsection was added to discuss monitoring the fuel composition changes that occur over time in a homogenous core with fuel burnup.

The coolant system discussion in Section 3.3 was revised to reflect the process for heat removal from the reactor fuel to the environment. This includes heat transfer to a primary cooling system. The discussion on coolant chemistry requirements was revised to reflect the use of a coolant salt instead of water.

The discussions of containment or confinement in Sections 3.4 and 4.4 were revised to reflect the low operating pressure of MSRs and the barrier approach to fission product release.

The discussion of effluents in Section 3.7.2 was revised to eliminate the specific discussion of ^{41}Ar, which does not need to be called out explicitly, because a broader spectrum of fission products must be considered based on the nature of MSR operation with fission products in the fuel salt.

The original list of references, which included an extensive list of references for heterogeneously fueled reactors, was updated to include references applicable to homogenously fueled reactors.

4 SUMMARY

This report is intended to propose an MSR technology guidance alternative for NUREG-1537. The proposed text is based on the current version of NUREG-1537 as amended by the AHR ISG. Generic MSR technology concepts and terms are used as the basis and are not intended to identify, exclude, or limit any specific MSR technology under development.

The proposed non-power MSR application format and content guide is in Appendix A of this report. The proposed non-power MSR standard review plan is in Appendix B of this report.

5 REFERENCES

1. NRC, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants; Light Water Reactor (LWR) Edition*, NUREG-0800, March 2007 (ML052340514).
2. NRC, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*, NUREG-1537, Part 1, February 1996 (ML042430055).
3. NRC, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria*, NUREG-1537, Part 2, February 1996 (ML042430048).
4. NRC, "Final Interim Staff Guidance Augmenting NUREG-1537, Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 17, 2012 (ML12156A069).
5. NRC, "Final Interim Staff Guidance Augmenting NUREG-1537, Part 2, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria*, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 17, 2012 (ML12156A075).
6. ORNL, *Proposed Guidance for Preparing and Reviewing Molten Salt Non-Power Reactor License Applications (NUREG-1537)*, ORNL/TM-2018/834, May 2018.
7. NRC, "Final Rulemaking: Non-Power Production and Utilization Facility License Renewal (RIN 3150-A196, NRC-2011-0087)," SECY-19-0062, June 17, 2019 (ML18031A001)"
8. IAEA, Research Reactors: Purpose and Future, 2016.
9. ORNL, *Molten Salt Reactor Initiating Event and Licensing Basis Event Workshop Summary*, ORNL/TM-2019/1246, July 2019.

APPENDIX A. PART 1, *GUIDELINES FOR PREPARING AND REVIEWING APPLICATIONS FOR THE LICENSING OF NON-POWER MSRS: FORMAT AND CONTENT*

ABSTRACT

Part 1 of this guide provides guidance to non-power liquid-fueled molten salt reactor (MSR) licensees and applicants on the format and content of applications to the Nuclear Regulatory Commission for licensing actions. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, decommissioning, and license termination.

INTRODUCTION

BACKGROUND

This document describes acceptable format and content of the safety analysis report (SAR) to be submitted to the U.S. Nuclear Regulatory Commission (NRC) by an applicant or licensee of a non-power liquid fueled molten salt reactor (MSR)[2] for a new license, license renewal, or license amendment. MSRs are a class of reactors in which a molten salt performs a significant function in the active reactor core. A companion document, Part 2 of this guide gives criteria to assist NRC staff reviewers in effecting comparable, complete, and consistent reviews of licensing applications for non-power MSRs. Applicants could peruse Part 2 of this guide to gain further insight into the review process for finding non-power reactor applications acceptable.

Unless otherwise noted, the terms *MSR* and *reactor* as used in this guidance are understood to mean a liquid fueled non-power MSR. The use of terminology in this guidance is consistent with the following:

For the purpose of this guide, *Non-power production or utilization facility* (NPUF) means a production facility or a utilization facility, licensed under 10 CFR 50.21(a), 50.21(c), or 50.22, as applicable, that is not a nuclear power reactor or a production facility as defined under paragraphs (1) and (2) of the definition of *production facility* in 10 CFR 50.2.

Use of the term *NPUF* in this guide means that the guidance is applicable to both utilization facilities (e.g., MSRs) and production facilities (such as fuel cleanup systems or other facilities used for the processing of irradiated materials containing special nuclear material). If the guidance is applicable to only MSRs or only production facilities, then the term *NPUF* is not used, and instead the term *MSR* or *production facility* (or *fuel cleanup facility*, if appropriate) is used.

The term *MSR NPUF* is used when the intention is to provide guidance that is applicable to both production facilities and utilization facilities that use molten salt technology. MSR NPUF is not used to refer to a single facility, and instead MSR, production facility, or facility is used, as appropriate. If a single site contains both a production facility and a utilization facility that are interconnected or collocated in a single building, then the term *facility* could be used to refer to them as a whole.

Non-power reactor means:

(1) A testing facility; or
(2) A research reactor, which is an NPUF that is a nuclear reactor licensed under 10 CFR 50.21(c) and is not a testing facility, or
(3) A commercial or industrial reactor, which is an NPUF that is a nuclear reactor licensed under 10 CFR 50.22 and is not a testing facility.

Therefore, this guide uses the terms *non-power MSR* or *reactor* to mean that the guidance is applicable to all types of non-power reactors (i.e., testing facility, research reactor, or commercial or industrial non-power reactor) that use MSR technology. The term *MSR* should be considered synonymous with the term *non-power MSR* because this guidance document is specifically directed at non-power facilities. If the guidance intends to point out a distinction between non-power reactors using MSR technology and non-power reactors that use other technologies, then the term *non-power MSR* will be used specifically to highlight the distinction. The term *testing facility* will be used when the guidance is applicable to only

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[2] There are also salt-cooled reactor designs that propose using fixed-position, coated-particle ceramic fuel. The discussion in this guide is focused on MSRs operating with liquid fuel.

testing facilities and the term *research reactor* will be used when the guidance is applicable to only research reactors.

The class of MSRs discussed in this guidance document use liquid fuel rather than solid heterogeneous fuel, and the resulting fission products—both liquid and gaseous—must be contained within the facility barriers rather than within heterogeneous fuel cladding. The homogeneous reactor fuel (fuel salt) is contained within a fuel system boundary. The fuel system boundary consists of materials that mitigate the release of radionuclides from the reactor fuel, including volatile fission products (e.g., krypton, xenon, iodine).

MSRs can be designed to operate in a thermal spectrum or a fast spectrum depending on the active reactor core geometry, active reactor core materials, and salt selection. MSRs can act as fuel burners or fuel breeders. Heterogeneous fuel assemblies have a limited life based on the buildup of fission products within the individual fuel rods. This process requires significant excess reactivity be loaded into the active reactor core to provide for acceptable fuel cycle length. However, MSRs do not require significant excess reactivity because homogenous fuel can be added periodically. Actinides can be continuously burned, and fission product poisons can be removed in batch or continuous modes. This leads to longer fuel cycles that are limited by reactor material properties and not by the fuel.

The fluoride and chloride salts used in MSRs have high boiling points; allowing the reactor to operate at higher temperatures, which provide more efficient heat transfer to the power conversion system in power reactors. The high boiling points and low vapor pressure associated with the eutectic fuel salt mixtures allow MSRs to operate solely in a liquid phase at approximately atmospheric pressure. As a result, components such as the reactor vessel can be thinner than a comparable light water reactor (LWR) vessel. Likewise, MSRs are not subject to high pressure driven radioactive material releases following design basis accidents.

The MSR guidance provided here is based on NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." However, unlike MSRs, NUREG-1537 focuses on reactors operating with heterogeneous fuel elements. Part 1 of this guide is intended to provide the necessary SAR format and content clarification to applicants for a non-power MSR operating with homogenous fuel. The guidance herein is based on the Code of Federal Regulations, Title 10, Section 50.34 (10 CFR 50.34) which describes the information to be supplied in a SAR.

All reactors, both power and non-power, are licensed to operate as utilization facilities under Title 10 in accordance with the Atomic Energy Act of 1954, as amended (AEA or Act). The AEA was written to promote the development and use of atomic energy for peaceful purposes and to control and limit its radiological hazards to the public. These purposes are expressed in paragraph 104 of the Act for non-power reactors, which states that utilization facilities for research and development should be regulated to the minimum extent consistent with protecting the health and safety of the public and promoting the common defense and security. These concepts are promulgated in 10 CFR 50.40 and 50.41, and in other parts of Title 10 that deal with non-power reactors. The licensed thermal power levels of non-power reactors are several orders of magnitude lower than current power reactors. Therefore, the accumulated inventory of radioactive fission products in non-power reactors is proportionally less than power reactors and requires less stringent and less prescriptive measures to give equivalent protection to the health and safety of the public. In MSRs, fission products are generated and entrained in the fuel salt because there is no fuel cladding. Gaseous fission products, such as xenon and krypton bubble off continuously and are typically removed from the cover gas space through a gas management system without any significant impact on reactor operation. Soluble and insoluble fission products remain in the fuel salt. Insoluble fission products tend to plate out on reactor surfaces, while soluble fission products are typically removed from the fuel salt by chemical processing, polishing, or filtration. Thus, even though many of the regulations of Title 10 apply to both power and non-power reactors, the regulations may be implemented in a different way for each category of reactor and are intended to be consistent with protecting the health and safety of the public. Because the potential hazards may also vary widely among non-power reactors, regulations also may be implemented in a different way within the nonpower reactor category.

Sections 50.20 through 50.22 of Title 10 specify two classes of reactor licenses to be issued to applicants by the NRC: Class 104 (medical therapy and research and development facilities) and Class 103 (commercial and industrial facilities). These classes derive from definitions in the AEA.

Currently, non-power reactors are typically licensed as Class 104 facilities. However, a non-power reactor for commercial purposes could be licensed as a Class 103 facility, and Section 104c of the AEA and 10 CFR 50.22 contain criteria for judging if a non-power reactor is a Class 103 facility. A Class 104 non-power reactor can be licensed as a Class 104a facility for conducting medical therapy or as a Class 104c facility for conducting research and development (or both a and c). Most of the design, operation, and safety considerations for non-power reactors apply to both research reactors and testing facilities. All non-power MSR applicants should be guided by the format and content for licensing applications in this document. Testing facilities are subject to additional requirements, such as preparation of an environmental impact statement, mandatory conduct of a hearing related to the construction permit, and review by the Advisory Committee on Reactor Safeguards (ACRS).

The principal safety issues that differentiate testing facilities 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 are generally compared with 10 CFR 20.1001 through 20.2402 and Appendices. Occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301.

If the facility conforms to the definition of a testing facility, 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 testing facilities only.

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

The regulations in 10 CFR 2.105(c) for the initial licensing of a research reactor facility do not preclude a joint application for a construction permit and the initial operating license. If well planned, the final facility design and the final SAR descriptions, analyses, and conclusions will not differ significantly from those in the initial application, and a one-step licensing procedure can be undertaken. To initiate this process, the applicant should request both a construction permit and an operating license to be issued when construction and operating readiness are acceptable to NRC. The applicant should submit only one SAR that is complete, appropriate, and acceptable for both permits. This will enable NRC to publish a joint notice of intent in the Federal Register at the construction permit stage that includes issuance of the operating license when appropriate. The joint application and joint notice procedure streamlines the licensing process. If a final SAR documenting changes during construction is submitted, it must demonstrate that the facility design and the safety conclusions of the previous SAR documentation are unchanged.

The design information in an SAR should reflect the current state of the facility design, or the current as-built system at the time of the submittal. If certain information noted herein is not yet available because the design has not progressed sufficiently, the SAR should contain (1) the criteria and design bases used to develop the required information, (2) the concepts and alternatives under consideration, and (3) the schedule for completing the design and submitting the missing information. The SAR for a new facility should describe the current design of the facility in sufficient detail to enable the reviewers to determine whether or not the facility can be constructed and operated in accordance with applicable regulations. The licensing process conforms to the legislative requirement for minimum regulation stated in Section 104 of the AEA. A license for facility operation constitutes the legal agreement between the licensee and NRC, and both parties must adhere to it rigorously. Quite often, because of applicant choice, the licensing process leads to two or more facilities with the same type of fuel and the same intrinsic safety limits being licensed for operation at maximum power levels differing by at least a factor of 10. The resultant difference between licensed operating conditions and safety limits may vary by at least an order of magnitude. However, each facility is obligated to adhere to its own license conditions.

The regulations in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," specify environmental reviews that shall be completed as part of various licensing actions for NPUFs. Guidance for applicants and the NRC staff related to these regulations is provided in "Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12156A069) and "Interim Staff Guidance Augmenting NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," (ADAMS Accession No. ML12156A075). Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance for environmental reviews associated with MSR NPUF licensing.

Guidance for applicants and the NRC staff related to financial qualifications is provided in NUREG-1537 Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content" (ADAMS Accession No. ML042430055), NUREG-1537 Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria" (ADAMS Accession No. ML042430048), "Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," and "Interim Staff Guidance Augmenting NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors". Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance on NRC regulations related to financial qualifications that must be met as part of MSR NPUF licensing.

Guidance for applicants and the NRC staff related to decommissioning is provided in NUREG-1537 Part 1, NUREG-1537 Part 2, "Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," and "Interim Staff Guidance Augmenting NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors". Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance on NRC regulations related to decommissioning that must be met as part of MSR NPUF licensing.

DOCUMENT STRUCTURE

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

The structure of this document is summarized below. The general requirements of the safety analysis are presented along with information on the purpose, applicability, and use of this document. The chapter numbering is consistent with NUREG-1537, Part 1.

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

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

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

Chapter 4 describes the design bases and the functional characteristics of the reactor vessel, the active reactor core, and its components. In this chapter, the safety considerations and features of the reactor are discussed including enrichment, uranium loading, and chemistry of the fuel salt (including fresh and reloaded fuel composition), salt type, plutonium loading (if applicable), expected fissile density in solution at operational pressure, temperature and redox conditions, fissile material solubility, build-up of fission products and related decay daughters in the fuel salt, precipitates, control elements, neutron moderation (if applicable), nuclear design, thermal-hydraulic design, and the gas management system.

Chapter 5 lists the design bases and describes the functions of the MSR fuel system boundary, all heat transfer mechanisms from the active reactor core (Chapter 4) to the environment, heat transfer mechanisms from other inter-connected systems such as the drain tank system or the gas management system, and fuel salt makeup and purification systems. The chapter describes provisions for adequate heat removal while the reactor is operating and while it is shut down.

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

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

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

Chapter 9 lists the design bases and describes the functions of such auxiliary systems at the facility as heating, ventilation, air exhaust, air conditioning, service water, compressed air, and fuel handling and storage of special nuclear material (SNM) used for reactor fuel, both new and irradiated, including components (tanks, valves, pumps, instrumentation, controls), related cooling systems, processes (chemical blending, SNM transfers, waste storage, preparation for shipment), criticality control and monitoring, vaults, shielding, and contamination control.

Chapter 10 lists the design bases and describes the functions of experimental facilities. Non-power reactors are designed with irradiation capabilities for research, education, and technological development. This chapter discusses the characteristics of experiment and irradiation facilities on the basis of the proposed experimental programs. In addition to traditional experimental purposes, a non-power MSR may be used to gather information and data that could be useful for the purposes of licensing future prototype facilities and power reactors. Special safety features and added instrumentation for any new or unique structures, systems, and components (SSC) that may be demonstrated by the MSR should also be addressed in this chapter of the SAR.

Chapter 11 lists the design bases and describes the functions of the radiation protection and the radioactive waste management (RPRWM) programs at the facility. In an MSR, fission products are released to the liquid salt fuel solution and contained by the fuel system boundary. Noble gaseous and fission products will be gathered and processed within the fission product barrier and may require holdup for decay or further treatment before being released to the environment or disposed as waste. Residue from any cleanup and polishing of the liquid fuel will be laden with fission products and fuel-salt materials will likely require treatment as radioactive waste. This chapter also describes the control of nuclear fuel and byproduct materials produced in the reactor and utilized under the 10 CFR Part 50 reactor operating license. The description of the RPRWM program should include health physics applications, radiation protection standard operational procedures (SOPs), personnel monitoring programs for radiation exposures and environmental effluent releases, radiation dose assessments to workers and the public, and the program to maintain radiation exposures and releases as low as is reasonably achievable (ALARA) from routine reactor operations and/or from experimental programs.

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

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

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

Chapter 16 is reserved to discuss other license considerations that may be specific to MSRs. Issues not discussed elsewhere in the SAR can be included in this chapter.

GENERAL REQUIREMENTS

Section 50.34 of Title 10 of the Code of Federal Regulations requires each applicant for a license to submit an SAR in the application. The application must be prepared and submitted in accordance with the following regulations:

- 10 CFR 2.101, Filing of Application
- 10 CFR 50.4, Written Communication

The content of an application must be in accordance with the following regulations:

- 10 CFR 50.33, Contents of applications; general information
- 10 CFR 50.34, Contents of applications; technical information
- 10 CFR 50.34(a), Preliminary safety analysis report
- 10 CFR 50.34(b), Final safety analysis report
- 10 CFR 50.34(c), Physical security plan
- 10 CFR 50.34(d), Safeguards contingency plan
- 10 CFR 50.34(e), Protection against unauthorized disclosure

The SAR performs the following important functions:

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

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

The SAR is the basic document that gives the NRC justification for licensing the facility. It gives information for understanding the design bases for the 10 CFR 50.59 change process, for training reactor operators, for preparing reactor operator licensing examinations, and for preparing for NRC inspections. For these reasons, and for others, it is important that the SAR remain an accurate, current description of the facility. Even though regulations do not require the licensee for a non-power reactor to periodically update the SAR (as is required in 10 CFR 50.71 (e) for power reactors), the NRC staff encourages non-power reactor licensees to maintain current SARs on file at NRC after initial licensing or license renewal by submitting replacement pages along with applications for license amendment and along with the annual report that summarizes changes made without prior NRC approval under 10 CFR 50.59.

An applicant can jointly apply for a construction permit and the initial operating license by submitting a complete and well-prepared SAR, in addition to other necessary documents, that demonstrates the licensee and facility will meet all of the regulatory requirements applicable to both a construction permit and an operating license. Otherwise, per the regulations cited above, each application for a construction permit must contain a preliminary safety analysis report (PSAR) and each application for an operating license must contain the final safety analysis report (FSAR), physical security plan, safeguards contingency plan, and plan for protection against unauthorized disclosure. Applicants should segregate documents that are subject to frequent change with forethought to the ease of maintaining updated documents. For example, the quality assurance program description required in the PSAR may be identified by reference in the PSAR as an appendix or as an independent part of the application. The same may be done with the FSAR requirements for an emergency plan, technical specifications, and the operator requalification program.

This guide provides general guidance for the format and content of a complete SAR (or FSAR). The applicant should use the same format and chapter headings for the PSAR. However, the PSAR content pursuant to 10 CFR 50.34(a) is less detailed and sub-chapter headings should be modified as appropriate to match its focus on the criteria and standards used for the design and analysis rather than that of a detailed description and safety analysis of the completed operational facility.

It is recognized that the guidance addresses topics that may not apply to all applications and therefore those topics need not be discussed. Some technologies and applications may involve terminology that is not defined and used in the guidance documents, in which case the more appropriate terminology should be defined and used.

PURPOSE OF THE FORMAT AND CONTENT GUIDE

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

APPLICABILITY OF THE FORMAT AND CONTENT GUIDE

The NRC staff recommends this guide for license applications for new non-power reactors. This document also gives guidance to licensees preparing SARs for other licensing actions, such as license amendments. This guide should help licensees prepare complete packages and, thus, should reduce potential delays caused by NRC requests for additional information. Applications for license amendments should be written in accordance with applicable sections of the guide; however, a complete revision of the existing SAR should not be required in support of such an application. For license amendment requests, the corresponding sections of the SAR should be amended and submitted to NRC along with the amendment application.

Not every suggestion given here will be applicable to every non-power MSR. As applicants consult this document, they will identify guidance that they believe is not applicable to their particular reactor design. Applicants should carefully consider what guidance is applicable to their reactor design. The applicant does not need to discuss the reasoning for not providing all of the information suggested in this document. However, the applicant should be able to justify such deletion upon request of the NRC reviewer.

USE OF THIS GUIDE

Although applicants are not required to consult this guide in preparing the application for a license or license amendment, the NRC staff strongly encourages its use because all applications will be reviewed and evaluated on the basis of their technical content and completeness. Upon receiving an application, the

NRC staff will review and evaluate the SAR against Part 2 of this guide to determine if the SAR contains the information necessary to form the bases for the staff findings required for the issuance of an operating license or granting of a license amendment.

Applicants should:

- Submit the application in accordance with 10 CFR 50.30, 10 CFR 50.4, and other applicable regulations. The regulations in 10 CFR Part 170, "Fees for Facilities, Materials, Import and Export Licenses, and other Regulatory Services Under the Atomic Energy Act of 1954, as Amended," contains information on licensing and amendment fees.
- Use the electronic submission guidance for new reactor-related application submittals. This guidance is found in the electronic submission guidance referenced in 10 CFR 50.4(a) and available on the NRC public website at http://www.nrc.gov/site-help/e-submittals/guide-electronic-sub.pdf.

REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21, "Format and Content for Safety Analysis Reports for Research Reactors," 2012 (R2018).

U.S. Nuclear Regulatory Commission, "Guidance for Electronic Submissions to the NRC," Revision 8, May 2017.

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

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

U.S. Nuclear Regulatory Commission, Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 2012.

U.S. Nuclear Regulatory Commission, Interim Staff Guidance Augmenting NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 2012.

MSR DEFINITIONS

The following glossary contains terms often used when discussing an MSR.

Where terminology in this guidance does not properly characterize new technology, vendors and applicants should introduce appropriate substitute terminology and provide definitions.

Active Reactor Core: In an MSR, the vessel region occupied by the fuel salt, where the majority of prompt neutrons are generated and where most fissions occur. In an MSR, the active reactor core geometry might change with time as a result of changes in density and voiding of the solution.

Coating or Cladding: Intervening protective layer of material between the fuel salt and the structural container alloy. Also included are surface modifications of the structural alloy to enhance its chemical or mechanical performance by altering its microstructure or composition (e.g., carbiding, phosphiding, or nitriding the surface).

Control Element(s): Object(s) employed to adjust reactivity. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned.

Emergency Cooling System: A system that provides decay heat removal from the reactor fuel following an accident (e.g., a direct reactor auxiliary cooling system or a reactor vessel auxiliary cooling system). Similar cooling systems for fuel drain tanks are also included, as are systems such as in-floor heat pipes, to provide cooling to fuel located in outer containment layers in the event of fuel system breach type accidents.

Fuel Barrier: Portion of the fuel system boundary in contact with the liquid fuel after addition to the fuel circuit and prior to transfer to waste handling (principally the vessel, chemical processing system boundary, drain tank [if used], heat exchanger, cooling thimbles, control element thimbles, instrumentation thimbles, piping, tanks, and valves).

Fuel System Boundary: The material that mitigates the release of radionuclides from the reactor fuel, including volatile fission products (e.g., krypton, xenon, iodine). For an MSR, this includes the Vessel, drain tank (if used), cooling thimbles, heat exchangers, chemical processing system boundary, waste-handling tank, pumps, valves, and piping—it essentially includes the radionuclide barrier and the fuel barrier.

Gas Management System: The cover gas system provided to capture volatile fission products (e.g., krypton, xenon, iodine) until ultimate discharge and to provide venting of any pressure/density transients that could result in damage to the Vessel or the fuel salt/ primary cooling system salt heat exchanger, thus resulting in loss of the fuel system boundary.

Heat Dissipation System: A set of components or system(s) that interface with the primary cooling system to provide the principal means of transferring the heat from the active reactor core to an ultimate heat sink. The heat dissipation system might use a variety of coolants (e.g., salt, liquid metal, gas, or water) but does not contain fuel.

Neutron Moderator: In an MSR, materials in or near the active reactor core that consist of light elements (e.g., H, B, C). Moderators are generally solid form.

Primary Cooling System: The system that directly interfaces with the fuel system boundary at the fuel salt/primary cooling system salt heat exchanger(s) to provide the principal means of removing heat from the fuel salt during operation by transferring the heat to the heat dissipation system. The primary cooling system may employ a variety of coolants (e.g., nitrate salt, halide salt, etc.), but it does not contain fuel.

Radionuclide Barrier: The portion of the fuel system boundary that serves as the innermost low-leakage barrier to the radionuclides within the gas management system.

Reactor Fuel: In an MSR, fuel salt consisting of fissionable and possibly fertile halide salts, fission products, and generally solvent halide salt(s).

Vessel: For an MSR, the structure containing the active reactor core. In certain design configurations, other components such as heat exchangers might reside in the vessel but outside the active reactor core.

1 THE FACILITY

Chapter 1 of this guide is applicable to providing a description of the facility for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should present an introduction to the SAR and the facility. The introduction should state the purpose of the SAR and briefly describe the MSR application.

Chapter 1 should contain the following topics:

- Introduction
- Summary and conclusions of principal safety considerations
- General description of the facility
- Shared facilities and equipment
- Comparison with similar facilities
- Summary of operations
- Compliance with the *Nuclear Waste Policy Act* of 1982
- Facility modifications and history

1.1 INTRODUCTION

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

1.2 SUMMARY AND CONCLUSIONS ON PRINCIPAL SAFETY CONSIDERATIONS

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

- potential radiological consequences from the operation and use of the non-power MSR, and the methods used to ensure the safety of the reactor;
- safety considerations that influenced the selection of the facility site, the type of reactor and liquid fuel, the reactor thermal power level, the type of building housing the reactor, and any special factors;
- any inherent or passive safety features designed to contribute to facility safety, protection of the health and safety of the public and staff, and protection of the environment;
- design features and design bases for any systems and components that promote safe operation and shutdown of the facility; and
- potential accidents at the facility, including the maximum hypothetical accident, and any design features that prevent accidents or mitigate the potential consequences.

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

1.3 GENERAL DESCRIPTION OF THE FACILITY

The applicant should briefly describe the reactor facility as follows:

(1) geographical location; (2) principal characteristics of the site; (3) principal design criteria, operating characteristics, and safety systems; (4) any engineered safety features; (5) instrumentation, control, and

electrical systems; (6) heat removal and other auxiliary systems; (7) radioactive waste management provisions (or system) and radiation protection; and (8) experimental facilities and capabilities. The general arrangement of major structures and equipment should be indicated with plan and elevation drawings. Safety features of the facility that are likely to be of special interest should be briefly identified. Such items as unusual site characteristics, the containment building, novel designs of the reactor, or unique experimental facilities should be highlighted. The information and discussions in this section in no way should substitute for the complete discussion and analysis found in (and referenced to) subsequent chapters of the SAR.

1.4 SHARED FACILITIES AND EQUIPMENT

The applicant should briefly describe the following:

- Systems and equipment that are shared with facilities not covered by the SAR or the operating license. Examples of shared facilities and equipment could be water purification systems; electrical supplies; heating, ventilation, and air conditioning systems; and the building that houses the reactor room.
- Any other reactor, subcritical assembly, irradiation facilities, fuel storage systems, fission product cleanup systems, or hot cell located within the confinement or containment structures, or the restricted areas to which this SAR supplies.
- Any safety barriers and any special isolation provisions for the shared facilities and equipment.

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

1.5 COMPARISON WITH SIMILAR FACILITIES

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

1.6 SUMMARY OF OPERATIONS

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

1.7 COMPLIANCE WITH THE NUCLEAR WASTE POLICY ACT OF 1982

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

2 SITE CHARACTERISTICS

Chapter 2 of this guide is applicable to providing a description of the site characteristics for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should discuss and describe the geographical, geological, seismological, hydrological, and meteorological characteristics of the site and vicinity in conjunction with present and projected population distributions, industrial facilities and land use, and site activities and controls. In this presentation, the applicant should define the site characteristics for use in design and analyses discussions in other chapters of the SAR, e.g., Chapter 3, “Design of Structures, Systems and Components”; Chapter 11, “Radiation Protection Program and Waste Management”; and Chapter 13, “Accident Analyses.” The regulations in 10 CFR 100.10 specify factors to consider in selecting sites for testing facilities, while there are no regulations that specify factors to consider for siting research reactors. However, IAEA-TECDOC-403, “Siting of Research Reactors,” provides guidance for siting research reactors.

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

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

2.1.1.2 Boundary and Zone Area Maps

Maps of the site area of suitable scale (with explanatory text as necessary) should be included to show the boundaries and zones associated with the facility. Boundaries and zones are defined in the documents in the reference section related to emergency preparedness at non-power reactors (NRC Regulatory Guide 2.6, NRC NUREG-0849, ANSI/ANS 15.16). The maps should clearly show the following:

- the general area in which the reactor will be located and sufficient secondary detailed maps to show the location of the reactor facility and adjacent surroundings;
- location of the area directly under the NRC facility operating license;
- location of the operations boundary;
- location of the site boundary;
- population density out to a distance of 5 miles (8 kilometers) from the reactor;
- location of emergency preparedness zones (EPZs), as applicable;
- true north;
- highways, railways, and waterways that traverse or are in close proximity to the site; and
- the general topography of the area near the reactor that could affect diffusion and dispersion of airborne effluents, including building at least as tall as the reactor building and any stacks or other air-exhaust facilities.

2.1.2 Population Distribution

Population data presented should be based on the most recently available (last decade or later) census data. Information on population distributions should be in suitable form to use in dose analyses in

Chapters 11 and 13, in which potential doses down to a small percentage of 10 CFR Parts 20 or 100 may be applicable.

On a map of suitable scale that identifies places of significant population grouping (such as cities and towns) within a 5-mile radius, concentric circles should be drawn, with the reactor at the center point, at distances of 0.5, 1, 2, 3, and 5 miles. The population in each area at the time of application and a projection of the population in five years and at the end of the license period should be given. The basis for population projections should be described. Information should be given about the direction and distance of the nearest permanent residence to the reactor and any reactor effluent exhaust points. Any part-time, transient, or seasonal occupation of buildings should be described, such as classrooms or dormitories on a university campus, giving best estimates of occupation times and numbers of occupants.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

In this section, the applicant should establish whether the effects of potential accidents in the vicinity of the reactor from present and projected industrial, transportation, and military installations and operations should be used in the safety analyses and should establish the reactor facility design parameters related to accidents selected. The applicant should consider all facilities and activities within 5 miles of the reactor. Facilities and activities at greater distances should be included as appropriate to their significance of accident impact on the facility.

2.2.1 Locations and Routes

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

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

2.2.2 Air Traffic

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

- sites located within 5 miles of an existing or projected commercial or military airport
- sites located between 5 and 10 miles from an existing or projected commercial or military airport with more than approximately 500 d^2 (where d is the distance in miles from the airport to the reactor site) commercial or military aircraft movements per year.

Special consideration should be given when siting the facility within the trajectory of a runway of any airport. The analysis should demonstrate that there is a low potential that any aircraft, including general aviation aircraft, could affect the reactor or that the consequences from any aircraft-associated accident are already bounded or considered in the accident analysis.

2.2.3 Analysis of Potential Accidents at Facilities

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

2.3 METEOROLOGY

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

2.3.1 General and Local Climate

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

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

2.3.2 Site Meteorology

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

2.4 HYDROLOGY

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

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

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

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

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

2.5.1 Regional Geology

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

2.5.2 Site Geology

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

2.5.3 Seismicity

The applicant should list all historically reported earthquakes that could have reasonably affected the region surrounding the site. The list should include all earthquakes of modified Mercalli intensity greater than IV or magnitude (Richter) greater than 3.0 that have been reported in all tectonic provinces, any part of which is within 125 miles of the site.

2.5.4 Maximum Earthquake Potential

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

2.5.5 Vibratory Ground Motion

The applicant should proceed from discussions of the regional seismicity, geologic structures, and tectonic activity to a determination of the relation between seismicity and geologic structures. The earthquake-generating potential of tectonic provinces and any active structures should be identified. Finally, the applicant should assess the ground motion at the site from the maximum potential earthquakes associated with each tectonic province or geologic structure and should consider any site-amplification effects. Using the results, the applicant should establish the vibratory ground motion design spectrum.

2.5.6 Surface Faulting

The applicant should discuss any potential for surface faulting at the site and should list all historically reported earthquakes that can be reasonably associated with faults, any part of which is within 5 miles of the site.

2.5.7 Liquefaction Potential

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

2.6 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, “Emergency Planning for Research Reactors,” 2015.

International Atomic Energy Agency, IAEA-TECDOC-1347, “Consideration of External Events in the Design of Nuclear Facilities Other Than Nuclear Power Plants, with Emphasis on Earthquakes,” 2003.

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,” Revision 2, 2017.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Chapter 3 of this guide is applicable to providing a description of the design of structures, systems, and components (SSC) for a non-power MSR. In this chapter of the SAR, the applicant should identify and describe the principal architectural and engineering design criteria for the SSC that are required to ensure MSR safety and protection of the public. The material presented should emphasize the safety and protective functions and related design features that help provide defense in depth against uncontrolled release of radioactive material. The bases for the design criteria for some of the systems discussed in this chapter may be developed in other chapters and should be appropriately cross referenced. For example, confinement or containment design criteria may be summarized here and discussed in detail in Chapter 6, "Engineered Safety Features."

As stated above, facility and system design must be based on defense-in-depth practices. Defense-in-depth practices means a design philosophy, applied from the outset and through completion of the design, that is based on providing successive levels of protection such that health and safety will not be wholly dependent upon any single element of the design, construction, maintenance, or operation of the facility. The net effect of incorporating defense-in-depth practices is a conservatively designed facility and system that will exhibit greater tolerance to failures and external challenges. The risk insights obtained through performance of accident analysis can then be used to supplement the final design by focusing attention on the prevention and mitigation of the higher risk potential accidents.

The design must incorporate both of the following, to the extent practicable:

(1) Preference for the selection of engineered controls over administrative controls to increase overall system reliability
(2) Features that enhance safety by reducing challenges to engineered safety features or administrative controls.

3.1 DESIGN CRITERIA

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

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

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

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

- Design to cope with anticipated transients and potential accidents, including those discussed in Chapter 13, “Accident Analyses,” of the SAR. Anticipated transients and potential accidents should include malfunction of any control function or other malfunction of a SSC, experiment malfunction, single operator error, testing and surveillance activity, reactor startup and shutdown, and power-level change. These design criteria should be based on a systematic examination of the most limiting transients and accidents to identify the needed facility SSC. They should ensure that each needed SSC stays within acceptable operational and safety limits for conservative assumptions of initial conditions, operating history of the facility for the proposed license term and required operating characteristics. The most limiting conditions of each type should be analyzed in detail in Chapter 13 of the SAR.
- Design redundancy for reactor protective and safety features, so that any single failure of any active component will not prevent safe reactor shutdown or result in unsafe conditions as verified by Chapter 13 analyses.
- Design to facilitate inspection, testing, and maintenance of the SSC whose integrity and reliability are important to safe reactor shutdown and to the protection of the public, reactor facility personnel, and environment.
- Provisions to avoid or mitigate the consequences of fires, explosions, and other potential manmade or natural conditions.
- Quality standards commensurate with the safety function and potential risks. For example, fuel fabrication may be consistent with the applicable guidance in ANSI/ANS 15.2.
- Analyses and designs for meteorological, hydrological, and seismic effects.
- The bases for technical specifications necessary to ensure the availability and operability of required SSC.

3.2 METEOROLOGICAL DAMAGE

In this section the applicant should describe the design for the protection from meteorological conditions of facility structures (e.g., buildings and barriers), systems (e.g., cooling and ventilation systems), and components that are assumed to be operable in the SAR and included in technical specifications. The design criteria should be based on data given in Chapter 2, “Site Characteristics,” on such factors as historical data on maximum wind velocity, vertical velocity profiles, gust factors, applied loads, recurrence intervals, tornado loadings, and snow and ice loads. The applicant may refer to local building codes, standards, or other criteria to ensure that significant meteorological damage to the facility is very unlikely. Further, the design criteria should provide reasonable assurance that potential meteorological damage would not significantly affect designed SSC (i.e., they would continue to perform necessary operational and safety functions). An example would be consideration of adverse wind conditions that affect ventilation systems. The bases for appropriate technical specification surveillances to verify capability and reliability of the design features should be given.

3.3 WATER DAMAGE

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

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

3.4 SEISMIC DAMAGE

In this section the applicant should specify and describe the SSC that are required to maintain the necessary safety function if a seismic event should occur, as well as the required facility seismic design criteria. The seismic characteristics of the site should be summarized in Chapter 2. Seismic design for non-power reactors should, at a minimum, be consistent with local building codes and other applicable standards. However, any local building codes and standards used for seismic design must ensure operability of safety related SSCs during and following a potential seismic event such that dose limits are not exceeded. Otherwise, more stringent codes and standards will be necessary. For NRC-licensed non-power reactors, the seismic event considered in the analyses 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 seismic guidance. In addition, IAEA-TECDOC-1347 contains guidance on the seismic design of SSC.

The reactor facility seismic design should provide reasonable assurance that the reactor could be shut down and maintained in a safe condition or that the consequences of accidents would be within the acceptable limits. For most NRC-licensed non-power reactors, this may involve analysis to show that the conditions of the SAR for safe reactor shutdown remain valid for the potential seismic events (e.g., fuel system boundary is not breached and MSR shutdown capability is not impaired). The applicant should include detail on the scope and complexity of seismic models used to determine potential seismic damage to SSCs. Consideration should be given to movement of the liquid fuel and to any shutdown mechanisms that are susceptible to changes in shape or dimensions. The applicant can also show that the radiological consequences of a potential seismic event are bounded by the accident analyses in Chapter 13.

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

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

3.5 SYSTEMS AND COMPONENTS

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

3.6 REFERENCES

American National Standards Institute, ANSI N323, “Radiation Protection Instrumentation Test and Calibration,” ANS, LaGrange Park, Illinois, 1978 (R1996).

American National Standards Institute, ANSI N323c, “Radiation Protection Instrumentation Test and Calibration – Air Monitoring Instruments,” ANS, LaGrange Park, Illinois, 2009.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.2, “Quality Control for Plate-Type Uranium-Aluminum Fuel Elements,” ANS, LaGrange Park, Illinois, 1999 (R2016).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, “Quality Assurance Program Requirements for Research Reactors,” ANS, LaGrange Park, Illinois, 1995 (R2018).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, “Radiological Protection at Research Reactor Facilities,” ANS, LaGrange Park, Illinois, 2016.

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 (R1986). (withdrawn)

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

International Atomic Energy Agency, IAEA-TECDOC-1347, “Consideration of External Events in the Design of Nuclear Facilities Other Than Nuclear Power Plants with Emphasis on Earthquakes,” Vienna, Austria, 2003.

International Atomic Energy Agency, IAEA-TECDOC-403, “Sitting of Research Reactors,” Vienna, Austria, 1987.

4 MOLTEN SALT REACTOR DESCRIPTION

Chapter 4 of this guide is applicable to providing a description of the design and functional characteristics of the reactor for the licensing of a nonpower MSR. In this chapter of the SAR, the applicant should discuss and describe the principal features, operating characteristics, and parameters of the MSR. The analysis in this chapter should support the conclusion that the reactor design provides reasonable assurance of adequate protection of public health and safety during operation and shutdown under all credible conditions. Information in this chapter of the SAR should provide the design bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many technical specifications (TS).

4.1 SUMMARY DESCRIPTION

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

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

- Thermal power level
- Fuel type and enrichment
- Vessel
- Forced and/or natural-convection cooling
- Type of fuel salt, moderator (if any), and reflector
- Principal features for experimental programs (if any)
- Novel concepts requiring substantial new development

4.2 ACTIVE REACTOR CORE

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

4.2.1 Reactor Fuel

In this section, the applicant should describe the reactor fuel and the fuel system boundary. Included should be the design features selected to ensure that the fuel system boundary can withstand all credible environmental and irradiation conditions during its life cycle at the reactor site. The discussions should address the vessel fuel operating conditions. Fuel and fuel salt support systems and operations such as receipt from off-site, storage, melting, solidification, dissolution, blending, fuel makeup and removal, fission heat extraction, fuel cleanup, and draining of the system should be discussed in Chapter 5, “Molten Salt Reactor Cooling Systems,” or in Chapter 9, “Auxiliary Systems,” of the SAR. Drawings and tables of design specifications and operating characteristics of the fuel should be presented.

In MSRs, fission product gases build up within the vessel, or at a free surface provided above the liquid fuel. Gases generated during fission also collect in the cover gas space. Therefore, information relevant to the head space and gas management systems should be provided.

Information should be current; supported by referenced tests, measurements, and operating experience; and compared with additional applicant experience where applicable. The information should include the following:

- Chemical composition, enrichment, fissile material loading, and chemistry of the fuel salt. Information should be provided for fresh and reloaded fuel composition, salt type, plutonium loading (if applicable), expected fissile density in solution at operational pressure, temperature and redox conditions, fissile material solubility, buildup of fission products and related decay daughters in the fuel salt, precipitates, and sparging and sweep gas system.
- Information on fission gas formation and impact on active reactor core chemistry, homogeneity, and reactivity. Information on void formation collapse on reactor performance should be discussed.
- Short-term changes in the chemistry of the fuel, such as changes in redox, temperature fluctuations, and fission gas release. The range of these fluctuations, and their effects on reactor operation and controls should be described.
- Long-term changes in the chemistry of the fuel. In particular, buildup of fission products, activation products, and corrosion products would be of interest. Any plans for stabilizing or adjusting fuel characteristics or composition should be included. Any plans regarding periodic reconstitution or purification of the fuel should also be included. Any scheduled periodic analysis plans for the fuel should be described. Finally, a description of the fuel at the end of life should be given.
- Description of the volume occupied by the fuel solution, including the height and diameter and portion of the volume occupied by solids. Separate descriptions should be given for conditions with and without significant power and fission gas generation. Special features such as moderators (if applicable) reflectors, external geometrical designs to enhance cooling capability, and inherent safety or feedback provisions should be discussed.
- Physical properties significant to safety that are important for the thermal-hydraulic analyses, such as fuel salt density, power density and distribution (composition change over time), temperature, pressure, heat capacity, viscosity, gas evolution or diffusion (including fission product gas), changes in void fraction, precipitation of fuel or fission product complexes, and sparging or sweep gas.
- Material and structural information for the vessel and piping (including the fuel salt/primary cooling system salt heat exchanger) that relate to the integrity of the fuel system boundary, such as dimensions, fabrication methods, compatibility of materials, irradiation effects, temperatures, and specifications with tolerances
- Descriptions of all types of fuel salt chemical constituents used should be described, as well as the fuel preparation method and location.
- Information on material parameters that could affect the integrity of the vessel, the fuel salt/primary cooling system salt heat exchanger, control element channels, and fuel transport piping, such as melting, softening, or blistering temperatures; corrosion; erosion; and mechanical factors, such as swelling, bending, twisting, compression, and shearing.
- A brief history of the fuel type, with references to the fuel development program, including summaries of performance tests, qualification, and operating history
- Hydraulic forces, thermal changes and temperature gradients, internal pressures including that from fission products and gas evolution (including removal to gas treatment), pressure, precipitation, malfunctions of the gas management system, and radiation effects on the solution chemistry. Extended and more detailed discussion of these characteristics and effects may be included in Section 4.7, “Gas Management System” (addressed below).
- Adequate mixing of the fuel solution based on convection and gas evolution

- Features that ensure accurate and secure positioning and adequate coolant flow for any experimental facilities should be described.

Information and analyses should support the limits on operating conditions for the fuel. These limits are specified to ensure that the integrity of the fuel system boundary will not be impaired by loss of redox control, fission gas evolution, power oscillations, precipitation from fuel salt, temperature and pressure extremes or distributions, and materials compatibility. They should form the design bases for this and other chapters of the SAR, the reactor safety limits (SLs), and other fuel-related TS.

Information and analyses should support the limits on operating conditions for the fuel. These limits are specified to ensure that the integrity of the fuel system boundary will not be impaired by loss of redox control, fission gas evolution, power oscillations, precipitation from fuel salt, temperature and pressure extremes or distributions, corrosion, and materials compatibility. They should form the design bases for this and other chapters of the SAR, the reactor safety limits (SLs), and other fuel-related TS.

4.2.2 Control Elements

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

- The number and types of control elements (e.g., material, phase, and the intended function of the control element), their designed locations in and around the active reactor core, whether they are positioned passively or actively, and their designed reactivity worths. The considerations and bases for redundancy and diversity should be provided. Limits on the active reactor core configuration should be discussed.
- The structural and geometric description, including control element withdrawal rate limits, shape, size, materials, cladding, fabrication methods, and specifications with tolerances for the control elements. This should include the type and concentration of neutron absorber, or emitter, if applicable. Also, calculations of changes in reactivity worth due to burnup and assessment of radiation damage, heating effects, and chemical compatibility with the fuel salt and other active reactor core components should be given. If the control elements have followers, the design, composition, and reactivity effects of the follower should be discussed.
- The structural and mechanical design relative to the vessel penetrations provided for the control elements. Information should be included on whether the penetrations are closed or open-ended thimbles or tubes, whether the control elements require cooling during operation at power, and how the thermal-hydraulic design keeps the reactor within the specified operational and safety limits. Cooling calculations may be included in Chapter 5, "Molten Salt Reactor Cooling Systems."
- The design of mechanical supports for the active component, the method of indicating and ensuring reproducible positioning in the active reactor core, and the manipulation mechanism of each type of control element. This information should include the source of motive power, usually electrical, and the systems ensuring scram capability.
- The kinetic behaviors of the control elements, showing either the positive or negative rate of reactivity change, in the normal manipulation and scram modes of operation. This information should be supplied for all control elements.
- Evidence that the control elements design conforms to the shutdown margin requirements.

- Summarize the scram logic and circuitry, interlocks and inhibits on control element manipulation, trip release and insertion times, and trip or scram initiation systems. They should also be described in detail in Chapter 7, "Instrumentation and Control Systems."
- Special features of the control elements, their locations relative to the vessel and the active reactor core, power sources, manipulation mechanisms designed to ensure operability and capability to provide safe reactor operation and shutdown under all conditions during which operation is required in the safety analysis if there is a single failure or malfunction in the control system itself. Such features may include mechanisms to limit the speed of control elements movement.
- TS requirements for the control elements and their justification. These are the limiting conditions for operation (LCOs), surveillance requirements (SRs), and design features as discussed in Chapter 14, "Technical Specifications," of this format and content guide.

4.2.3 Neutron Moderator (if applicable) and Reflector

In this section, the applicant should discuss the materials and systems designed to moderate (if applicable) the neutrons within the fuel region and reflect leakage neutrons back into the fuel region. The information should include the materials, geometries, reflector or moderator structure including designs for changes or replacement, provisions for cooling, radiation damage considerations, and provisions for experimental facilities or special uses. Multiple use systems and features such as moderator coolant, fuel moderator, and reflector shield should be described. If moderators or reflectors are encapsulated to prevent contact with fuel salt, the effect of failure of the encapsulation should be analyzed. It should be possible to operate the reactor safely until failed encapsulations are repaired or replaced. If reactor operations cannot be safely continued, the reactor should be placed and maintained in a safe condition until encapsulations are repaired or replaced. TS requirements should be proposed and justified for the moderator and reflector in accordance with the guidance in Chapter 14 of this format and content guide. The nuclear design of the moderator and reflector should be discussed in Section 4.5 of the SAR.

4.2.4 Neutron Startup Source

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

4.2.5 Core Support Structure

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

- The vessel and reflector vertical and lateral support structure, as well as the support for the reactor control and cooling components and any other components connected to the vessel. It is important to include these items because the active reactor core is fluid fuel salt and, therefore, the MSR active reactor core support structure is the vessel. The fuel-positioning function of a heterogeneous reactor core support structure is not applicable to an MSR.

- The materials of construction, including considerations for radiation damage, corrosion, erosion, chemical compatibility with fuel salt and active reactor core components, potential effects on reactivity, induced radioactivity, and maintenance.
- Design features of the support structures that accommodate other systems and components such as radiation shields, reflectors, fuel salt piping (including accommodation for hydraulic forces, buoyant and dynamic loads such as vibration), control element manipulation mechanisms, fuel salt plenums or deflectors, gas treatment systems, and nuclear detectors. Piping for fuel transfer to and from the vessel should be specifically addressed.
- Design features of the active reactor core support structure that accommodate other systems and components such as radiation shields, beam ports, or other experimental facilities.
- TS that control important design features, LCOs, and SRs, as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4.3 VESSEL

The active reactor core of the MSR is a solution of fuel salt within a gas-tight vessel. In this section, the applicant should present all information about the vessel necessary to demonstrate its integrity. The information should include the following:

- Design and considerations to ensure that no hydrodynamic, hydrostatic, mechanical, seismic, chemical, and radiation forces or stresses could cause failure or loss of integrity of the vessel during its projected lifetime over the range of design characteristics.
- Design and dimensions to ensure sufficient shielding to protect personnel and components. (Also see Sections 4.4 and 4.6 and Chapter 11, “Radiation Protection Program and Waste Management,” of this format and content guide.)
- Designs and description of materials, including dimensions, supporting structures, chemical compatibility with the fuel salt and other reactor system components, radiation fields and any consequences of radiation damage, protection from corrosion in inaccessible regions, and capability to replace components, if necessary.
- Locations of penetrations and attachment methods for other components and pipes. The relationships of these penetrations to salt surface elevations should be discussed. Safety-related features that prevent loss of fuel salt should be discussed and related to Sections 4.4 and 4.6 and to the reduction in cooling scenarios analyzed in Chapter 13, “Accident Analyses,” as applicable.
- If the inner surface of the vessel is coated to alleviate the impact of contact with the fuel, the effect of failure of the coating should be analyzed.
- Planned methods for assessing radiation damage, chemical damage, erosion, pressure pulses, or deterioration during the projected lifetime. In this section the applicant should assess the possibility of uncontrolled leakage of fuel salt into the reactor cavity and should discuss preventive and protection features.
- TS that control important design features, LCOs, and SRs as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4.4 BIOLOGICAL SHIELD

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

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

- The design details and methods used to achieve the design bases. The applicant should discuss the protection of personnel and equipment functions. The information should specify the general size and shape of the shields and the methods used to ensure structural strength, rigidity, and functional integrity. The applicant should discuss the distribution of shielding factors between liquid (water) and solid (concrete, lead, etc.) materials. If loss of shield integrity could cause a reduction in cooling, the features to prevent the loss of integrity should be described.
- The materials used and their shielding coefficients and factors, including a detailed list of constituents and their nuclear and shielding properties. The applicant should discuss radiation damage and heating or material dissociation during the projected lifetime of the reactor, induced radioactivity in structural components; potential radiation leakage or streaming at penetrations, interfaces, and other voids; shielding at experimental facilities (if applicable); and shielding for facilities that store fuel and other radioactive materials within the vessel and outside the vessel.
- The assumptions and methods used to calculate the shielding factors, including references to and justification of the methods. Detailed results of the shielding calculations should give both neutron and gamma-ray dose rates at all locations that could be occupied. The applicant should calculate shield penetrations, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.
- Methods used to prevent neutron irradiation and activation of groundwater or soils surrounding the reactor shield that could enter the unrestricted environment. The applicant should estimate the maximum activity in case such activation occurs and describe remedial actions.
- TS that control important design features, LCOs, and SRs as discussed in Chapter 14 of this standard format and content guide. The applicant should justify these TS in this section of the SAR.

Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," issued May 2009 (ADAMS Accession No. ML090820425), describes a method acceptable to the NRC staff for complying with the regulations with regard to the design and construction of concrete radiation shields in nuclear power plants. It encompasses applicable material previously endorsed in Regulatory Guide 2.1, "Shield Test Program for the Evaluation of Installed Biological Shielding in Research and Training Reactors," and can be used by applicants for non-power MSRs.

4.5 NUCLEAR DESIGN

In this section, the applicant should give information on the nuclear parameters and characteristics of the active reactor core and should analyze the kinetic behavior of the reactor for steady-state and transient operation throughout its life cycle as discussed in the safety analysis. The descriptions, analyses, and results should address all safety issues in the design and operation of the reactor and should support the conclusion that the reactor can be built and operated without unacceptable risk to the health and safety of the public. A detailed description of the analytical methods used in the nuclear design should be given. Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments. Code descriptions should include methods of obtaining parameters such as cross sections. Estimates of the accuracy of the analytical methods should be included. Tables and figures should be used as necessary to present information clearly.

4.5.1 Normal Operating Conditions

In this section, the applicant should present information on the active reactor core geometry and configurations. The limiting configurations for a reactor are the active reactor core conditions that would yield the highest power density, the highest excess reactivity, and other possible limiting parameters that are of safety interest using the fuel specified for the reactor. All other active reactor core configurations should be demonstrated to be encompassed by the safety analysis of the limiting active reactor core configuration. Sections 4.5.3 and 4.6 should give further information on power density limitations. The information in the SAR should include the following:

- Discussion and analyses to reflect the impact of the gas management system that is contained within the radionuclide barrier on the active reactor core physics. The discussion should include holdup times and subsequent release of fission product gases. Sweep gas or sparging gas operation and discharge limits, and reactivity impacts associated with these operations will need to be included in this section and in Section 4.7.
- Discussion and analyses of the reactor operating characteristics. The applicant should describe in detail the effects of changes in configuration and fuel chemistry, including effects related to redox, pressure, temperature, reactivity, and power oscillation, and the control philosophy and methodology for each parameter. If applicable, the applicant should analyze safety-related considerations for all requested operating modes.
- Changes inactive reactor core reactivity with fuel composition changes, fissile material buildup, activation products, and poisons, both fission products and those added by design. The application should also discuss reactivity impacts of fission gas and void formation, fission product gas removal, and fuel salt addition.
- Analyses of the reactor kinetic behavior and the design requirements and dynamic features of the control elements that allow controlled operation for all possible reactor conditions. This includes the expected effects of fission product gases on power oscillations resulting from formation, collapse, and movement of voids and the effects of temperature changes or gradients in the fuel salt.
- Analyses of the basic reactor criticality physics, including the interacting effects of fuel, neutron moderators (if any) and neutron reflectors, control elements, and any experimental facilities. This also includes discussion of the subcritical storage and handling of the full reactor fuel mass outside the vessel and during transport from and to the active reactor core, the reactivity swing of the processed fuel salt material after selected fission product removal, and any compensatory measures (such as fissile addition or dilution) to achieve or maintain criticality after reinsertion into the vessel.
- Discussion of the safety considerations for different active reactor core configurations, including a limiting active reactor core configuration that would yield the highest power densities and fuel temperatures achievable with the planned fuel. This includes the power stability effects of phenomena that might affect stability.
- The number, types, and locations of all active reactor core experimental components and active reactor core-associated cooling components. If this information appears elsewhere in the SAR, the section where it is located should be referenced.
- The individual reactivity worths of fuel elements, reflector components, in-vessel and in-reflector components, experimental components, and control elements in allowed positions. If experimental facilities or components could be voided or flooded, the reactivity effects and safety considerations should be included.
- The calculated active reactor core reactivities for all possible configurations, including the limiting configuration that would yield the highest possible power density.

- Discussion of the administrative and physical constraints to prevent inadvertent addition of positive reactivity.
- The magnitude and impact of the distribution of delayed neutrons, in fuel-containing systems outside the active reactor core, on the ability to control the reactor. Include the magnitude and rate of reactivity addition associated with sudden reduced fuel salt flow out of the active reactor core and the impact on the capability to control the reactor.
- TS that control important design features, LCOs, and SRs as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4.5.2 Active Reactor Core Physics Parameters

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

- Analysis methods and values for neutron lifetime and effective delayed neutron fraction. The applicant should describe the effects of reactor operating characteristics and fuel composition changes.
- Analysis methods, values, and signs for coefficients of reactivity (e.g., fuel temperature and moderator temperature, void, and power). The applicant should describe the effects of reactor operating characteristics and fuel composition changes. This analysis, along with the analysis in Chapter 13, should show that the net effect of reactivity coefficients is sufficiently negative to prevent or mitigate damaging reactor transients.
- The axial and radial distributions of neutron flux densities, justifications for the methods used, and comparisons with applicable measurements. The applicant should describe changes in flux densities with power level, fuel composition changes, active reactor core configurations, and control element configurations. The information on neutron flux density should include peak-to-average values for thermal-hydraulic analyses. The applicant should validate these calculations by comparing them with experimental measurements and other validated calculations.
- The analysis methods used to address the dynamic behavior of changes in void fraction because of fission product gas formation and the agglomeration and transport of bubbles to the fuel solution surface. The neutronic impacts of these phenomena should be discussed to demonstrate that they have no adverse effect on safe reactor operations.
- TS that control important design features, LCOs, and SRs as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4.5.3 Operating Limits

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

- Reactivity conditions, excess reactivity, and negative reactivity for combinations of control elements that are activated and analyzed for the limiting active reactor core and operating active reactor cores during the life of the reactor. The applicant should discuss operational and safety considerations for excess reactivity.
- Excess reactivity based on reactor temperature coefficients, poisons, fission gas formation, changes in void fraction, fuel additions or removal operations, and experiment worth. The applicant should justify the upper limit on excess reactivity to ensure safe reactor operation and shutdown.
- The amount of negative reactivity that must be available by control element action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe shutdown state. The analyses should assume that the most reactive control element is non-functional (i.e., one stuck control element), non-scrammable control elements are at their most

reactive position, and normal electrical power is unavailable to the reactor. The applicant should discuss how the shutdown margin will be verified. The analyses should include all relevant uncertainties and error limits.

- The limiting active reactor core configuration that is possible with the planned fuel in this reactor. The limit should be imposed by the maximum neutron flux density and thermal power density compatible with heat removal capability and maintaining operational stability. The safety limits and limiting safety system settings for the reactor should be derived from this configuration. The detailed analyses should be included in Section 4.6. Normal operating conditions and credible events, such as a stuck control element, should be considered.
- A transient analysis assuming that an instrumentation malfunction actuates the most reactive control element in a way that it causes a continuous ramp reactivity insertion in its most reactive region. This analysis can also be based on a credible failure of a movable experiment. It should show that neither the reactor nor the fuel system boundary is damaged.
- The redundancy and diversity of control elements necessary to ensure reactor control for the considerations noted above.
- Stability definition with criteria for acceptable performance. The applicant should describe protection solutions to maintain power oscillations within operational or safety limits and might include operational limits on parameters such as power density.
- TS for safety limits, limiting safety system settings (LSSSs), LCOs, and SRs as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4.6 THERMAL-HYDRAULIC DESIGN

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

- Various systems and approaches for removing heat from the active reactor core (e.g., heat exchangers, auxiliary passive heat removal systems, gas management [heat removal] system). The expected fraction of heat removed by each approach should be discussed. The ability of the combined systems and approaches to accommodate the varying power, from gas formation, changes in void fraction, and transport, during normal and transient operation should also be discussed.
- Hydraulic characteristics of the active reactor core, fuel salt/primary cooling system salt heat exchanger number and arrangement and total system flow rates; fuel salt flow and pressures; pressure changes at piping exits and entrances; material compatibility and heat transfer between fuel salt and heat exchanger, to include plating or precipitation of material on the surfaces of the heat exchanger; natural circulation within the fuel salt; temperature profile along heat exchanger surfaces from entrance to exit; and frictional and buoyant forces. The applicant should address individual heat exchangers, as well as the active reactor core as a whole, for all flow conditions in the primary cooling system, including temperature variations and wave propagation caused by vibration and chemistry changes resulting from fuel salt to coolant salt breaches, if applicable. The transition from forced to natural convection flow in the heat exchangers should be calculated, and the applicant should prepare calculations for an event during which normal electrical power is lost and the active reactor core decay heat must be removed. The discussion should also describe

the fission gas heat removal and any additional auxiliary cooling systems and the effect that the loss of these systems would have on active reactor core coolability and decay heat removal.

- Thermal power density distribution in the fuel salt and heat fluxes into the fuel salt/primary cooling system salt heat exchangers
- Calculations and the thermal-hydraulic methodology for the transfer of heat to the coolant salt. The applicant should take into account uncertainties in thermal-hydraulic and nuclear parameters and such engineering factors as heat exchanger surface wall thickness and buildup of any layers of corrosion products both inside and exterior to the heat exchange surface. The calculations should be based on fuel measurements and procurement specifications, as well as operating history and conditions. The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use.
- Calculations and experimental measurements to determine the fuel salt conditions that ensure fuel salt temperature limits are not exceeded and fuel system boundary integrity is not lost. The applicant should calculate at least the limiting active reactor core configuration. The discussion should also examine the positive reactivity feedback characteristics of overcooling. Operating conditions should include steady fission power, shutdown decay heat, and the transients analyzed in Chapter 13. The applicant should take into account operational and fuel characteristics from the beginning to the end of fuel life.
- Thermal-hydraulic analysis considering the effects of partial or complete loss of flow in frozen piping caused by loss of heaters or overcooling. This analysis is necessary because fuel salts have high melting temperatures.
- For the active reactor core geometry and the fuel salt thermal-hydraulic characteristics (including flow instability), a discussion to establish the fuel heat removal conditions that ensure fuel system boundary integrity, solubility of fuel salt and fission products, temperature distributions, and fission gas retention capacity. The discussion should show correlations among these factors and justify their use in deriving SLs and LSSSs for the TS.
- The design bases for the primary cooling system, emergency cooling system, fission gas treatment system, and other systems designed to maintain fuel system boundary integrity should also be discussed in Chapter 5 of the SAR. The analyses here and in Chapter 13 should describe reduction of cooling scenarios for forced-flow reactors. Natural-convection cooling that removes decay heat to ensure thermal stability should also be discussed. Flow blockages should be analyzed in Chapter 13.
- In the case of MSRs, the coolant salt flows through heat exchange surfaces in contact with the fuel salt; thus, the breach of a heat exchange surface should be analyzed in Chapter 13, as should the effects of mixing of fuel and coolant salts.

4.7 GAS MANAGEMENT SYSTEM

In this section, the applicant should describe the design of the system for removing fission product gases from the active reactor core and cover gas of the MSR. The gas management system might also provide some reactor cooling; Section 4.6, Chapter 5, and Section 9.6 of the SAR should describe this aspect of its function. The applicant should describe the major components of the gas management system and their functions.

The essential functions of a gas management system are to capture volatile fission products (e.g., krypton, xenon, iodine) until suitably immobilized or stored outside the reactor and provide venting of any pressure/density transients that could result in damage to the vessel or the fuel salt/primary cooling system salt heat exchanger and result in loss of the fuel system boundary. The applicant should describe the gas management system features that perform these duties in sufficient detail to demonstrate that the active reactor core can be operated safely and in accordance with applicable environmental release criteria. This information should include the geometric dimensions of the major components and piping (including whether it is favorable geometry), materials of construction (including chemical compatibility with evolved gases), composition of any trap media/filters, pressure the equipment is designed to withstand, surge capacity for fission product storage, and any additional passive or active devices, such as alarms and pressure-relief devices, needed to perform the system's intended function.

The proper function of the gas management system, which is part of the radionuclide barrier, is essential. Malfunction or failure of components in this system could cause excessive pressure that could have positive reactivity feedback to the fuel solution and operating instability. All of the process variables controlling the gas management system must be analyzed and appropriate limits assigned in the TS to avoid such consequences.

The technical rationale for this section is that the specific components in the gas management system can vary from one applicant to another; this system is designed to be general in nature. A system for trapping entrained uranium and holding fission products until they can be safely disposed of is essential. There would be essentially three classes of hazards to be considered: an inadvertent criticality outside the active reactor core, a release of gaseous fission products, and an increase in the pressure in the free surface space above the active reactor core. The means of preventing these events must be described. One of these hazards, pressure increase, could potentially increase the density of the active reactor core and affect the power density. These potential events should be discussed in terms of reactor control.

4.8 REFERENCES

Corrosion Behavior of Reactor Materials in Fluoride Salt Mixtures, ORNL-TM-328, DeVan and Evans, 1962. In particular, Figure 9 explains the relationship between redox, temperature, and corrosion.

Compatibility Studies of Potential Molten-Salt Breeder Reactor Materials in Molten Fluoride Salts, ORNL-TM-5783, Keiser, 1977. Provides more detail about measured removal rates for long-term applications.

Stability Analysis of the Molten-Salt Reactor Experiment, ORNL-TM-1070, Ball and Kerlin, 1965. Explains the response of MSRE to oscillations.

Theoretical Dynamic Analysis of MSRE with 233U Fuel, ORNL-TM-2571, Steffy and Wood, 1969. Provides a good explanation of transient damping in MSRs.

Format and Content for Safety Analysis Reports for Research Reactors, American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21-2012, ANS, LaGrange Park, Illinois, 2012.

The Development of Technical Specifications for Research Reactors, American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, ANS, LaGrange Park, Illinois, 2013.

5 MOLTEN SALT REACTOR COOLING SYSTEMS

Chapter 5 of this guide is applicable to providing a description of the cooling systems for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should give the design bases, descriptions, and functional analyses of the MSR cooling systems. The principal purpose of the cooling systems is to safely remove the fission and decay heat from the liquid fuel and dissipate it to the environment. The discussions should include all significant heat sources in the reactor and should show how the heat is safely removed and transferred to the environment.

The class of MSRs discussed in this chapter use liquid fuel rather than solid heterogeneous fuel. The homogeneous reactor fuel (fuel salt) is contained within the fuel system boundary. Reactor fuel can typically be added online, and fission products can be removed from the fuel salt online through a fuel salt cleanup system and gas management system. Under normal operation, the fuel salt moves through the active reactor core where it is brought into a critical configuration and heat is added to the fuel salt as discussed in Chapter 4, “Molten Salt Reactor Description,” of the SAR. Subsequently, the fuel salt continues through a fuel salt/primary cooling system salt heat exchanger where heat is transferred from the fuel salt to the primary cooling system containing a compatible non-fuel salt. The fuel salt then returns to the active reactor core. The coolant salt passes heat to the heat dissipation system through subsequent system interfaces. The subsystems can be forced convection or natural thermal convection. In this chapter, an MSR applicant should describe and discuss all systems that remove and dispose of heat from the reactor fuel, as well as major components.

Heat from the fuel salt is normally transferred through one or more fuel salt/primary cooling system salt heat exchangers to the primary cooling system. On reactor shutdown, decay heat can be removed through an auxiliary heat exchanger instead of the fuel salt/primary cooling system salt heat exchanger. Some non-power MSR designs might be capable of dissipating decay heat in the natural convection mode. Other MSR designs might use a fuel salt drain tank or other irradiated fuel salt storage tanks to allow for safe storage of the fuel salt in the event of a design-basis accident or for maintenance. Decay heat generated by fuel salt in the fuel salt drain tank is removed independently and the drain tank heat removal system is discussed as a separate system in Section 5.7, “Fuel Salt Drain System.” Supplemental cooling systems might also be necessary to remove heat from the cover gas space above the active reactor core, from a fuel salt cleanup system, and from any experimental facilities. The design of the MSR cooling systems is based on selecting among interdependent parameters, including thermal power level, research capability, available fuel type, active reactor core physics requirements, and radiation shielding.

If the MSR is designed to operate at such low power levels that no significant temperature increases will occur during normal operation, an engineered cooling system for heat removal is not required. For those reactors, the applicant should, in Chapter 4 of the SAR, discuss the disposition of the heat produced, estimate potential temperature increases during operation, and justify why an engineered cooling system for heat removal is not required. In this chapter, the applicant should summarize those considerations and conclusions.

For all other non-power MSRs, the applicant should describe and discuss in this chapter systems to remove and dispose of the waste heat. The design bases of the MSR cooling systems for the full range of normal operation should be based on ensuring acceptable reactor conditions established in Chapter 4 of the SAR. The design bases of any features of the active reactor 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,” of the SAR. These features, such as an auxiliary heat exchanger or a fuel salt drain tank, 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 (TS) where analyses are used as the basis for a requirement.

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

5.1 SUMMARY DESCRIPTION

In this section, the applicant should briefly describe the reactor cooling systems and supplementary vessel heat removal pathways, summarizing the principal features. Information should include the following:

- Summary of the fuel system boundary emphasizing heat transfer mechanisms (a description of the fuel system boundary is provided in Chapter 4).
- Type of fuel salt flow: forced convection, natural convection, or both.
- Type of primary cooling system, if one is present, and the method of heat transfer.
- Type of heat dissipation system, if one is present, and the method of heat transfer to the environment.
- Description of the capability to provide sufficient heat removal to support continuous operation at full licensed power.
- Description of any supplementary methods of removing active reactor core decay heat.
- Description of any methods employed to prevent fuel salt from freezing in the vessel or in any attached piping or tank, such as a drain tank.
- Description of any methods employed to prevent primary cooling system salt from freezing in the piping or in any attached tank, such as a drain tank.
- Description of special or facility-unique features.

For an MSR, the applicant should provide additional information in this section on the active reactor core cooling systems unique to the principal features of MSRs, including supplementary active reactor core heat removal pathways. If the active reactor core cooling system is not the sole means of heat removal and the active reactor core heat removal is partitioned between supplementing pathways (such as a gas management system above the active reactor core or a fuel salt cleanup system), these additional pathways should be mentioned. The energy partitioning should be given. These other means of heat transport from the active reactor core should be summarized, including the corresponding amount of heat transported from the active reactor core and the fraction of total active reactor core heat removed.

5.2 FUEL SYSTEM BOUNDARY AND FUEL SALT HEAT TRANSPORT

The basic requirements and design bases of the fuel salt and the fuel system boundary are to maintain reactor facility conditions within the range of design conditions and accident analyses assumptions derived from other chapters of the SAR, especially Chapters 4 and 13. The applicant should show the interrelationships among all SAR chapters and the way heat transport is provided for the fuel salt. The following information should be included:

- Design bases and functional requirements of the fuel system boundary.
- Schematic and flow diagrams of the fuel system boundary, showing such essential components as the heat source (active reactor core), heat sinks (heat exchangers), drain tank, gas management system, pumps, piping, valves, control and safety instrumentation, interlocks, and other related subsystems.
- Tables of allowable ranges of important design and operating parameters and specifications for the fuel salt and the fuel system boundary and its components, including:
    - Materials and coatings

- Fuel salt flow rates
- Inlet and outlet temperatures and pressures throughout the system
- Elevation of components and fuel salt levels relative to the vessel
- Construction materials of components
- Fabrication specifications of safety-related components
- Fuel salt chemistry and composition control limits
- Fuel salt quality requirements for operation and shutdown conditions, including redox and conductivity at a minimum
- Key parameters of interfacing components and systems

- Discussions and analyses keyed to drawings showing how the fuel system boundary provides the necessary cooling for all heat loads and all potential reactor conditions analyzed in the thermal-hydraulics section of Chapters 4 and 13, including the following:
  - Removal of heat from the fuel salt by forced-convection or natural-convection cooling, or both for those reactors licensed to operate in both modes. Discussion and analyses of the effect of the size, shape, and structural features of the fuel system boundary on cooling characteristics.
  - Transfer of heat from the fuel salt to the primary cooling system for all reactor conditions. This discussion should include any heat exchanger design and operating conditions. Some MSRs might have only a fuel system boundary that functions as a heat reservoir. For such systems, the analyses should include any factors that limit continuous operation, such as fuel salt temperature and the proposed TS that ensure operation within the analyzed limits.
  - Safe reactor shutdown, including passive or fail-safe transition from forced- to natural-convection cooling and removal of decay heat from the fuel. This discussion should include the methods of decay heat removal in the event of loss of off-site electrical power. The discussion should also include any use of an auxiliary heat exchanger or a drain tank for accident mitigation and decay heat removal. emergency cooling system (ECS) features, such as an auxiliary heat exchanger or a fuel salt drain tank, should be summarized in this chapter and discussed in detail in Chapter 6 of the SAR. If applicable, Section 5.8 of the SAR specifically addresses operation of a fuel salt drain tank.
  - Locations, designs, and functions of such essential components as heat exchangers, drains, drain tanks, cover gas space, syphon-breaks, pumps, isolation valves, and check valves. These components ensure that the fuel system boundary is operable and that uncontrolled loss or discharge of fuel salt from the fuel system boundary does not occur. Radiological effects of potential fuel salt releases should be analyzed primarily in Chapter 11, “Radiological Protection Program and Waste Management.”
- Discussion of the control and safety instrumentation, including location and functions of sensors and readout devices. The scram or interlock functions that prevent safety limits from being exceeded should be shown and discussed, including the related TS.
- Description and function of any special features of the fuel system boundary such as special coatings, the design of the fuel salt/primary cooling system salt heat exchanger, the use of a fuel salt drain tank and the use of freeze plugs or other special valves.
- Brief description of design to control fission product leakage through heat exchange surfaces.
- Brief description and functions of special features or components of the fuel system boundary that affect or limit personnel radiation exposures from fission product gases, soluble fission products, insoluble fission products, delayed neutrons, tritium, and radioactive contaminants. Description of radiation monitors or detectors incorporated into the fuel system boundary and discussion of their functions.

- Brief discussion and references to detailed discussions in later sections of auxiliary systems using fuel salt, such as fuel salt cleanup, salt makeup, emergency cooling, experiment cooling, and experimental facility cooling. The direct effect of these auxiliary systems on the design and functioning of the fuel system boundary should be discussed.
- The effect of any special shielding features, such as fuel storage facility shielding and experimental facility shielding (e.g., beam tubes), on the function of the fuel salt system boundary and fuel salt system cooling should be discussed.
- Discussion of leak detection and allowable leakage limits, if any.
- Discussion of normal fuel salt radiation concentration limits, including sampling frequency, isotopes of interest, and actions to be taken if limits are exceeded.
- Discussion of TS requirements for parameters of the fuel system boundary, including the bases and surveillance requirements

A fuel salt drain tank may be provided to allow for safe storage of the fuel salt in the event of a design-basis accident or for maintenance. The fuel salt drain tank maintains the fuel salt in a noncritical configuration and is cooled separately to remove decay heat. A detailed discussion of an engineered safety feature (ESF) function for the fuel salt drain tank system and its activation should be provided in Chapter 6 of the SAR. The fuel salt drain tank cooling system should be discussed in Chapter 9 of the SAR. Radiation protection for this system should be discussed in Chapter 11 of the SAR. Accident analyses regarding ESF response for this system should be discussed in Chapter 13 of the SAR. If included in the design, the applicant should summarize the fuel salt drain tank operation in this section of the SAR.

The applicant should provide the following information about the fuel salt drain tanks:

- Design bases and functional requirements of the fuel salt drain tanks.
- Schematic and flow diagrams of the fuel salt drain tanks, showing such essentials as how the drain tank connects to the respective fuel salt system, pumps, piping, valves, heaters, continuous mixing, control and safety instrumentation, and interlocks.
- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks.
- Specifications and limitations on fuel salt chemistry, composition limits, quality and corrosion of the drain tank components, including off gassing.
- Discussion of any TS requirements, including the bases and surveillance requirements.
- Discussion on how gaseous fission products or other gases generated from the fuel salt while in the fuel salt drain tank will be managed.

5.3 COOLING SYSTEMS

Most MSRs include a primary cooling system that interfaces directly with the fuel salt through the fuel salt/primary cooling system salt heat exchanger(s). The primary cooling system is detailed in Section 5.3.1. Heat is transferred from the primary cooling system to the environment through a heat dissipation system. The heat dissipation system is detailed in Section 5.3.2.

5.3.1 Primary Cooling System

In this section, the applicant should give information about those MSRs that include a primary cooling system. For other MSRs, the applicant should state that a primary cooling system is not needed and should justify that conclusion. The applicant should provide the following information:

- The design bases and functional requirements of the primary cooling system, including whether the system is designed for continuous full-power reactor operation.
- Schematic and flow diagrams of the primary cooling system, showing such essentials as how the fuel salt/primary cooling system salt heat exchanger connects the fuel system boundary (i.e., the heat source) to the primary cooling system, pumps, piping, valves, control and safety instrumentation, and interlocks and how the primary cooling system interfaces with the heat dissipation system for ultimate release of the heat.
- Tables of the range of important design and operating parameters and specifications of the primary cooling system, including the following:
  - Coolant salt materials and their source
  - Coolant salt flow rates
  - Location of primary cooling system in relation to the reactor and heat exchangers
  - Construction materials and fabrication specifications of components
  - Primary cooling system salt chemistry and composition control limits, including equipment used to control and/or monitor salt chemistry.
  - Specifications and limitations on coolant salt quality and corrosion of the primary cooling system components including the environmental effects of the use of primary cooling system salt chemicals
- Discussion and functional analyses keyed to the drawings showing how the primary cooling system provides the necessary heat transfer for all potential fuel system boundary conditions. These discussions should address the following:
  - Inlet and outlet temperatures and pressures throughout the primary cooling system, including the pressure differential between the fuel salt and primary cooling system in the fuel salt/primary cooling system salt heat exchanger. The applicant should discuss how the pressure in the primary cooling system is maintained above that in the fuel system boundary for all operating conditions or analyze the radiological effect of leakage of fuel salt into the primary cooling system. Isolation of the fuel salt/primary cooling system salt heat exchanger during shutdown periods is an acceptable method to control potential fuel salt-to-primary cooling system leakage if primary cooling system pressure is lower than fuel system boundary pressure only during periods of system shutdown. The applicant does not need to perform an analysis of fuel salt-to-primary cooling system leakage if primary cooling system pressure is lower than fuel system boundary pressure for only short periods for system testing or repair. If the transfer of fuel salt into the primary cooling system is caused by an abrupt event, such as a heat exchange surface rupture, the analysis should be given in Chapter 13 and summarized here.
  - Control of heat removal from the primary cooling system necessary to maintain fuel temperatures in the fuel system boundary within the limits derived in the thermal-hydraulics analyses in Chapters 4, 6, and 13 of the SAR.
  - Transfer of heat from the fuel salt/primary cooling system salt heat exchangers to the heat dissipation system when the fuel system boundary operates in all anticipated and licensed modes, including forced-convection flow and natural-convection flow, as applicable.
  - Safe reactor shutdown and removal and dissipation of decay heat, including evaluation of the fuel salt system change from forced-convection flow to natural-convection flow if forced convection flow is an allowed mode of operation.
  - Response of the primary cooling system to the loss of fuel salt, including dumping the fuel salt into a fuel salt drain tank, if applicable.

    - Locations, designs, and functions of such essential components as drains, sumps, pumps, makeup salt, and check valves that ensure fuel salt is not inadvertently transferred to the primary cooling system and released to the environment.
- Discussion of control and safety instrumentation, including locations and functions of sensors and readout devices and interlocks or safety capabilities
- Brief description and functions of special features or components of the primary cooling system that affect or limit personnel radiation exposures from neutron activation of coolant salts and contaminants, including the use of pressure differential between the primary cooling system and the fuel salt.
- Descriptions of functions of any radiation monitors or detectors incorporated into the primary cooling system. Discussion of surveillance to measure primary cooling system activity including frequency, action levels, and action to be taken.
- Brief comments and reference to detailed discussion in other sections of auxiliary cooling systems that transfer heat to the primary cooling system such as the emergency cooling system or experiment cooling.
- Discussion of TS requirements, as appropriate, for the primary cooling system, including the bases and surveillance requirements.

The primary cooling system can also include a salt drain tank to provide for system maintenance. Any required cooling system for a coolant salt drain tank should be discussed in Chapter 9 of the SAR. The applicant should provide the following information about primary cooling system salt drain tanks:

- Design bases and functional requirements of all primary cooling system salt drain tanks.
- Schematic and flow diagrams of primary cooling system salt drain tanks, showing such essentials as how the drain tank connects to the respective coolant salt system, pumps, piping, valves, heaters, continuous mixing, control and safety instrumentation, and interlocks.
- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks.
- Discussion of any TS requirements, including the bases and surveillance requirements.

5.3.2 Heat Dissipation System

In this section, the applicant should give information about those MSRs that include a heat dissipation system. For other MSRs, the applicant should state that a heat dissipation system is not needed and should justify that conclusion.

The applicant should provide the following information:

- The design bases and functional requirements of the heat dissipation system, including whether the system is designed for continuous full-power reactor operation.
- Schematic and flow diagrams of the heat dissipation system, showing essentials such as how the primary cooling/heat dissipation heat exchanger(s) connect the primary cooling system to the heat dissipation system, pumps, piping, valves, control and safety instrumentation, interlocks, and interface with the environment for ultimate release of the heat.
- Tables showing the range of important design and operating parameters and specifications of the heat dissipation system, including the following:
    - Coolant materials and their source.
    - Coolant flow rates.
    - Type of heat dissipation system, such as cooling tower, refrigerator, radiator, or body of water.

  - Location of heat dissipation system in relation to the primary cooling system.
  - Construction materials and fabrication specifications of components.
  - Heat dissipation system specifications related to environmental factors (e.g., temperature and humidity).
  - Heat dissipation system chemistry and composition control limits.
  - Specifications and limitations on coolant quality and corrosion of the heat dissipation system components, including the environmental effects of the use of system chemicals.
- Discussion and functional analyses keyed to the drawings showing how the heat dissipation system provides the necessary cooling for all potential reactor conditions. These discussions should address the following:
  - Inlet and outlet temperatures and pressures throughout the heat dissipation system, including the pressure differential between the primary cooling system, and the heat dissipation system and interfacing systems in all system heat exchangers.
  - Control of heat removal from the heat dissipation system necessary to maintain fuel salt temperatures in the fuel system boundary within the limits derived in the thermal-hydraulics analyses in Chapters 4 and 13 of the SAR.
  - Removal of heat from the heat dissipation system heat exchangers and release to the environment when the fuel system boundary operates in all anticipated and licensed modes, including forced-convection flow and natural-convection flow, as applicable.
  - Safe reactor shutdown and removal and dissipation of decay heat, including evaluation of the fuel system boundary change from forced-convection flow to natural-convection flow if forced-convection flow is an allowed mode of operation.
  - Response of the heat dissipation system to the loss of fuel salt (or primary cooling system salt), including dumping the fuel salt into a fuel salt drain tank (or primary cooling system salt into a drain tank), if applicable
- Discussion of control and safety instrumentation, including locations and functions of sensors and readout devices and interlocks or safety capabilities
- Descriptions of functions of any radiation monitors or detectors incorporated into the heat dissipation system. Discussion of surveillance to measure heat dissipation system coolant activity including frequency, action levels, and action to be taken.
- Brief comments and reference to detailed discussion in other sections of auxiliary cooling systems that transfer heat to the heat dissipation system.
- Discussion of TS requirements, as appropriate, for the heat dissipation system, including the bases and surveillance requirements.

The heat dissipation system might include coolant drain tank(s) to provide for system maintenance. Any required cooling system for a coolant drain tank is discussed in Chapter 9 of the SAR.

The applicant should provide the following information about heat dissipation system drain tanks:

- Design bases and functional requirements of all heat dissipation system drain tanks.
- Schematic and flow diagrams of heat dissipation system drain tanks, showing such essentials as how the drain tank connects to the respective coolant system, pumps, piping, valves, control and safety instrumentation, and interlocks.
- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks.
- Discussion of any TS requirements, including the bases and surveillance requirements.

5.4 FUEL SALT CLEANUP SYSTEM

In MSRs fission products are generated and entrained in the fuel salt because there is no fuel cladding. Gaseous fission products, such as xenon and krypton bubble off continuously and are typically removed from the cover gas space through a gas management system without any significant impact on reactor operation. The gas management system is discussed in Chapter 4 and Chapter 11 of the SAR. Gas management system cooling is discussed in Chapter 9 of the SAR. Soluble and non-soluble fission products remain in the fuel salt. Therefore, the fuel salt in the non-power MSR will become highly radioactive. Insoluble fission products tend to plate out on reactor surfaces. The soluble fission products can be removed from the fuel salt by chemical processing, polishing, or filtration (operated in batch mode or continuously). The radiological controls for the fuel salt cleanup system are discussed in Chapter 11 of the SAR. Cooling for the filtering system or chemical processing loop is discussed in Chapter 9 of the SAR. In some non-power MSR designs, the filtering system or chemical processing loop can also be used to add additional fuel to the fuel salt. Fuel handling is discussed in Chapter 9 of the SAR; if appropriate, fuel addition is summarized here. The purity of the fuel salt should be maintained as high as reasonably possible for the following reasons:

- To limit the chemical corrosion of the fuel system boundary
- To maintain the thermal-dynamic properties of the fuel salt within the operational limits established for the fuel

The applicant should provide the following information:

- The design bases and functional requirements of the fuel salt cleanup system. The design bases should be consistent with the discussions in Chapter 4 of the SAR. Any recommendations from the fuel vendor should also be addressed.
- The design bases should be consistent with the discussions in Chapter 4 of the SAR. Any recommendations from the fuel vendor should also be addressed.
- Schematic drawings and flow diagrams of the fuel salt cleanup loop.
- Table of specifications for the cleanup system demonstrating that it is designed for the volume and throughput of the fuel salt.
- Locations and functions of control and monitoring instrumentation, including sensors, recorders, and meters. The discussion of monitors should include methods for continuously assessing fuel salt quality and effectiveness of the cleanup system.
- Locations and functional designs of cleanup system components such as branch points, pumps, valves, heaters, filters, and demineralizers.
- Interface with any liquid fuel addition system components as discussed in Chapter 9 of the SAR.
- Discussion of schedules and methods for replacing or regenerating cleanup system components and disposing of resultant radioactivity to ensure that radiation exposures do not exceed the limits discussed in Chapter 11 of the SAR.
- Summary of methods for predicting, monitoring, and shielding radioactivity deposited in cleanup system components from routine operations. The detailed discussion should be in Chapter 11 of the SAR.
- Summary of methods for predicting and limiting exposures of personnel in the event of inadvertent release of excess radioactivity in the fuel salt system and deposition in cleanup system components. The detailed discussion should be in Chapter 13 of the SAR.
- Summary of methods for preventing an inadvertent criticality outside the active reactor core in the fuel salt cleanup system (e.g., subcritical when filled with optimally moderated fuel salt).
- Provisions in the design and operation of the cleanup system to avoid malfunctions that could lead to significant loss of fuel salt and fission products, which could cause radiological exposure

of personnel or release to the unrestricted environment to exceed the requirements in 10 CFR Part 20 and the facility ALARA (as-low-as-is-reasonably-achievable) program guidelines.

- Discussion of TS requirements for the fuel salt cleanup system, including the bases and surveillance requirements.

5.5 SALT MAKEUP SYSTEMS

There might be a need for salt to be replaced or replenished in the fuel salt system or the primary cooling system because of operational activities or a need to adjust the fuel salt composition. If applicable, salt makeup for the fuel salt is detailed in Section 5.5.1. If applicable, primary cooling system salt makeup is detailed in Section 5.5.2.

5.5.1 Fuel Salt Makeup System

MSR designs should include a system or a procedure that meets the projected needs for salt makeup in the reactor fuel. The makeup salt system need not be designed to provide a rapid, total replacement of the fuel salt inventory, but it should be able to maintain the minimum acceptable fuel salt quantity and quality for reactor operation.

The applicant should provide the following information:

- The design bases for the fuel salt makeup system that account for all activities that could necessitate the addition of generally solvent halide salt(s) to the fuel salt. Although a required emergency cooling system need not be a part of the fuel salt makeup system, if it exists, it should be discussed in Chapter 6 of the SAR.
- Schematic diagrams and functional discussions that show the source of salt, the methods of addition to the fuel salt, and the requirements for pretreatment before addition.
- If appropriate, discussion of steps taken to ensure no significant change in fuel salt characteristics occurs (e.g., redox chemistry, temperature) when makeup fuel salt is added.
- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks. Methods should be discussed for tracking additions of makeup salt to detect significant changes that might indicate leaks, or other malfunction of the fuel system boundary.
- Interface with any liquid fuel addition system components as discussed in Chapter 9 of the SAR, including a summary of methods for preventing an inadvertent criticality outside the active reactor core in the fuel salt makeup system (e.g., subcritical when filled with optimally moderated fuel salt).
- Discussion of TS requirements for the fuel salt makeup system, including the bases and surveillance requirements.

5.5.2 Primary Cooling Makeup System

MSR designs should include a system or a procedure that meets the projected needs for primary cooling system salt makeup. The primary cooling makeup system need not be designed to provide a rapid, total replacement of the cooling salt inventory, but it should be able to maintain the minimum acceptable cooling salt quantity and quality for primary cooling system operation.

The applicant should provide the following information:

- The design bases for the primary cooling makeup system that account for all activities that could cause a decrease in the cooling salt.

- Schematic diagrams and functional discussions that show the source of salt, methods of addition to the cooling salt, and requirements for pretreatment before addition.
- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks. Methods should be discussed for tracking additions of makeup salt to detect significant changes that might indicate leaks or other malfunction of the primary cooling system Boundary.
- Discussion of TS requirements for the primary cooling makeup system, including the bases and surveillance requirements.

5.6 AUXILIARY COOLING SYSTEMS

In addition to the systems discussed previously that are associated with the fuel salt or primary cooling system, other auxiliary cooling systems might require the use of fuel salt or primary cooling salt and could affect the operation or safety of the reactor. If the reactor design includes an emergency cooling system, such as an auxiliary heat exchanger, it should be described and discussed in Chapter 6. The following auxiliary systems that use fuel salt, primary cooling system salts, or heat dissipation system coolants for cooling should be discussed in this section (if applicable):

- Experiment cooling
- Experimental facility cooling
- Biological shield cooling
- Thermal shield cooling
- Fuel storage cooling and shielding
- Reflector cooling
- Drain tank cooling

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

- Design bases and functional requirements of the auxiliary cooling systems based on discussions elsewhere in the SAR, such as Chapters 4, 9, and 10, “Experimental Facilities and Utilization.”
- Schematic drawings and flow diagrams that show the source of the salt, locations of sensors and instruments, and locations of the components cooled.
- Tables of the range of important parameters of the systems and specifications of materials and components.
- Discussion of components to be cooled, the source of heat, the source of the coolant salt, heat transfer to the coolant salt, and coolant salt heat dissipation.
- Discussion of the provisions in the auxiliary cooling system designs to prevent interference with safe reactor shutdown.
- Discussion of the provisions in the auxiliary cooling system design to prevent the uncontrolled release of fuel salt or radiation exposures that would exceed the requirements in 10 CFR Part 20 and the facility ALARA program guidelines.
- Discussion of any TS requirements for the auxiliary cooling systems, including the bases and surveillance requirements.

5.7 REFERENCES

“MSRE Design and Operations Report, Part 1: Description of Reactor Design,” ORNL-TM-728, Robertson, 1965.

Format and Content for Safety Analysis Reports for Research Reactors, American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21-2012, ANS, LaGrange Park, Illinois, 2012.

The Development of Technical Specifications for Research Reactors, American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, ANS, LaGrange Park, Illinois, 2013.

6 ENGINEERED SAFETY FEATURES

Chapter 6 of this guide is applicable to providing a description of the engineered safety features for the licensing of a nonpower MSR. In this chapter, the applicant should discuss and describe engineered safety features (ESFs) for an MSR. ESFs are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to the public, the facility staff, and the environment within acceptable values. The concept of ESFs evolved from the defense-in-depth philosophy of multiple layers of design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The need for ESFs is determined by the SAR analyses of accidents that could occur, even though prudent design of the facility has made the incidence of an accident very unlikely. It is also possible that for a particular MSR design, the SAR analyses will show that ESFs are not needed.

Normal operation of an MSR is defined as operation with all process variables and other reactor parameters within allowed conditions of the license, technical specifications (TS), applicable regulatory limits, and design requirements for the system. Accidents at MSR facilities assume failure of a major component such as the fuel system boundary or a reactivity addition event. Licensees analyze a maximum hypothetical accident that assumes an incredible failure that leads to unacceptable fuel system boundary degradation. These postulated accidents are compared with acceptance criteria such as the safety limits from the TS or, where there are radiological consequences, to accepted regulatory limits (10 CFR Part 20 or 100). Consideration must also be given to the fission product decay heat generated in the interfacing gas management system and fuel salt cleanup system (if applicable). Cooling systems described in Chapter 9, "Auxiliary Systems," are designed to provide normal cooling for these systems. However, the maximum hypothetical accident that assumes an incredible failure should also consider these interfacing systems because of the radioactive material content and the potential radiological consequences to accepted regulatory limits (10 CFR Part 20 or 100) resulting from interfacing system boundary degradation. The results of the accident analyses are presented in the SAR, Chapter 13, "Accident Analyses." ESF systems must be designed to function for the range of conditions from normal operation through accident conditions.

The analyzed accident scenarios that the applicant should present in Chapter 13 of the SAR include the following:

- Loss of fuel salt
- Loss of fuel salt flow
- Insertion of excess reactivity (rapid or ramp)
- Loss of fuel system boundary integrity or mishandling of fuel salt
- Loss of gas management system integrity or fuel salt cleanup system integrity
- Precipitation of fuel
- Failure or malfunction of an experiment
- Other uncontrolled release of radioactive material
- Loss of electrical power
- External events such as floods and earthquakes

The SAR accident analyses for many MSRs might show that ESFs are not required, even for the maximum hypothetical accident. In other cases, the accident analyses might show that ESFs need to be considered in mitigating the potential release of radioactive material to the environment. Note that there could be several systems containing highly radioactive materials within separate boundaries that should be analyzed apart from the fuel system boundary when considering the maximum hypothetical accident.

The accident analyses provide the design bases for any required ESF. The ESF design should be as basic and fail-safe as practical. Because MSRs are designed to provide reasonable assurance of protection of public health and safety, few, if any, accidents should require redundant or diverse ESF systems. However, consideration should be given to adding redundancy and diversity to ESF systems if the reactor is of a higher power level (2 MW or greater thermal power level), if an ESF system would be susceptible to loss of capability to function because of a single failure, or if the radiological consequences to the public of the accident that the ESF is designed to protect against would be very serious if the ESF failed.

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

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

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

Typical ESFs that might be required at MSRs are (1) the confinement, (2) the containment, and (3) the emergency cooling system (ECS). In addition, features required in the facility heating, ventilation, and air-conditioning (HVAC) system to mitigate the consequences of accidents should be treated as part of the ESFs of the confinement or containment system. HVAC systems are discussed in Chapter 9 of the SAR. The applicant should discuss any additional ESFs in a comparable way.

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

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

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

venting of containment, and the design should provide for both normal and emergency operational modes. A containment can be designed to be integral with the facility HVAC and liquid waste systems:

For an MSR that employs a confinement system, such a system:

- Usually responds to accidents by reducing and changing the airflow paths to and from the building (a containment seals the building. from the environment and significantly reduces releases of radioactive material to the environment).
- Has doors with gasket-type seals (airlocks for containments).
- Might not have sealing isolation dampers on air penetrations (sealing isolation dampers for containments).
- Cannot maintain as high a negative differential pressure as a containment.
- Is not as leak tight as a containment and the leak rate normally cannot be confirmed through testing.
- Cannot control the release from an event that results in positive pressure in the reactor building.
- Usually has less direct radiation shielding capacity than a containment because the walls are thinner.
- Is less resistant than a containment to challenges placed on the building by the external environment.

If the analyses show that a confinement ESF will mitigate the consequences of the most limiting accident scenario to acceptable levels, a containment ESF would not be required, although some licensees have chosen to build containments as an additional design conservatism. MSR designs can employ multiple individual containments around interfacing systems containing highly radioactive materials such as the gas management system or the fuel salt cleanup system.

(3) An ECS is designed to provide a source of heat removal to limit fuel system boundary damage or interfacing system boundary degradation from decay heat should the normal path for heat removal be lost.

The objective of non-power reactor ESFs is to ensure that projected radiological exposures from accidents are kept below the regulatory limits. The regulations defining the limits on releases from non-power reactors during accident conditions depend on whether the non-power reactor (see 10 CFR 50.2) is a testing facility (also called a test reactor in some regulations) or a research reactor (see 10 CFR 170.3). For a research reactor, the results of the accident analyses must show that exposures generally meet the requirements of 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.

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

6.1 SUMMARY DESCRIPTION

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

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

6.2 DETAILED DESCRIPTIONS

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

6.2.1 Confinement

The applicant should discuss in detail the confinement and associated HVAC systems that function as ESFs. For the confinement to function as an ESF, the design bases for the consequence-mitigation functions should be derived from the accident analyses in the SAR Chapter 13. Confinements and HVAC systems can also have functions that are not considered functions of ESFs and these should be addressed in other chapters of the SAR, such as Chapter 9.

Most MSRs release small quantities of airborne radioactive material, primarily fission product gaseous radionuclides, to the environment during normal operations. To protect the health and safety of the public and the staff, it could be necessary to control airflow through spaces containing radioactive materials and release the air in a controlled manner at a location that allows for dilution and diffusion of the radioactive material before it comes in contact with the public. In some cases, it could also be efficient to use the confinement and HVAC systems to prevent an uncontrolled release to the environment of radioactive effluents resulting from operation. This aspect of the use of ESFs during normal system operation is not considered an ESF function. However, the design bases and detailed discussions of these systems for normal operations to control releases should be given in Chapter 3, “Design of Structures, Systems, and Components,” and Chapter 9. Diffusion and dispersion of airborne radioactivity in both restricted and unrestricted environments should be discussed in Chapter 11, “Radiation Protection Program and Waste Management.”

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

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

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

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

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

6.2.2 Containment

Most MSRs will likely include a containment or a system of barriers that act as a functional containment. When the containment and associated HVAC system are required for a reactor to mitigate the consequences of a postulated accident, they are considered ESFs.

Containment for an MSR should be designed to prevent the rapid, uncontrolled release of radioactive material to the environment. A possible scenario for such a release could be an accident that involves a loss of fuel system boundary integrity and the rapid release of fission products from the fuel salt into a guard pipe or into the reactor room. The containment is designed to control the release to the environment of airborne radioactive material released in the reactor room even if the accident is accompanied by a pressure surge within the room. The walls of the containment or individual barriers can also help mitigate direct radiation exposure during certain accidents. The analyses in Chapter 13 of the SAR should include details of the postulated scenario, including the assumptions and justification for the initiating event, the progression of the scenario, the consequence-mitigating effects of the containment, and the potential radiological exposures to the most exposed member of the public. The design bases for the containment should include the postulated peak pressures, duration of the event, pressure-versus-time envelope, time during which containment integrity must be maintained while recovery from the event is implemented, limits on leakage or controlled release from the containment to the environment, loss of fuel system boundary integrity, and the quantity and type of released radioactive material.

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

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

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

To qualify as a containment, the reactor building should be a robust structure with airlocks and all other penetrations sealed (e.g., cable penetrations sealed with epoxy) or sealable (e.g., hydraulic dampers on ventilation penetrations). The building should be capable of maintaining a negative pressure in relation to the atmosphere (e.g., at least -0.5 inches of water) during normal operation and have a measurable leakage rate (e.g., less than 5 percent over 24 hours). The actual performance requirements are determined from the accident analyses in Chapter 13 of the SAR. For example, the normal function of the containment ventilation exhaust system can be divided into two trains—one that ventilates the reactor room and one that ventilates areas with high airborne radiation generation such as the gas management system, experimental facilities, or cleanup system holdup facilities. The ventilation system is normally equipped with high-efficiency particulate filters, and the accident ventilation system has a separate train(s) equipped with high-efficiency particulate and activated charcoal filters to sorb iodine.

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

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

6.2.3 Emergency Cooling System

An ECS might be required at some MSRs to remove decay heat from the fuel salt to prevent failure or degradation of the fuel system boundary if cooling is lost. It could also be required to mitigate interfacing system boundary degradation in systems with significant decay heat such as the gas management system or the fuel salt cleanup system. Cooling systems described in Chapter 9 typically provide these functions. The applicant should give the analysis of the ECS if one was identified as needed in the Chapter 13 accident analyses.

A schematic diagram should show the relationships among the major system components such as tanks, valves, pumps, piping, and any I&C systems. Special ECS I&C systems should be described briefly in this section and described fully in Chapter 7. In this section, the applicant should discuss any effects of the ECS design on normal operations and reactor safety. Analyses for MSRs should demonstrate that fuel system boundary integrity and, if applicable, interfacing system integrity will be maintained for postulated-accident scenarios.

If the ECS is a passive system (e.g., a gravity-driven system or a natural-convection cooling system), a complete description with associated analyses and data should show how cooling flow is initiated and why the system is effective. The information should demonstrate that the ECS will provide the required decay heat removal function in terms of minimum flow and time of operation for all accidents considered. If the ECS is an active system that requires sensors and an action or event to initiate operation, descriptions should include details of initiation response times and backup or redundant sensing and

control systems. The discussion should include the source of electrical power, source of coolant, heat sink, or other systems required to operate the ECS and show how operability and availability are ensured. The ECS design should show how radioactive material such as emergency coolant is controlled.
In this section, the applicant should also give the bases for TS that ensure that the ECS is available and operable when required. TS should include minimum operability requirements and the possible operations and conditions under which the ECS would be required. Test and surveillance functions and intervals should be stated in the TS to ensure operability of the ECS. See Chapter 14 of this format and content guide for details on what TS requirements should be identified and analyzed in this section.

6.3 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, “The Development of Technical Specifications for Research Reactors,” ANS, LaGrange Park, Illinois, 2013.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21-2012, “Format and Content for Safety Analysis Reports for Research Reactors,” ANS, LaGrange Park, Illinois, 2012.

7 INSTRUMENTATION AND CONTROL SYSTEMS

Part 1, Chapter 7, of the format and content guide, is applicable to providing a description of the instrumentation and control (I&C) systems for a non-power MSR. In this chapter of the SAR the applicant describes and discusses the design and operating characteristics of the I&C systems. Sufficient information should be included to explain the design criteria and bases, and to discuss the functional and safety analyses of the I&C subsystems. The I&C subsystems generally comprise the reactor control system (RCS), process instruments, the reactor protection (safety) system (RPS), instruments to initiate operation of engineered safety features (ESFs), and radiation safety monitoring systems. The I&C systems provide protective functions, such as scramming the reactor or initiating safety systems, and control functions, such as monitoring reactivity, pressure, temperature, flow, level, etc. In MSRs, fission product gases build up within the vessel, or at a free surface provided above the liquid fuel. Gases generated during fission also collect in the cover gas space connected to a gas management system. Therefore, I&C subsystems relevant to the head space and gas management systems should be provided. As described in Chapter 4, “Molten Salt Reactor Description,” I&C subsystems should support the limits on operating conditions for the homogenous fuel salt (reactor fuel) and the fuel system boundary. These limits are specified to ensure that the integrity of the fuel system boundary will not be impaired by loss of redox control, changes in reactor fuel flow, fuel addition, fission gas evolution, power oscillations, precipitation from the reactor fuel, temperature and pressure extremes or distributions, and materials compatibility. In addition, fuel handling systems as described in Chapter 9, “Auxiliary Systems,” and fuel salt cleanup systems as described in Chapter 5, “Molten Salt Reactor Cooling Systems,” should be monitored by I&C systems. These systems and their outputs can be consolidated into a control console, along with the devices and circuits that control the operation of the reactor.

The guidance in this chapter of the SAR is based on the principle that most non-power reactors can be designed and operated to pose acceptably small or insignificant risk to the public without isolating or separating the RPS from other subsystems. Additional design features, such as separation of systems, may be necessary for testing facilities. Applicants who need additional guidance beyond that given in this chapter should contact their project manager.

The RPS of a non-power MSR should monitor selected reactor operating parameters such as active reactor core neutron flux; reactor fuel flow, temperature, and level at the interface with the cover gas; primary cooling system flow, temperature, and level; heat dissipation system flow and temperature; and radiation intensity in selected areas around the reactor, including the gas management system, cleanup system, and fuel handling system. The RPS is designed to ensure reactor and personnel safety by limiting parameters to operate within analyzed operating ranges. The RPS can also give the ESF actuation system information for the operation of ESFs when the instruments indicate that abnormal or accident conditions could occur. The RCS may monitor many of the same parameters as the RPS and give information for automatic or manual control of the reactor operating conditions (e.g., reactor power, such as by adjusting reactor fuel flow, adding fuel, operation of the fuel salt cleanup system, or manipulating control elements). The instruments present operating parameter and system status information to the operator for monitoring reactor operation and for deciding on manual control actions to be taken. Instrument systems are the means through which automatic or operator control actions are transmitted for execution by the RCS. Radiation instruments show radiation levels in selected areas in the facility and could give data to the RPS, give information to help in the control of personnel radiation exposure, or monitor the release of radioactive material from the reactor and the facility.

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

7.1 SUMMARY DESCRIPTION

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

- *Type of instruments*—System instruments should be described by type [e.g., hardwired analog, computerized digital that uses stored programs (software), or combinations of these]. If a combination is used, the applicant should clearly note which portions or functions are analog and which are computerized digital, and how they relate to each other. The applicant could refer to existing systems reviewed and approved by NRC that are similar to the described system.
- *Classification of systems*—I&C systems and equipment should be classified into categories by function performed (e.g., the RCS, RPS, ESF actuation system, control console and display instrument systems, and radiation protection instruments).

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

- For the RCS, a brief discussion of each major subsystem such as manual control system, automatic control system, control element manipulation systems, bypass and interlock systems, and any integrated experiment I&C systems.
- For the RPS, the type of parameters monitored, both nuclear and non-nuclear, the number of channels designed to monitor each parameter, the actuating logic that determines the need for actions to change reactor conditions and that takes these actions, and number and type of reactivity control devices and methods.
- For the ESF actuation system, a discussion of the subsystems that detect the need for operation and that initiate operation including identification of the parameters monitored the source of input information and the number of channels designed to monitor, process, and act on the information.
- For the control console and display instruments, a discussion of the parameter display systems and equipment by which the operator can observe and control the operation of the MSR NPUF and important subsystems.
- For radiation protection instruments, a brief discussion of area and effluent radiation detection systems that monitor, alarm, or provide input to other subsystems of potentially hazardous radiation levels. The applicant should address radiation systems that monitor effluent streams from the facility, state the type of effluent (such as airborne or liquid), and list alarms or signals to other subsystems.
- A summary of the human-machine interface principles to discuss how the information displays and the characteristics of the displays (e.g., location, range, type, and resolution) (a) support the system design; and (b) incorporate human factors principles to discuss the functions and the characteristics of the controls to provide indications of the MSR NPUF status and allow operators to take appropriate manual actions, when needed.

7.2 DESIGN OF INSTRUMENTATION AND CONTROL SYSTEMS

7.2.1 Design Criteria

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

- Systems and components (including I&C systems) determined by the analyses in the SAR to be important to the safe operation or shutdown of the reactor and support systems should be designed to be in accordance with local building and siting codes, and should be able to withstand the effects of natural phenomena without loss of capability to perform their safety function (see Chapter 3, “Design Structures, Systems, and Components,” for additional information).
- I&C systems and components determined in the SAR analyses to be important to the safe operation or shutdown of the reactor and support systems should be designed, located, and protected so that the effects of fires or explosions would not prevent them from performing their safety functions.
- I&C systems and components determined in the SAR to be important to the safe operation and shutdown of the reactor and support systems should be designed to function reliably under anticipated environmental conditions (e.g., temperature, pressure, humidity, and corrosive atmospheres) for the full range of reactor operation, during maintenance, while testing, and under postulated accident conditions, if the systems and components are assumed to function in the accident analysis.
- The RPS should be designed to automatically initiate the operation of systems or give clear warning to the operator to ensure that specified design limits of the reactor and support systems are not exceeded as a result of measured parameters indicating the onset of potential abnormal conditions. The ESF actuation system should be designed to automatically initiate operation to mitigate the consequences of abnormal conditions or accidents.
- I&C systems should be designed to have functional reliability, including redundancy and diversity, commensurate with the safety function to be performed and the consequences of failure of the system to perform the safety function. For example, an I&C system for a non-power MSR should be designed to perform its protective function after experiencing common cause failures or a single random active failure within the system.
- I&C systems should be designed to fail into a safe state on loss of electrical power or exposure to extreme adverse environments.
- I&C systems should be designed so that a single failure will not prevent the safe shutdown of the reactor, i.e., the levels of redundancy used in the safety-related systems assure that: (1) no single failure results in loss of the safety function; and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the I&C design can be otherwise demonstrated.

7.2.2 Design-Basis Requirements

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

- The function or purpose of systems or instruments considering which parameters are monitored or controlled.
- The range of values that monitored variables may exhibit for normal operation, shutdown conditions, and for postulated accidents.
- Safety or control functions and any unique or facility-specific functions performed by the I&C system or subsystems.
- Specification of alarm, trip, and actuation setpoints derived from accident or other operational analyses of the instrumented system or function.
- Any special requirements such as redundancy, diversity, quality assurance, and environmental requirements derived from the results of analyses of the full range of operating conditions and postulated accidents.
- The specification of precision and accuracy requirements for the instruments, control subsystems, or components.
- The specification of number and type of channels required to monitor variables.
- The system operational and support requirements such as those for electrical, mechanical, structural, cooling, heating, and signal input.
- The requirements that controls and instruments be grouped and located so that operators can easily reach and manipulate he controls while readily observing on meters and displays the results of their actions (operator interface requirements).
- For digital computer systems, in addition to the foregoing, the applicable guidelines from IEEE 7-4.3.2-2016, "IEEE Standard Criteria for Programmable Digital Devices in Safety systems of Nuclear Power Generating Stations," for the design, application, and evaluation of digital computer hardware and software and ANSI/ANS 10.4-2008 (R2016), "Verification and Validation of Non-Safety-Related Scientific and Engineering Computer Programs for the Nuclear Industry," for evaluating the verification and validation programs for software for use in the I&C system. However, neither of these standards was uniquely developed for non-power reactors and may contain sections and requirements that do not apply to a particular situation. Furthermore, the technology and safety principles on which computerized I&C systems are based are changing. Regulatory Guide (RG) 1.152, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," discusses the use of digital computers in nuclear power plant safety systems. Applicants should verify the current requirements with the NRC. The testing programs for computer systems help to verify that the software will not cause unintended effects under some combinations of circumstances or conditions, or some malfunctions. ANSI/ANS Standard 15.15, "Criteria for the Reactor Safety Systems of Research Reactors," may be useful in the design, implementation, and evaluation of I&C systems for non-power reactors and should be used where applicable.

7.2.3 System Description

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

7.2.4 System Performance Analysis

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

- The SAR should describe the operation of the I&C system and present the analysis of how the system design meets the design criteria and design basis. The discussion should include accuracy, reliability, adequacy and timeliness of I&C system action, trip setpoint drift, quality of components and, if required by the analyses, redundancy, independence, and how a single failure affects both its ability to perform its safety function and the effect on operation or safe shutdown of the reactor and support systems.
- Technical specification LSSSs, LCOs, and surveillance requirements for the I&C system should be established. These parameters and requirements should include system operability tests, trip or actuation setpoint checks, trip or actuation setpoint calibrations, and any system response-time tests that are required. Surveillance intervals should be specified and the bases for the intervals, including operating experience, engineering judgement, or vendor recommendation should be discussed. The criteria for the content of TS are identified in 10 CFR 50.36 with the TS and bases for those TS provided in Chapter 14, “Technical Specifications” and Appendix 14.1. This section should identify the applicability of 10 CFR 50.36 to a system, structure, or component (SSC) or parameter, and is not a replacement for the guidance provided for Chapter 14.

7.2.5 Conclusion

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

7.3 REACTOR CONTROL SYSTEM

The RCS may perform several MSR functions, such as maintaining the reactor in a shutdown state, reactor startup, changing power levels, maintaining operation at a set power level, and shutting down the reactor through manipulation of the control elements, adjustment of fuel salt flow, manipulation of valves, etc. The RCS may be discussed using such subsystems as nuclear instruments, process instruments, control elements, and interlocks. In describing each subsystem in the SAR, the applicant should include design considerations and TS requirements.

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

The process instruments are designed to measure and display such parameters as fuel salt flow, temperature, or level at the fuel salt interface with the gas management system; fuel salt thermophysical properties; gas management system parameters; fuel salt addition; fuel salt drain; or fuel salt cleanup parameters. Note that overcooling in associated heat removal systems could interrupt fuel salt flow. In some designs, this information may also be sent to the RPS.

The typical non-power reactor has an automatic control (servo) system that controls the reactor power about a point set by the operator. Control of power and manipulation of control elements is based on a feedback or error signal. The servo control systems compare the output of a linear neutron flux channel to the desired power level setpoint, and automatically manipulate a control elements or active reactor core flow to change the neutron flux density to reduce the differential error signal until the actual reactor power level is very nearly equal to the desired power level. This process can be performed by analog control equipment or by software in a digital computer system.

The location and sensitivity of at least one reactor startup channel should be designed to ensure that changes in reactivity will be reliably indicated even with the reactor shut down. Additionally, the introduction and emission rate of any neutron startup source to allow a monitored startup with the reactor instrumentation, should be described, (see Section 4.2.4).

The RCS for non-power reactors should be a set of equipment protection interlocks and inhibits that prohibit or restrict operation of the reactor and support systems unless certain conditions are met. For example, there should be an interlock that prohibits control element manipulation unless the neutron flux in the active reactor core produces a neutron count rate sufficient to help ensure that nuclear instruments are responding to neutrons. There may also be additional equipment protection interlocks to ensure, for example, that there is sufficient fuel salt flow, shielding is intact, heat removal systems are operable, fuel salt level is sufficient, and required neutron instruments and recorders are functional. There may also be personnel protection interlocks to prevent reactor operation if certain radiation fields are excessive. Control elements may be automatically manipulated to reduce the reactor power (runback) when certain specified reactor conditions approach a predetermined limit, but total reactor shutdown (scram) is not warranted. Experimental facilities may be interlocked with the RCS to prevent reactor operation is the experimental facility is not in the correct configuration. If experiments conducted in non-power MSRs could interact with the active reactor core to change reactivity or otherwise modify the reactor operating conditions, data to the RCS or RPS from the experiment instruments may be needed to detect reactivity changes. All experiments should be carefully considered for interaction with the I&C system when the safety analysis for the experiments is performed. The analysis should consider interaction with the RCS or RPS. Where such interactions are warranted, they should meet the standards (e.g., safety significance) used for the design of the systems to which the experimental facilities will be connected.

The applicant should include the following for each RCS subsystem:

- Discuss the design criteria for the RCS as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.
- Discuss the design bases information specified in Section 7.2.2 and any additional design bases of facility-specific subsystems.
- Describe the system as specified in Section 7.2.3, including any additional system descriptive material specific to subsystem design and implementation not covered in Section 7.2.
- Analyze the operation and performance of the system as specified in Section 7.2.4 including analyses and results of any features or aspects specific to the facility design and implementation not specified in Section 7.2. Include the bases of any TS and surveillance tests with intervals specific to the design and operation of the systems. Address the specific design features of the RCS, such as the following:

    - Detector channels directly monitor the neutron flux density for presentation of reactor power level and power rate-of-change.
    - Detector channels directly monitor fuel salt flow for association with reactor power level and reactor period (or startup rate).
    - Detector channels to directly monitor all sources of MSR reactivity changes including gas management system parameters; fuel salt addition; fuel salt drain; or fuel salt cleanup parameters.

  - A continuous indication of the neutron flux density from subcritical multiplication source level through the full licensed power range. If multiple detector channels are used, this continuous indication should overlap a minimum of one decade during detector changeover.
  - A reactor period channel that covers subcritical neutron source multiplication from the approach to critical, through critical, and into the licensed power range. Depending on the analysis in the SAR some reactors may not have this channel.
  - The RCS protects against a failure or operation in a mode that could prevent the RPS from preforming its intended safety function.
  - The system and equipment are designed to assume a safe state on loss of electrical power.
  - The RCS has at least two channels of reactor power indication through the licensed power range.
  - The startup and operating power detector channels can discriminate against strong gamma radiation, such as that present after long periods of operation at full power, to ensure that indicated changes in neutron flux density are reliable.
  - The reactor power indication of at least one channel should remain reliable for some predetermined range above the licensed power level. For reactors with power level as a safety limit, the instrumentation should be able to indicate if the safety limit was exceeded. For other reactor types, at least one channel should be able to indicate if the power level was exceeded, with is the basis for limiting licensed power level.
  - The status or state of all control elements should be indicated at the control console throughout manipulation and should indicate when they are at a limit. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned.
  - The rate of change of control elements in “manual” and “automatic” modes of operation should be limited to that analyzed and allowed.
  - The rate of change of fuel salt flow in “manual” and “automatic” modes of operation should be limited to that analyzed and allowed.
  - The status of fuel addition and fuel salt cleanup systems should be indicated at the control console.
  - While in “automatic” reactor control mode, the RCS should indicate being placed in or removed from automatic control.
  - Bypasses of interlocks should be under the direct control of the reactor operator and should be indicated in the control room.
  - The RCS should meet the requirements of minimum shutdown margin defined in the TS.
- The applicant should discuss the conclusion about capability and suitability of the RCS requested in Section 7.2.5.

7.4 REACTOR PROTECTION SYSTEM

The RPS is designed to detect the need to place the reactor in a subcritical, safe shutdown condition (scram) when any of the monitored parameters exceeds the limit as determined in the SAR. Upon detecting the need, the RPS should promptly and automatically place the reactor in a subcritical, safe-shutdown condition (scram) and maintain it there. An MSR scram may include a combination of dumping fuel salt to a drain tank, minimizing reactor fuel flow, or manipulating control elements. This prevents or mitigates unintended operation in regions where risks of the following types could occur: fuel system boundary damage, uncontrolled release of radioactive materials to the unrestricted environment, or overexposure of personnel to radiation. Parameters monitored for this purpose could include active reactor core neutron flux; reactor fuel flow, temperature, and level; primary cooling system flow, temperature, and level; area radiation levels; and air concentration, or release, of radioactive materials.

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

- Discussion of design criteria for the RPS as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.
- Safety and system design bases information as specified in Section 7.2.2, including any supplemental facility-specific design bases not specified in the general system requirements.
- System descriptions consistent with that specified in Section 7.2.3, along with any subsystem description that is facility specific and that may not be identified in the general system requirements.
- Analyses of the operation and performance of the RPS similar to that specified in Section 7.2.4. This should include analysis of any features, aspects, or TS including surveillance tests that may be specific to the reactor and support systems and not identified in the general system requirements. These analyses should be based on postulated credible accidents, transients, and other events that could require RPS intervention, and should include all of the applicable features noted in Section 7.3 for the RCS. The analyses should include quantitative performance of all scrams, runbacks, interlocks, and ESF initiators. The specific design features of the RPS that should be addressed include the following:

    - Independent redundant or diverse reactor power level trips.
    - A log power level channel with a reactor period or rate-of-flux change output with a rate or period cannel set to scram in accordance with the analysis (certain reactor design do not require the period scram design feature because they are designed to accommodate rapid additions of reactivity). The log channel and a linear flux monitoring channel should accurately sense neutrons even in the presence of intense high gamma radiation.
    - Neutron flux (power) monitor channels covering the range from subcritical source multiplication to beyond the licensed maximum power level to monitor neutron flux over the anticipated range for normal operation and for accident conditions, as appropriate, to assure adequate safety.
    - A startup channel measuring neutrons at subcritical with a minimum count rate should be available to ensure that changes in reactivity will be reliably indicated even with the reactor shut down and a count rate interlock that prohibits control element manipulation unless the neutron flux in the active reactor core produces a neutron count rate sufficient to help ensure that nuclear instruments are responding to neutrons. If the count rate interlock is deemed unnecessary, the applicant should justify no needing the interlock. The detector is capable of detecting neutrons in a high gamma field and can be verified so that subcritical neutron multiplication can be determined and all reactivity changes can be monitored until the startup channel indication is overlapped by the log or linear channel power indication.
    - RPS scram time as established in the accident analysis, and any other requirements to ensure operability.

  - The scram circuit is designed for the protective action to go to completion once it is initiated. The scram circuit cannot be reset until the desired scram action has occurred and the condition(s) causing the scram has cleared.
  - Each scram channel has a separate set of contacts or other bi-stable components to trip the RPS system.
  - The manual scram switch is located where the operator has ready access.
  - Upon receipt of a scram signal, the RPS will annunciate the scram and signify the circuits that are in a tripped state.
  - The RPS shall always be capable of shutting down the reactor at least to the shutdown margin defined in the TS.
- Conclusions about capability, operability, and suitability of the RPS requesting in Section 7.2.5.

7.5 ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS

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

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

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

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

- Design criteria for the ESF actuation system as outlined in Section 7.2.1, including any criteria specific to the reactor design not outlined in the section.
- Design bases information for the ESF actuation system as specified in Section 7.2.2 and any additional facility-specific design bases not specified in the general system requirements.
- System description of each ESF actuation system similar to that specified in Section 7.2.3. The description should include:
  - Any additional facility-specific system design

- Features of the individual initiation and actuation systems which provide for them to function in concert to prevent or mitigate the consequences of postulated accident.

- System performance analysis:

  - An analysis of the operation and performance of each ESF actuation system similar to the specified in the general system requirements of Section 7.2.4, including analysis of the designs of any facility-specific features or aspects
  - A discussion of an analysis of the operation and performance of the individual systems which allow them to function in concert to prevent or mitigate the consequences of postulated accidents
  - The bases of any TS, including surveillance tests and intervals specific to the design and operation subsystem

- Conclusions about capability, operability, and suitability of the ESF actuation systems requested in Section 7.2.5.

7.6 CONTROL CONSOLE AND DISPLAY INSTRUMENTS

The control console and display instrument systems and equipment enable the operator to safely control facility operations and includes displays for the reactor operator to view the current values of important operating parameters and the status of systems and equipment. The system should annunciate abnormal conditions such as: high radiation levels; fuel system boundary integrity issues; gas management system failures; the need to actuate barrier, containment, or confinement systems; and, the need for personnel radiation protective actions. The system should also indicate automatic initiation of any protection functions.

The applicant should describe how the control console and display instruments have been designed to collect and display the operating information in such a manner that it can be readily observed and interpreted by the operator. It should describe how the manual control inputs (pushbuttons, switches, and other equipment) have been grouped, oriented, and located with respect to the relevant display instruments to enable the operator to best observe and interpret the operating information and thereby take prompt and accurate steps to supply control inputs on which the reactor control systems can act. In addition, the combined and integrated functioning of the control console and display system should be described to demonstrate how major equipment is designed to function as an integrated information-handling system to readily aid the operator in controlling operation of the reactor. The control console design should prevent unauthorized operation of the reactor and provide physical or electronic access controls to protect vital facility information (e.g., protective system setpoints) from alteration or corruption, while allowing the information to remain accessible and productive to its intended users.

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

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

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

7.7 RADIATION MONITORING SYSTEMS

Radiation monitoring instrument systems should be designed to perform several important diverse functions in the operation of a non-power reactor. MSR NPUF monitors should indicate radiation intensity and may be used for reactor operations such as to indicate the following: low fuel salt level as observed at the cover gas interface; fuel system boundary integrity issues; gas management system failures; fuel salt cleanup system failures; fuel addition system issues; the need to actuate barrier, containment, or confinement systems; the need for personnel radiation protective actions; and to monitor release of radioactive material to the environment. These systems include area radiation monitors, with displays near the instrument location and in the control room. These systems may monitor radioactive effluents in the form of gases, liquids, and airborne particulates and provide continuous air monitoring (CAM) for airborne radioactivity in occupied spaces such as the reactor room. Portable radiation monitors and personal dosimetry systems should also be included to help assess exposure and prevent overexposure of workers and other personnel. The radiation protective instruments and measures should be discussed in detail in Chapter 11, “Radiation Protection Program and Waste Management.” The information presented in this chapter should concentrate on the I&C aspects of the radiation monitoring systems and should be coordinated with the information in Chapter 11.

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

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

7.8 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, 2008 (R2016).

American National Standards Institute/American Nuclear Society, ANSI/ANS 10.5, “Accommodating User Needs in Scientific and Engineering Computer Software Development,” ANS, LaGrange Park, Illinois, 2006.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, “Quality Assurance Program Requirements for Research Reactors,” ANS, LaGrange Park, Illinois, 1995 (R2018).

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 (R1986). (withdrawn)

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

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

Electric Power Research Institute, “Guideline on Licensing Digital Upgrades,” EPRI TR-102348, Rev. 1, Palo Alto, CA, December 1993.

Electric Power Research Institute, “Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications,” EPRI TR-106439, Palo Alto, CA, October 1996.

Electric Power Research Institute, “Generic Requirements Specification for Qualifying a Commercially Available PLC for Safety-Related Applications in Nuclear Power Plants,” EPRI TR-107339, Palo Alto, CA, December 1996.

Electric Power Research Institute, “Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule,” Palo Alto, CA, 2002. 1002833.

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, 2016.

Institute of Electrical and Electronics Engineers, IEEE Standard 1023-2004, “IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities,” New York, New York, June 2005.

Nuclear Energy Institute, NEI 96-07 Revision 1, “Guidelines for 10 CFR 50.59 Evaluations,” November 2000.

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.

U.S. Nuclear Regulatory Commission, “NRC Regulatory Issue Summary 2002-22: Use Of EPRI/NEI Joint Task Force Report, “Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule,” November 25, 2002.

U.S. Nuclear Regulatory Commission, NRC Information Notice 2010-10, “Implementation of a Digital Control System Under 10 CFR 50.59,” May 28, 2010 (Adams Accession No. ML100080281).

U.S. Nuclear Regulatory Commission, NUREG-0700, Rev. 2, “Human-System Interface Design Review Guidelines,” May 2002 (ADAMS Accession Nos. ML021700337, ML021700342, ML021700371).

U.S. Nuclear Regulatory Commission, NUREG 0711, Rev. 3, “Human Factors Engineering Program Review Model,” November 2012 (ADAMS Accession No. ML12324A013).

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. (ADAMS Accession No. ML042430055).

U.S. Nuclear Regulatory Commission, NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” February 1996 (ADAMS Accession No. ML042430048).

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” November 1973, (ADAMS Accession No. ML003740125).

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.5, Rev. 1, “Quality Assurance Program Requirements for Research and Test Reactors,” June 2010, (ADAMS Accession No. ML093520099).

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.6, Rev. 2, “Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities,” September 2017, (ADAMS Accession No. ML17263A472).

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.152, Rev. 2, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” January 2006 (ADAMS Accession No. ML053070150).

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.152, Rev. 3, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” July 2011 (ADAMS Accession No. ML102870022).

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” November 2000 (ADAMS Accession No. ML003759710).

8 ELECTICAL POWER SYSTEMS

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

8.1 NORMAL ELECTRICAL POWER SYSTEMS

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

- The design bases of the normal electric power system, including how safe reactor shutdown will be ensured if offsite power is lost. (This discussion should address both short- (transient) and long-term electrical outages.)
- The ranges of electrical power capability required, for both reactor operation and utilization, in terms of various principal voltages, currents, wattage, and frequencies.
- Use of a substation, either one devoted exclusively to the reactor facility or a shared service with other activities.
- Special processing of the electrical service by such components as isolation transformers, noise limiters, lightning arrests, or constant voltage transformers.
- Schematic diagrams showing the basic distribution systems and circuits.
- Design and performance specifications of principal components, including any that are unique or not standard.
- Special routing or isolation of wiring or circuits for both reactor operations and experimental facilities. (The description should include provisions for isolating electrical power service from instrumentation and control circuits and safety-related circuits to avoid electromagnetic interference.)
- Any deviations or exceptions from national or local electrical codes.
- Technical specifications, if required, with bases that ensure the operability of the normal electrical service, including surveillance requirements.

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

8.2 EMERGENCY ELECTRICAL POWER SYSTEMS

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

- Power for reactor power level monitors, recorders, and necessary safety-related instruments.
- Power for effluent, process, and area radiation monitors, including recorders.
- Power for physical security control systems, information systems, or communications. (In this section, the applicant should only mention the existence of such emergency electrical power and should confine details to the facility physical security plan.)
- Placing or maintaining experimental equipment in a safe condition.
- Power for active confinement or containment engineered safety feature (ESF) equipment and control systems, such as blowers, fans, or dampers, and heating, ventilation, and air conditioning equipment. (This is the equipment necessary to maintain equipment and personnel habitability or to control concentrations or release of airborne radioactive material and to mitigate accident consequences.)
- Power for coolant pumps or systems, including instrumentation and control systems.
- Power for trace heating to prevent fuel salt or primary cooling system salt from freezing.
- Power for other ESF equipment, if applicable.
- Power for emergency area lighting and communication equipment.
- Power for those instrument and control systems necessary to monitor reactor shutdown. (These could include fuel salt flow, fuel insertion, fuel salt cleanup system, fuel temperature, control element positions, or fission product monitors.)

The types of emergency electrical power systems discussed in this section should be commensurate with the required design bases develop in other chapters of the SAR, such as Chapter 4, “Molten Salt Reactor Description”; Chapter 5, “Molten Salt Reactor Cooling 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. If required, these systems may range from automatic start generators to wet-cell or dry-cell batteries. In this section, the applicant should justify why an emergency electrical system is not necessary. Otherwise, the applicant should present a detailed functional description and circuit diagrams. In the design bases, the applicant should discuss if non-interruptible electrical power is required in the transfer from normal to emergency electrical service and if the transfer is manual or automated. The design bases should also provide voltage and power requirements for the emergency electrical power systems, the time duration over which these could be needed, and assurance that fuel will be available for the time required. The designs of the emergency electrical systems should provide that any use for non-safety-related functions could not cause loss of necessary safety-related functions. The design discussion should show how the emergency power supply system is isolated or protected, if necessary, from transient effects, such as power drains, short circuits, and electromagnetic interference. If the emergency electrical power systems are required during analyzed accidents, the designs should include this capability. The minimum emergency electrical power functions that would be required to protect the health and safety of the public should be included in technical specifications based on the discussion in Chapter 8. The technical specifications should also identify the minimum equipment to be supplied by the emergency power system, important design parameters, and surveillance and inspection functions that ensure operability of the emergency electrical power systems and the supplied equipment.

9 AUXILIARY SYSTEMS

Chapter 9 of this guide is applicable to providing a description of the auxiliary systems for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should discuss the auxiliary systems at the MSR. 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, facility staff, and the environment. The applicant should provide sufficient information for all auxiliary systems to support an understanding of the design and functions of the systems, with emphasis on those aspects that could affect the reactor and its safety features, radiation exposures, and the control or release of radioactive material.

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

(1) Design basis.
(2) System description, including drawings and specifications of principal components and any special materials.
(3) Operational analysis and safety function.
(4) Instrumentation and control requirements not described in Chapter 7, "Instrumentation and Controls Systems," of the SAR.
(5) Required technical specifications (TS) and their bases, including testing and surveillance.

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

- Heating, ventilation, and air-conditioning (HVAC) systems for normal reactor operation. (The applicant should discuss any engineered safety feature functions of the HVAC systems for accident conditions in Chapter 6, "Engineered Safety Features," of the SAR.)
- For reactors designed for homogeneous fuel, handling and storage of special nuclear material (SNM) used for reactor fuel, both new and irradiated, including systems (tanks, valves, pumps, instrumentation, controls), related cooling systems, processes (chemical blending, SNM transfers, waste storage, preparation for shipment), criticality control and monitoring, vaults, shielding, and contamination control.
- Fire protection systems that could affect reactor safety or protection of licensed materials.
- Communication systems, both internal and external to the facility.
- Control, storage, or use of byproduct, source, and special nuclear material produced, used, or possessed under the reactor operating license. (The applicant should also discuss applicable laboratory facilities designed to handle or use byproduct materials other than radioactive waste.)
- Gas management system control and processing. (The applicant should include features of MSRs designed to control fission gases.)
- Fission gas collection and storage systems in homogeneous-fueled reactors where fission gas generated in the fuel salt is collected and directed to a radioactive waste treatment system. Details of this system are discussed in Chapter 11, "Radiation Protection Program and Waste Management," of the SAR. The applicant should demonstrate that the auxiliary system and any malfunction could not create conditions or events that could cause an unanalyzed accident or the uncontrolled release of radioactive material beyond those analyzed in Chapter 13, "Accident Analysis," of the SAR.
- Auxiliary cooling systems for irradiated fuel storage tanks that are not part of cooling systems described in Chapter 5, "Molten Salt Reactor Cooling Systems," for example experimental

facilities, gas management system cooling, flush and drain tank cooling, and other equipment and uses that are not part of the primary cooling system described in Chapter 5, of the SAR.

- Demineralizer resin regeneration system and other filtration systems.
- Control and storage of radioactive waste and reusable radioactive components (e.g., experiments). (If applicable, the applicant should describe the systems and show how they are designed to perform the design-basis functions derived in Chapter 10, “Experimental Facilities and Utilization,” or Chapter 11 of the SAR.)
- Control of contaminated air, gas, or liquid from experimental facilities (If applicable, the applicant should describe the systems and show how they are designed to perform the design-basis functions derived in Chapters 10 or 11.)
- Compressed air or gas systems for reactor operating systems, fuel transfer, and experiment equipment.
- Auxiliary physical protection and access control that are not part of the facility physical security plan.

These examples are not intended as a complete list of auxiliary systems that might be discussed in this chapter of the SAR. The descriptions of some auxiliary systems might be better suited to other chapters, which should be referenced in this section.

9.1 HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEMS

All used spaces in a facility could require HVAC systems to provide acceptable environments for personnel and equipment. In this section, the applicant should describe how temperature and humidity are controlled and discuss the bases, including how the control function is integrated into the HVAC systems. The applicant should address the prevention of uncontrolled releases of airborne radioactive effluents to the environment for normal operation. Special consideration should be given to a reactor with a homogeneous-fueled active reactor core because of the concentrations of fission gas in the fuel. Chapter 5 of the SAR addresses numerous systems unique to liquid salt–fueled reactors that should be considered if applicable. The discussions should contain explanations of how airborne radioactive material from operations and experiments is limited in occupied areas to maintain radiation exposures below the requirements of 10 CFR Part 20 and the facility ALARA (as-low-as-is-reasonably-achievable) program guidelines. Controls limiting diffusion or leakage of radioactive material to adjacent spaces should be presented. The applicant also should discuss how air exhaust systems or stacks are designed to reduce the radiological impact on the unrestricted environment during normal reactor operations.

Analyses of radiation exposures in Chapter 11 should include the applicable normal operating characteristics of the HVAC systems described in this section of the SAR. The interactions among airflow patterns in the reactor room, the air exhaust stacks, and the effluent and continuous air monitors should be discussed. If the HVAC systems also are designed to mitigate the consequences of accidents, the engineered safety features should be noted in this section of the SAR but described in detail in Chapter 6.

The applicant should describe instrumentation and control systems that control the release of radioactive material (automatic and manual) in Chapters 7 and 11 of the SAR. The information in this section of the SAR should be sufficient to support an understanding of the safety functions of radiation sensors that initiate alarms and automatic closures, fail-safe dampers, interlocks, and function displays during normal reactor operations. The applicant should discuss the bases and purpose of TS that apply to the HVAC systems, including calibrations, testing, and surveillance.

The applicant should discuss the possible effects of malfunctions of the HVAC systems on safe reactor operation or on the release of airborne radioactive material during normal reactor operation. The radiological effects of malfunctions should be discussed in Chapter 11.

9.2 HANDLING AND STORAGE OF REACTOR FUEL

In this section, the applicant should discuss the life cycle of reactor fuel from the time it enters its jurisdiction until it is released from such jurisdiction. For most reactors this means from arrival on-site until shipment off-site. The safety and performance of the fuel while in the active reactor core is discussed in Chapter 4, "Molten Salt Reactor Description," of the SAR.

MSRs operate with minimal excess reactivity. Fuel or fuel salt is added as necessary to continue operation. The applicant should describe the various systems and operations involving fuel and fuel salt injection into the vessel. Fuel insertion may utilize the fuel salt makeup system as described in Section 5.5.

For facilities designed to fabricate fuel on-site, the discussion should include a description of the form in which the fissile material is received, how and where it is stored as it is made ready for use and when ready for use, any purifications systems required, and how it is blended into a useable fuel salt, including criticality control measures and monitoring. The applicant should provide analyses and discuss how subcriticality is ensured (k_{eff} not to exceed 0.90) under all conditions, except during transportation off-site. During transportation, the shipping container license is applicable. (Existing usage with k_{eff} greater than 0.90 will be acceptable if the usage was previously reviewed and approved by the NRC.) The applicant should address the applicability and implementation of 10 CFR 70.24, which addresses criticality monitors. NUREG-1520, Revision 2, and ANSI/ANS 8.24 provide guidance on nuclear fuel cycle facilities and critical or subcritical conditions for nuclear criticality safety analyses.
The applicant should address the nuclear and chemical stability of materials subject to long-term exposure to irradiated fuel salt stored on-site (e.g., k_{eff} changes resulting from a heat-induced recombination of gas in a highly irradiated fuel salt storage tank). Container or tank corrosion and off-gassing considerations for irradiated fuel salt stored on-site should also be addressed.

The applicant should discuss briefly the methods that ensure the prudent control of fuel. The discussion should include a description that does not contain proprietary or safeguards information of the physical protection of fuel against theft or diversion in the facility physical security plan. Reference can be made to the physical security plan, which should be treated as proprietary or safeguards information.

The applicant should describe the various systems and operations involving the fuel and fuel salt such as receipt from off-site (if applicable), storage, melting, solidification, dissolution, blending, fuel makeup and removal, fission heat extraction, fuel cleanup, and draining of the system. However, if an extensive fuel cleanup process is part of the design, that material might fit better in Chapter 11.

Irradiated fuel cooling systems and methods may be described in detail in Chapter 5 of the SAR if they are integral to the primary cooling system or the heat dissipation system. Otherwise, the discussion should be in this section.

During fuel salt storage or handling, if a loss of fuel, fuel-handling accident, or system failure could result in the release of fission products, the applicant should discuss the mechanisms and analyze the consequences in Chapter 13, "Accident Analyses," of the SAR.

The applicant should include in its discussions of fuel handling and storage the bases of related technical specifications, including inspections, testing, and surveillance and applicable administrative controls and procedures.

9.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

In this section, the applicant should describe the systems and programs designed to protect the reactor facility from damage by fire and discuss how the facility meets all local building and fire codes. For a new facility, this could be a general discussion of how the facility meets local fire and building codes. Documentation from the local authority that authorizes the construction or verifies compliance with local codes could be submitted as part of the discussion. NRC construction inspectors would review design features for fire protection during facility construction. For existing facilities requesting license renewal for which the original construction documentation might be difficult to reproduce, the applicant could submit the results of a recent fire inspection to show compliance with local codes. The applicant should discuss additional active and passive design features required by the reactor design characteristics. Further, the discussion should address the potential for release of radioactive material as a result of a fire. Active systems might include sprinkler, suppression, hand extinguisher, and detection systems. Passive systems might include fire walls and doors, isolation, and control of combustible materials. Three important considerations for MSRs are:

1. Fire suppression with water in areas with high temperature components creates a potential for rapid pressurization (i.e., steam hazard).
2. Water always creates the potential for a criticality excursion around fissile material.
3. A properly written fire hazards assessment contains a discussion on the control of run off from fire suppression. In the case of MSRs, water is probably the most efficient transport mechanism for fuel salts.

The applicant should discuss how the potential release of radioactive materials as a result of fires in the reactor room and other applicable spaces was considered in the design of the facility. The discussion should include the reactor and all facilities where special nuclear material and other radioactive materials are stored or used under the reactor license. It should include any possible effects of a fire on safe shutdown of the reactor. The objectives of the fire protection program should include the following:

- Preventing fires, including limiting combustible materials.
- Detecting, controlling, and extinguishing fires to limit consequences.
- Protecting reactor systems so that a fire would not prevent safe reactor shutdown, result in an inadvertent criticality, or cause an uncontrolled release of radioactive material.

The applicant should discuss the bases of any technical specifications, including testing and surveillance, as they relate to the fire protection systems and programs. The discussion should also include the relationship between fire protection plans, operating procedures, and the facility emergency plan.

The National Fire Protection Association (NFPA) standard 802, 1993 edition, contains general information on research reactor fire protection. The applicant may also consult the NFPA performance-based standard for LWRs, NFPA 805, 2015 edition, or the performance-based standard for advanced reactors, NFPA 806, 2015 edition.

9.4 COMMUNICATION SYSTEMS

The applicant should describe the communication systems that will be used at the facility for which public disclosure is not limited by the physical security plan. Communication systems used between the control room, the reactor room, reactor access point or top, reactor utilities rooms, experiment areas, and all other required areas should be described. Systems such as telephone, paging, radio, or video that will be used to announce changes of reactor status to experimenters, summon supervisory operators, request radiation protection assistance, and announce emergencies should be discussed. For a complete description of communications, the applicant should also briefly summarize in this section the communication systems used for emergency or physical security purposes (this discussion should not contain proprietary or safeguards information).

The discussions of communication systems should include the bases of any related technical specifications, including testing and surveillance.

9.5 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

The 10 CFR Part 50 operating license applies to possession and operation of the reactor; possession and use of byproduct material produced by the operation of the reactor; and, to the extent authorized, the receipt, possession, and use of other byproduct, source, or special nuclear material needed for operation of the reactor and its experimental programs.

The NRC regulatory approach is to include in the reactor license only material that is produced by the reactor or that is required to directly operate the reactor and associated experimental facilities. Other material at a reactor facility is authorized by an NRC byproduct, source, or special nuclear materials license. If the facility is located in an Agreement State, an Agreement State license could also exist. This other material is normally not required to operate the reactor or associated experimental facilities.

The receipt, possession, or use of materials authorized by the reactor license may occur in the reactor room and contiguous operational spaces and in laboratory spaces for research and development purposes. Some licensees take a narrow view, transferring material produced in the reactor to another NRC or Agreement State license when the material is removed from the reactor or experiment facilities. Others take a broad view and allow all materials produced by the reactor or authorized by the license to be in various locations and laboratories in the facility. Spaces could be used to process and package byproduct materials for shipment or could be used for performing experiments involving the byproduct materials. A broad view of materials and areas authorized by the 10 CFR Part 50 reactor license avoids maintaining multiple licenses and allows, in some cases, indemnity protection for materials in laboratories and other auxiliary spaces. The applicant should clearly state the materials and areas of the facility requested to be authorized by the reactor license. The reactor license and TS will also include regulatory conditions that apply to the possession, management, and use of such materials, including requirements stated in 10 CFR Parts 20, 30, 40, or 70.

The applicant should discuss in this section laboratories under the reactor license in which reactor-licensed material will be used. This discussion should address all five factors noted at the beginning of this chapter for any such auxiliary laboratories. The applicant should specify the types and quantities of radionuclides authorized, as well as the general types of experiments or uses. Radiological design bases for handling radioactive materials and radioactive waste should be derived from Chapter 11 of the SAR. These design bases may apply to chemical, fume, and air exhaust hoods; drains for radioactive liquids; and radiation shields. The discussions should show how the physical security and emergency plans apply to the licensed spaces and possession of byproduct materials. The applicant should discuss the bases for special operating procedures. The administrative aspects of the use of materials in these areas should be addressed in Chapter 12, “Conduct of Operations,” of the SAR.

The use of fuel salts add numerous new issues pertaining to quantification of byproduct, source, and special nuclear material unique to operation of a homogeneous reactor. The applicant will have described in Chapter 4 of the SAR, the fuel salt cycle, its salient features being the following:

- Fuel salt has no start and end date in the homogeneous reactor analogous to each fuel element in a heterogeneous reactor where a fuel element is placed in the reactor for a defined period of time and then removed for ultimate shipment off-site during which the history of its location in the active reactor core, the energy extracted from it, and the transformative reactions (fission, neutron activation, radioactive decay) resulting in new materials (fission fragments, activation products, conversion to fissile material) can be tracked.

- The intent of the fuel salt concept is to replenish fissioned material with more fissile material than what is consumed in the fission process; replacing fissioned ^{235}U with ^{239}Pu resulting from the neutron capture of ^{238}U.
- The initial fuel salt–loading fissile material for a reactor could consist of ^{235}U only, ^{235}U and ^{239}Pu, or a ^{233}U/Th fuel cycle.
- Knowing the concentration and quantity of byproduct, source, and special nuclear material in the fuel salt at any time requires a complex calculational or measurement program or a combination of the two.

The applicant should discuss the bases of any technical specifications, including testing and surveillance, as they relate to the possession and use of byproduct material, source, or special nuclear material.

9.6 GAS MANAGEMENT SYSTEM

For reactors with gas management systems, the applicant should discuss control of the cover gas and all decay heat removal components, addressing the five factors listed at the beginning of this chapter. The discussions should describe gas management auxiliary systems that cool, circulate, decontaminate, recover, store, monitor, and dispose of the gas. Processing, storing, and recombining of reactive gases, if applicable, should also be discussed. The design bases should define which inert gases are acceptable to use, their impact on safe reactor operations and shutdown, and the methods for controlling the concentrations.

Gaseous fission products from MSRs must be collected in a gaseous fission products gas management system to protect workers from high radiation doses. The gas must be further treated to isolate it until decayed. This might be done through delay lines/tanks, cryogenic storage, or another delay-before-release system. The discussions should include the bases of any required technical specifications applicable to gas management systems, including testing and surveillance.

9.7 COOLING SYSTEMS

Among the auxiliary systems that should be addressed are any cooling systems that are part of the licensed facility. Chapter 5, “Molten Salt Reactor Cooling Systems,” identifies the following cooling systems that could be associated with an MSR:

- Fuel salt drain tank
- Primary cooling system drain tank, if applicable
- Gas management system cooling
- Cooling for chemical processing/polishing loop
- Other cooling systems

The applicant should describe and analyze each cooling system, addressing the five factors listed at the beginning of this chapter, and include the following:

- Demonstrate that the cooling system will function under normal operation and analyzed reactor accident conditions, if required.

- Demonstrate that the cooling system and any malfunction could not create conditions or events that could cause an unanalyzed reactor accident or the uncontrolled release of radioactive material beyond those analyzed in Chapter 13 of the SAR.
- Demonstrate that the cooling system could not prevent safe reactor shutdown.

The applicant should discuss the bases of any technical specifications, including testing and surveillance, as they relate to cooling systems.

9.8 OTHER AUXILIARY SYSTEMS

As noted previously, a unique set of auxiliary systems could exist at a non-power reactor. The previous examples are found at many reactor facilities; other facilities could have additional auxiliary systems. The applicant should describe and analyze all auxiliary systems, address the five factors listed at the beginning of this chapter, and include the following:

- Demonstrate that the auxiliary system will function under analyzed reactor accident conditions, if required.
- Demonstrate that the auxiliary system and any malfunction could not create conditions or events that could cause an unanalyzed reactor accident or the uncontrolled release of radioactive material beyond those analyzed in Chapter 13 of the SAR.
- Demonstrate that the auxiliary system could not prevent safe reactor shutdown.

The applicant should discuss the bases of any technical specifications, including testing and surveillance, as they relate to other auxiliary systems.

9.9 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 8.24, “Validation of Neutron Transport Methods for Nuclear Criticality Safety,” ANS, LaGrange Park, Illinois, 2017.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, “The Development of Technical Specifications for Research Reactors,” ANS, LaGrange Park, Illinois, 2013.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21, “Format and Content for Safety Analysis Reports for Research Reactors,” ANS, LaGrange Park, Illinois, 2012.

National Fire Protection Association, NFPA 802, “Recommended Practice for Nuclear Research and Production Reactors,” 1993.

National Fire Protection Association, NFPA 805, “Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants,” 2015.

National Fire Protection Association, NFPA 806, “Performance-Based Standard for Fire Protection for Advanced Reactor Electric Generating Plants Change process,” 2015.

U.S. Nuclear Regulatory Commission, NUREG-0849, “Standard Review Plan for Fuel Cycle Facilities License Applications,” Rev. 2, June 2015. (Adams Accession No. ML15176A258)

10 EXPERIMENTAL FACILITIES AND UTILIZATION

Chapter 10 of this guide is applicable to providing a description of any experimental facilities associated with a non-power MSR. In this chapter of the SAR, the applicant should describe and discuss the experimental facilities at the non-power MSR, their intended use, and the experimental program. This chapter of the SAR should contain a description of the proposed experimental program and the safety analyses for each type of experimental facility. The design, construction, and placement of each experimental facility should be analyzed for inherent safety questions that exist apart from the experiments accommodated therein. The experiments should be analyzed by using a separate experiment safety analysis methodology to show compliance with the technical specifications, primarily the associated limiting conditions for operation (LCOs) as indicated in Chapter 14, "Technical Specifications," of this format and content guide. The applicant should provide sufficient information to demonstrate that no proposed operations involving experimental irradiation or beam utilization will expose reactor operations personnel, experimenters, or the general public to unacceptable radiological consequences. In addition to the guidance in this format and content guide, Regulatory Guide 2.2 and ANSI/ANS 15.1 contain guidance on technical specifications and experimental programs that may be useful to the applicant in preparing the SAR.

All non-power reactors may be used for many purposes including radiation physics, chemistry and biology studies, materials irradiation, radionuclide production, and educational purposes. The experimental facilities may penetrate that active reactor core or reflector or be located near the active reactor core. Neutron or other radiation beams can be extracted from the active reactor core region through the biological shield. At many non-power reactors, the experimental facilities are integral components of the entire reactor.

In addition to traditional experimental purposes, as described above, a non-power MSR may be used to gather information and data that could be useful for the purposes of licensing future prototype facilities and power reactors. In this case, specific experimental facilities may not be included as part of the MSR. However, new or unique SSC could be demonstrated by the MSR. Special safety features and added instrumentation for these SSC should be discussed in this chapter of the SAR. QA programs for new or unique SSC being demonstrated by the MSR should be referenced here and discussed in Chapter 12, "Conduct of Operations."

In addition, the reactor itself can be considered an experimental facility to demonstrate MSR technology for eventual prototype and commercial scale up. When the reactor is the experiment, the safety analysis and reactor technical specifications will include appropriate operational limitations.

Utility, integrity, longevity, versatility, diversity, and safety should be considered for the experimental facilities in the same manner they are considered for the active reactor core and its operational components and systems. Therefore, the safety analyses of the reactor facility should include the experimental facilities and their interactions with the reactor components and systems. If changes in reactor operating characteristics are considered, potential interactions between the active reactor core and the experimental facilities should be analyzed.

Experimental programs and the range of experiments vary widely among non-power reactor facilities. Furthermore, as the licensee and the facility user gain experience and as technology develops, the experimental program and many of the specific experiments may change over the life of the reactor. This makes it very difficult and impractical for the applicant to describe specific experiments in the SAR. The applicant should describe and analyze in this chapter of the SAR and incorporate into the facility technical specifications enveloping conditions of experiment attributes such as reactivity limits or material properties to allow the greatest flexibility in the experimental program. Potential experimental needs should be considered when establishing these limiting safety aspects in the SAR, so that determinations in accordance with 10 CFR 50.59 can be made expeditiously. Experience has shown that most licensees have successfully implemented changes in experimental programs without prior NRC approval under the provisions of 10 CFR 50.59. This regulation allows licensees to 1) make changes in the facility as described in the SAR, 2) make changes in the procedures as described in the SAR, and 3) conduct tests or experiments not described in the SAR without prior NRC approval, unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. A proposed change, test, or experiment is deemed an unreviewed safety question if 1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, 2) a possibility for an accident or malfunction of a type different from any evaluated previously in the SAR may be created, or 3) the margin of safety as defined in the basis for any technical specification is reduced.

The applicant should provide an analysis to demonstrate that the reactor and experimental facilities can be operated safely. This analysis should include the range of normal operations, accidents, and malfunctions of experimental facilities. It should address any impact the experimental facility imposes on the reactor and any impact the reactor imposes on the experimental facility.

Consideration should be given to the possibility of the experimental facility causing an accident that requires analysis in Chapter 13, “Accident Analyses,” of the SAR. In some cases, the failure of an experiment can be the maximum hypothetical accident (MHA) for the reactor. This possibility is most prevalent with fueled experiments. Experiments can result in the maximum uncontrolled reactivity addition accident at a facility. Limiting experiment failure should be considered in Chapter 13.

The SAR should be written to accommodate the nature of varying experiments and meet the requirements of future experimentation. The applicant should show that there is no undue risk to the health and safety of the public.

Discussions in this chapter of the SAR should include design bases, facility descriptions, functional and safety analyses, and the applicant’s safety conclusions for all experimental facilities. The structural design and its potential impact on reactor operation should be analyzed for those experimental facilities that are permanently attached to the reactor support structure or the reactor vessel. The placement or use of experimental facilities shall not compromise the functionality of any reactor safety system or engineered safety feature. The discussion should include the capabilities, limitations, and controls for experiments, that ensure radiation doses do not exceed the requirements in 10 CFR Part 20 and are consistent with the facility program to keep exposure to radiation as low as is reasonably achievable (ALARA).

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

10.1 SUMMARY DESCRIPTION

In this section of the SAR, the applicant should briefly describe the principal features of the experimental and irradiation facilities associated with the reactor. The applicant should discuss the scope of the experimental program and define what is considered to be an experiment. Discussions should include experimental compatibility with normal reactor operations and accidents and measures taken to avoid interference with the reactor shutdown and other systems.

The applicant should include the following information:

- General focus of the experimental program (radiation science, medical, materials testing, teaching, etc.)
- A list of experimental facilities
- Basic type of experiments that will be conducted (in-vessel, thermal column, external beam, etc.)
- Limiting experimental characteristics (e.g., reactivity, contents)
- A brief description of experiment monitoring and control and the interaction between the experiment and the reactor control and safety systems
- A brief overview of design requirements for the experiment and of the review and approval process

Simple block diagrams and drawings may be used to show the location, basic function, and relationship of each experimental facility to the reactor. The summary description should contain enough information to support an overall understanding of the functions of the experimental facilities and the experiment review and approval process.

A brief description of typical experimental facilities found at non-power reactors follows. This list, however, is not exhaustive. Some of the listed facilities may be unlikely for most MSRs but are included for consideration.

- *In-vessel facilities* are those facilities that are surrounded on at least two sides by fuel salt. Such facilities are commonly called void tubes, flux traps, central irradiation facilities, in-vessel irradiation facilities, radioisotope facilities, fast and thermal neutron irradiation facilities, or central and offset thimbles.
- *In-reflector facilities* are those facilities that are physically located in the reflector and are surrounded either on all sides or on at least three sides by reflector material. In-reflector facilities might include lazy susans, void tubes, flux traps, thimbles, standpipes, or thermal neutron irradiation facilities.
- *Automatic transfer facilities*, sometimes called rabbits, are a special class of in-vessel and in-reflector experimental facility. They often protrude into or are adjacent to the active reactor core or reflector and contain the experimental material. However, rabbit facilities allow the experimental material to be moved quickly into and out of the desired flux region of the active reactor core by pneumatic, hydraulic, or mechanical means. The material can be moved while the reactor is operating if limits on reactivity changes in the reactor are observed.
- *Beam ports* are hollow tubes that can abut the active reactor core or protrude into the active reactor core or reflector. However, unlike the previously described in-vessel and in-reflector facilities, they may or may not contain the experimental material. Instead, they may be used to channel radiation from the active reactor core to a position, usually outside the reactor vessel and the biological shield, where the experiment is located. Neutrons and gamma-ray beams are tailored to suit the experimental needs.
- *Thermal columns* function in a way similar to that of beam ports in that they allow transport of radiation away from the active reactor core to areas where the experiment is located. Rather than a tube to guide radiation beams, they consist of a neutron moderator, typically a large volume of graphite blocks, enclosed in a container. The column is located at one face of the reactor in place of the reflector. Fast neutrons are thermalized within the moderator and may be used outside or inside the reactor shield for experiments.

- *Irradiation rooms* or other dry cavities in the biological shield may be located adjacent to the active reactor core for irradiation of large volumes of material or objects.

10.2 EXPERIMENTAL FACILITIES

In this section of the SAR, the applicant should describe and discuss in detail all experimental facilities. The design should ensure that risks to the public, staff, and experimenters are acceptable.
The applicant should discuss specifications and important design and operating parameters for the experimental facilities and give design details and the physical size, including all dimensions. Simplified engineering drawings or schematics may be used, especially for more complex facilities. The applicant should discuss the location of the experimental facility in relation to the active reactor core, safety systems, active reactor core support, neutron detectors, MSR cooling system components, and any other reactor systems, components, or structures.

Features of the experimental facility that could interfere with safe reactor shutdown or with adequate active reactor core cooling shall be included. The source of experiment cooling and any dependence on or interaction with MSR cooling systems should be discussed. For any experimental facilities that require a special cooling system independent of the fuel salt, primary cooling system, or heat dissipation system; the technical evaluation considerations are similar to those for the MSR cooling systems. The applicant should follow the applicable guidance in Chapter 5, "Molten Salt Reactor Cooling Systems," in this format and content guide for independent experiment cooling systems.

Since integrity of the experimental facility is important, the capability to contain or withstand any postulated pressure pulse and preclude any inadvertent coolant leakage, fuel salt leakage, or facility collapse should be discussed.

The applicant should discuss the materials used in the construction of the experimental facilities, addressing radiation and chemistry impacts. Materials and design, including physical dimensions, should limit any rapid reactivity insertion if the facility is suddenly voided or flooded with fuel salt or primary cooling system salt. The supporting analysis should be included in Chapter 13 of the SAR, where change in the limiting experiment failure reactivity should be analyzed. The bases of applicable LCOs for the technical specifications should be developed and justified.

The radiological considerations associated with the design and use of the experimental facilities, generation of radioactive gases, release of fission products or other radioactive contaminants, and exposure of personnel to neutron and gamma beams should be summarized in this section of the SAR and discussed in greater detail in Chapter 11, "Radiation Protection Program and Waste Management." Direct radiation streaming from the experimental facilities and the effect of scattered (sky shine) radiation should be discussed briefly in this section of the SAR and analyzed in Chapter 11. The analysis should clearly show all pertinent radiation sources, distances, dimensions, materials, radiation scattering, and material attenuation factors.

Facilities that could fail and release airborne radioactivity into the facility air or to the environment should be analyzed. The analysis in Chapter 13 of the SAR and summarized in this section should show the concentrations of radioactive material in the experimental facility, the release pathway, and the concentrations of radioactive material in the reactor facility and the outside environment. In some cases, this type of failure could be the MHA for the reactor, which should be analyzed in Chapter 13.

Any radiation monitors specifically designed and placed to detect experiment radiation and to monitor personnel should be discussed briefly in this section of the SAR and discussed in greater detail in Chapters 7, "Instrumentation and Control Systems," and 11. Additionally, reactor operating characteristics, including scrams and runbacks associated with experimental measurements, should be analyzed.

Any remote access equipment stations, physical restraints, shields, or beam catchers, both temporary and permanently installed, that are used to restrict access to radiation areas associated with experimental facilities should be described and analyzed. Descriptions and analyses should show that the placement, dimensions, and materials 1) are sufficient to limit the expected radiation doses to experimenters, reactor operators, and other personnel to levels below those specified in 10 CFR Part 20 and 2) are consistent with the facility ALARA program. For reactor beams, the applicant should describe the approach to compliance with the regulations concerning access to high radiation areas and very high radiation areas, as appropriate. These issues should be analyzed in Chapter 11 and summarized in this section of the SAR. Permanently installed safety instrumentation for the experiment facility, including the location and function of sensors, readout devices, and scram or interlock capabilities, should be discussed briefly in this section of the SAR and in greater detail in Chapter 7.

Technical specifications for experimental facilities, as discussed in Chapter 14 of this format and content guide, should be presented and justified in this section of the SAR.

10.3 EXPERIMENT REVIEW

Because of the variety of experiments that can be conducted in a non-power reactor, the administrative controls of the applicant should be adequate to ensure the protection of the public. The administrative procedures used by the applicant to review and approve experiments should be described in detail in this section of the SAR and summarized in Chapter 12, “Conduct of Operations,” and the operating limits should be included in the technical specifications. The applicant should state the safety analysis requirements for the experiment safety analysis report and the experiment review and approval methodology and should briefly discuss the authority and role of the experiment review committee.

The applicant should discuss experiment classification and approval authority. The applicant should state the methodology used to categorize proposed experiments according to risk potential, the categories expected at the reactor facility, and the safety requirements for each category. The methodology should describe how 10 CFR 50.59 will be used in the review of all experiments not described in the SAR, as well as how Regulatory Guide 2.2 and ANSI/ANS 15.1 will be used. The appropriate level of review authority required to approve experiments in each category should be discussed. The applicant should be specific in delineating the bounds of the risk categories, such as gram amounts, temperature degree limits, radioactivity limits, or reactivity limits, and should develop the bases of applicable technical specifications. The experiment safety analysis process should demonstrate compliance with these limits and establish any special controls on the experiment.

The applicant should discuss administrative controls for the experiment and list the administrative controls used to protect facility personnel and the public from radiation or other possible hazards, such as chemical releases, in the implementation of the experimental program. Where appropriate, the discussion should delineate areas where reactor operations and experiment operations are performed under separate authority and by different personnel. The discussion should include access to experimental facilities and areas, lockout procedures, communications with reactor operating personnel, alarms, and reactor scrams. The administrative procedures should address basic protection and recovery procedures after a malfunction of experiments or experimental facilities.

The applicant should discuss the generic safety assessment of experimental materials and limitations, consistent with the guidance in Regulatory Guide 2.2, from which experiment and reactor LCOs are incorporated in the technical specifications. Malfunctions or failures of experiments with significant potential for radiological consequences should be analyzed in Chapter 13 of the SAR and summarized in this section. For some reactors, the most serious accident or the MHA could be initiated by an experiment malfunction. Areas of assessment should include the following:

- Fissile materials and radiological risks from radiation fields or release of radioactive material
- Trace elements and impurities
- Effects on reactivity, both positive and negative
- Explosive, corrosive, and highly reactive chemicals
- Radiation-sensitive materials
- Flammable toxic materials
- Unknown materials
- Radiation heating or damage that could cause experiment malfunction
- Heating that could cause departure from nucleate boiling on surfaces

10.4 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 2007 (R2013).

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

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

Chapter 11 of this guide is applicable to providing a description of radiation protection and waste management programs for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should discuss and analyze all radiological consequences related to normal operation of the MSR. In general, the design and function of structures, systems, and components (SSC) and all facility operations and materials authorized by the operating license should be described in detail in other chapters of the SAR. Chapter 11 should contain the principal discussions of the facility program to control radiation and expected exposures due to operation, maintenance, and use of the reactor. In this chapter, the applicant should develop the methods for quantitative assessment of radiation doses in the restricted, controlled (if present), and unrestricted areas; should apply those methods to all applicable radiation sources related to the full range of operation; should describe the program and provisions for protecting the health and safety of the public (including workers) and the environment; and should provide the bases for analyzing radiological consequences from potential accidents addressed in detail in Chapter 13, "Accident Analyses."

In accordance with 10 CFR 20.1101, "Radiation Protection Programs," it is the responsibility of the applicant to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the regulations in 10 CFR Part 20. To the extent practicable, the applicant should also use procedures and engineering controls based on sound principles of radiation protection to keep doses to occupational workers and members of the public as low as is reasonably achievable (ALARA) as required by 10 CFR 20.1101(b).

Waste materials resulting from maintenance, normal operations, or accident conditions at non-power MSRs may contain radioactive isotopes and special nuclear materials (SNM). Such wastes are governed by the operating license, and, like other licensed materials, they must be controlled. At a non-power reactor, management and control responsibility for radioactive waste may be assigned to the organization responsible for reactor operations, and the radiation protection organization may provide independent oversight for monitoring, assessing, and limiting risks related to radiation sources. Alternatively, facility management could assign primary responsibility for handling and disposing of radioactive wastes to the organization responsible for radiation protection. In either case, the applicant should require procedures to ensure that radiation exposures and releases of radioactive material are adequately assessed and controlled. The applicant should discuss these issues and submit the information necessary for NRC review. This format and content guide for Chapter 11, of the SAR integrates radioactive waste management and radiological protection in some sections and provides separate sections for some information. The applicant should organize the functions and present the information as best suits the facility consistent with this guide.

Waste management for non-power MSRs has significant differences from conventional light water reactors (LWRs). In an MSR, fission products are released to the liquid salt fuel solution and contained by the fuel barrier. Inherent in the design and operation of the MSR is the need to remove gaseous fission products for pressure control, high cross section fission products for reactivity control, and other constituents in the fuel salt (corrosion products, noble metals, oxidizers, etc.) for chemistry control. An application for a license or an amendment should explain why such fuel cleanup does not constitute separation of special nuclear material.

Noble gasses, volatile fission products, and gaseous neutron activation products will be gathered and processed within the radionuclide barrier and may require holdup for decay or further treatment before being released to the environment or disposed as waste.

Residue from the cleanup and polishing of the fuel salt will be laden with fission products and will likely require treatment as radioactive waste. The applicant should address these unique aspects of waste management for the MSR in the manner discussed throughout this chapter, differentiating these cleanup activities from processes that have the potential for separating special nuclear material from the fission product laden fuel salt.

11.1 RADIATION PROTECTION

The sections that follow provide guidance on the information the applicant should include in the description of the radiation protection program. The program is applied to the design of the reactor and its equipment, the experimental facilities, reactor operations, design and use of associated laboratories, planning and procedures, and the instrumentation, techniques, and practices employed to verify compliance with the radiation dose limits and other applicable requirements specified in the regulations. Plans and the bases used to develop procedures for assessing and controlling radioactive wastes and the ALARA program should be included. The responsibilities of the health physics organization at the reactor facility, as well as any other applicant radiation protection organizations (e.g., under a separate materials license), should be described. Facility organization charts should be included that show independence of the radiation protection function from the facility operations function.

In this chapter, the applicant should address all radiation sources and radioactive materials produced in the reactor and possessed or used within the reactor facility under the authorization of the reactor license. Other byproduct, special nuclear material (SNM), and source material possessed or used under the authorization of the reactor license but not produced by reactor operation should be described. Program details should be given in the sections that follow.

11.1.1 Radiation Sources

In this section, the applicant should describe the sources of radiation that are monitored and controlled by the radiation protection and radioactive waste programs. In general, the sources should be categorized as airborne, liquid, or solid as discussed in the sections that follow.

The applicant should include in this description a tabulation of all standard, check, and startup neutron sources categorized by isotopic composition, principal radiations (e g., alpha, beta, gamma ray and neutron energies, abundance > 10%), radioactivity (curie or becquerel content), neutron characteristics, geometry, physical and chemical form, and whether the source is sealed or unsealed.

The applicant should describe the fuel management program and record keeping of all fissile and fissionable materials, showing the status (fresh, in-use, in interim storage, or spent), original enrichment (including ^{235}U, ^{233}U and total U content), and current enrichment (including current ^{235}U, ^{233}U, total U, and total Pu, *if appropriate*).

Because of the varied nature of experimental programs, the full range of source strengths of irradiated experimental materials are not necessarily tabulated in an SAR. However, the full range of source strengths expected to be encountered in the experimental program should be listed and discussed in the program. Experimental protocols should provide detailed source data and should be subject to the review of facility operations staff, the health physicist, and, in the case of new experiments and specified deviations from previous experiments, the reactor review or audit committee. In evaluating all experiments, the applicant shall also consider the requirements of 10 CFR 50.59, “Changes, tests, and experiments.”

Conservative estimates should be made of the quantities and types of radioactive wastes expected to result from reactor operations and use, based on previous or other similar non-power reactor facility experience, or previous experimental results. Identification of such wastes should distinguish, if possible, which are associated with the operation of the reactor and which are associated with the utilization of the reactor, if utilization occurs under the reactor license. Non-power reactor applicants tend to provide overly conservative estimates; although estimates should be conservative, they should also be realistic.

Where feasible, the applicant should include the physical and chemical form, amounts, use, storage conditions, and locations of all sources. In occupied or accessible areas, conservative estimates of external radiation fields and internal intake should be estimated. An estimate of the maximum annual dose and collective doses to workers and the public should be given for major and repetitive activities involving radiation. The applicant should discuss how the requirements of Subpart C of 10 CFR Part 20 (20.1201-20.1208), which contains regulations for occupational dose limits, and Subpart D of 10 CFR Part 20 (20.1301-20.1302), which contains regulations for radiation dose limits for individual members of the public, will be met. Regulations concerning compliance with dose limits for individual members of the public are given in 10 CFR 20.1302. Applicants that have licensed non-power reactors usually have historical information on radiation doses. They should discuss this information.

License conditions and, if applicable, TS, concerning material possession limits, enrichment, material forms and source strengths, should be developed and analyzed in this and other chapters, such as Chapter 4, “Molten Salt Reactor Description,” of the SAR. These will control the use of the sources discussed above.

11.1.1.1 Airborne Radiation Sources

Airborne radioactive sources should be described in a manner suitable for designing worker protective measures and assessing and controlling workers’ doses. In an MSR, there are multiple sources of gaseous radionuclides to be considered. Gaseous fission products build up in the fuel salt at a rate proportional to the reactor power level. Since the reactor fuel is a liquid, the fission gas migrates directly to the gas space above the vessel and enters the gas management system described in Section 9.6. Volatile actinides and volatile fission products could also migrate into the gas space above the vessel, yielding entrained radioactive particles or aerosols. Radioactive gases created from neutron activation, such as tritium from the neutron activation of certain salts are yet another source of gaseous radiation. Airborne radionuclides are important because they typically are the principal source of radiation exposure to the public from a non-power reactor.

The applicant should summarize in a table the predicted concentrations and quantities of airborne radionuclides during the full range of normal operation (which includes maintenance activities) according to the areas that could be occupied by personnel. The applicant should estimate the release of airborne radionuclides to the environment and should use these releases to determine consequences in the offsite environment. The applicant should discuss compliance with the applicable regulations (10 CFR Part 20). Note that while airborne radioactive sources from accidents are discussed in Chapter 13, the calculational methodologies developed here should be applicable to accident release analysis. Therefore, the models and assumptions used for the prediction and calculation of the dose rates and accumulative doses in both the restricted, controlled (if present), and unrestricted areas should be provided in detail. The guidance that follows gives an example of a description of appropriate methodology that is applicable to airborne and volatile radionuclides, provided the applicant accounts for the associated internal and external dose estimates.

The specific source locations, predicted production rates, release mechanisms and rates, concentrations in occupied areas, possible personnel doses and dose rates, release points from the restricted area, dilution air (quantities and sources), quantities and concentrations predicted to be released, annual average atmospheric conditions, diffusion and dispersion, predicted concentrations in unrestricted areas, and potential dose rates and annual doses, including gamma-ray shine from elevated plumes, should be addressed in detail.

For noble gases at non-power reactors, it is acceptable to assume that all significant radiation risk is from external exposure to beta and gamma radiation. Some radionuclides (e.g., halogens or particulates) could cause internal radiation risk by being ingested or inhaled. All these doses should be addressed, as applicable. The assumptions and methods should be conservative but physically realistic, and the validity of dose calculations should be assessed. Some non-power reactor applicants have used conservative assumptions and methods that have resulted in answers that, although acceptable, are conservative by large factors. The applicant should consider discussing the amount of conservatism built into the calculations. All assumptions should be justified, and sources of information should be adequately referenced. The calculations should address possible doses in the restricted areas, in the controlled areas (if applicable), and in the unrestricted areas. In the unrestricted area, potential doses should be analyzed for the maximally exposed individual, at the location of the nearest permanent residence, and at any locations of special interest, such as a classroom or a campus dormitory. Due care should be taken if finite or non-uniform airborne distributions are intermingled with infinite cloud approximations within buildings or in idealized gaussian plumes. Any such intermingling of models or assumptions should be justified. Similar discussions in this paragraph of the SAR should address the production of airborne particulates, aerosols, vapors, and other radionuclides.

The discussion and calculations should show how the facility design ensures that doses to the facility staff and the public will not exceed 10 CFR Part 20 limits and that its ALARA requirements for effluents are satisfied.

11.1.1.2 Liquid Radioactive Sources

The applicant should identify all expected liquid radioactive sources, such as fuel salt, coolant salt, experimental solutions, reference sources, and fissile material. The applicant should identify their origin and should specify whether they result from reactor operations or the utilization program or whether they exist for special purposes. Information should include radionuclides, concentrations, total radioactivity (curie or becquerel content), solubility, container characteristics, and planned release or disposition. Liquid radioactive wastes should be included. However, since the types of such wastes, their origins, and the source strengths will vary with time and with the nature of the utilization program, only limited descriptions of liquid wastes should be provided. The applicant should estimate the quantity of liquid effluent released to the unrestricted environment. The applicant should discuss if credit is taken for dilution preceding release. The applicant should discuss compliance with the applicable sections of 10 CFR Part 20, such as.10 CFR 20.2003 and any disposal of licensed material approved under 10 CFR 20.2002. Any storage or disposal facilities should be noted, with reference to their management and use and the design basis of their radiation protection capabilities.

11.1.1.3 Solid Radioactive Sources

The applicant should identify all expected solid radioactive sources, such as reactor fuel (spent, in-vessel, and fresh), calibration and test sources, experiment samples, and facility components. The information should include, among other things, radionuclides, radioactivity (curie or becquerel content), and physical characteristics and whether the source is sealed or unsealed. Solid radioactive waste should be noted, but because the types and quantities will vary with time and the utilization program, only limited descriptions of solid wastes need be provided. Provisions for classifying, monitoring, storing, packaging, volume reduction prior to shipment, and disposing of solid radioactive wastes should be discussed. The applicant should estimate the annual volume of solid waste expected to be removed from the site and its radioactive content (in curies). The applicant should discuss compliance with applicable sections of 10 CFR Parts 20, 61, and 71, and Department of Transportation regulations (49 CFR) for transporting radioactive material.

The applicant should discuss any capabilities or approvals received under NRC or State material licenses for onsite or offsite storage of solid radioactive wastes, including how the necessary characteristics of a restricted area are maintained. The applicant should discuss any disposal of licensed material approved under 10 CFR 20.2002.

This section should contain the design bases for temporary, permanent, and installed shielding components at the facility, including utilization, laboratory facilities, and radiation beams.

The following areas of the facility should be examined when developing the program for inventory and control of radiation sources:

- The exterior of the reactor biological shielding and reactor auxiliary locations (e.g., primary cooling system components and demineralizers) accessible to personnel
- The reactor experimental facilities, including beam ports, thermal columns, pneumatic or hydraulic transfer facilities, and all other irradiation facilities
- The radioactive material handling, preparation, packaging, and utilization facilities, including laboratories, hot cells, caves, and storage and processing areas
- Other extraneous sources, including, for example, neutron and gamma irradiation facilities, check and standard sources, neutron sources, fuel handling and storage facilities, experimental equipment storage facilities, and radioactive waste handling and storage facilities

11.1.2 Radiation Protection Program

In this section, the applicant should describe the structure of the organization that administers the radiation protection program required by 10 CFR 20.1101, including information about staffing levels, positions of authority and responsibility, and position qualifications and training required by 10 CFR Part 19. Working relationships with other safety organizations, including the reactor facility operations staff, should be described. The applicant should discuss the charters, standards, procedures, and other documents that specify the authority and responsibilities of the organization, including authority to interdict perceived unsafe practices. The administrative plans and procedures that implement the facility policy, the overall program, and the way the organization, policy, and program are designed for effective operation should be discussed. In this discussion, the applicant should describe the management policy governing the program and the allocation of policymaking responsibilities. Reference can be made to Chapter 12, “Conduct of Operations,” if such information appears there.

The information should include the records and document control measures employed to ensure that the plans and procedures relative to the radiation protection program, including changes, are reviewed for adequacy, approved by authorized personnel, and distributed to and used by the applicable staff at the locations where radiation exposures could be encountered.

The radiation safety training program should be described in detail. This discussion should give the scope, and a summary of the content, of the training provided or required for all personnel, including facility-employed personnel health physics personnel, non-facility-employed research and service personnel, visitors, and security, fire, and other emergency personnel.

The applicant should describe the purpose, organization, and functions of any review and audit committees with responsibilities relating to radiation safety, including the charter, frequency of meetings, audits, scope of any reviews, and qualifications and requirements for committee members. The applicant should, describe how each committee's work relates to the radiation safety organization and how a comprehensive program is ensured. If this information is discussed in Chapter 12, it can be referenced here.

The program for conducting facility radiation safety audits of all functional elements of the radiation protection program to meet the requirements of 10 CFR 20.1101(c) should be described, identifying the scope of the audits, the bases for scheduling the audits, the qualifications of the auditors, the management level to which reports are sent, and the process for following up on audit findings. The relationship of this program to any other self-assessment/internal appraisal program should be discussed. The bases for TS related to facility radiation safety audits should be provided.

The system that examines the experiences of the radiation protection program and uses these experiences to improve the program and the facility design for radiation protection should be described. This system should also examine problems and incidents and develop "lessons learned," root causes, and effective corrective actions.

For activities not described in the SAR or not governed by procedures, a work control process such as the use of radiation work permits should be used. The applicant should discuss the control program used at the facility.

The applicant should describe the recordkeeping process for the radiation safety program, including record-retention periods, accessibility, review, and archiving. Review of radiation safety records for accuracy and validity should be discussed. The use of records for developing trend analyses, informing management, planning radiation-related actions, and reporting to regulatory and other duly authorized entities should be discussed.

11.1.3 ALARA Program

In this section, the applicant should describe the ALARA program for the facility required by 10 CFR 20.1101. The description should include the basis for the program and the management level and authority by which the facility ALARA policy is established. The applicant should discuss how this program is implemented to maintain radiological doses of all personnel at the facility and releases of effluents to the unrestricted area ALARA. The applicant should discuss the criteria used to determine how low the projected doses should be to permit task implementation (i.e., ALARA goals). The discussion should include methods to ensure that the radiation protection staff, with their considerations of the facility ALARA program, is specifically involved during review and approval of design, in construction of facilities, in the planning and implementing of reactor utilization (experiment design and planning) and operation, in maintenance activities, and in the management and disposition of radioactive wastes.

11.1.4 Radiation Monitoring and Surveying

The program employed to routinely monitor workplaces and other locations accessible to people for identification and control of sources of radiation exposure should be described in this section, including the measures designed to ensure that air, liquids, and solids are monitored in all applicable areas. The applicant should also discuss the bases of the methods and procedures used for detecting and assessing contaminated areas, materials, and components, and should describe the records that document the applicability, quality, and accuracy of monitoring methods, techniques, and procedures.

The applicant should provide summary descriptions of all radiation monitoring equipment employed throughout the facility, including locations and functions of each device and system. This summary should also describe sampling equipment for liquid and gaseous process and effluent streams. This discussion may be combined with (and appropriately cross-referenced to) the discussions in Chapter 7, “Instrumentation and Control Systems.” The applicant should discuss the interface between the radiation monitoring system and engineered safety features discussed in Chapter 6, “Engineered Safety Features,” if any exist. Types of equipment should include systems of the following types (as appropriate to the facility):

- Continuous air monitors (CAMs), including fixed and moving filter, and gaseous monitors
- Portable survey instruments (radiation fields and contamination)
- Remote area radiation monitors (RAMs)
- Samplers
- Effluent radiation monitors
- Environmental radiation monitors (details should appear in Section 11.1.7)
- Personal dosimeters
- Portal monitors,
- Radwaste storage and release monitors
- Criticality detection monitors

The calibration of radiation protection instrumentation, including the procedures and standards governing calibration, control of the calibration process, use of national standards, and verification should be described. In this section, the applicant should also describe the calibration equipment and discuss sensitivities to environmental and other conditions with respect to the calibration requirements. The program to ensure that routine periodic calibration is performed in a timely manner and the bases of calibration schedules should be described.

The applicant should describe how routine monitoring provided at the facility is planned to ensure that radiation exposures to the public and workers or material releases can be detected and should discuss how the approach used for routine monitoring provides reasonable assurance that all radiation at, and released from, the site will be appropriately monitored.

TS and their bases related to the radiation monitoring equipment and procedures, as discussed in Chapter 14, “Technical Specifications,” should be justified in this section.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure is controlled by controlling radioactive materials and effluent radioactive material releases. In this section of the SAR, the applicant should describe the design bases for the equipment and procedures utilized for controlling exposures to personnel and releases of radioactive materials from the facility, and should discuss how the facility structures, systems, and components are designed to provide assurance that there will be no uncontrolled effluent radioactive releases to the environment or to work areas. Some systems, such as containment, confinement, and ventilation, may have been discussed in other chapters of the SAR; reference to those discussions in this chapter of the SAR is appropriate. The applicant should also discuss how the bases of radiation dosimetry, radiobioassay interpretation, radiation shielding, ventilation, and remote handling and decontamination equipment are designed to ensure that doses to the workers are maintained ALARA and within the applicable regulatory limits.

How the design of required entry control devices (i.e., alarms, signals, or locked entry ways) alerts workers to, or prevents entry into, high radiation and very high radiation areas should be described. The regulations in 10 CFR Part 20, Subpart G, “Control of Exposure from External Sources in Restricted Areas,” contain requirements for control of access to high and very high radiation areas. It should be noted that 10 CFR 20.1601(c) allows a licensee to apply to the Commission for approval of alternative methods for controlling access to high radiation areas if the licensee finds that the stated methods of control in the regulations would interfere with utilization programs. The application should contain a description of the proposed method along with a discussion of how the entrance or access point to high radiation areas will be controlled.

Equipment and materials (e.g., anti-contamination clothing and respiratory protection equipment) to protect personnel employed in the facility should be discussed. The applicant should describe the facility conditions for which this protective equipment should be employed and should also discuss whether respirators should be used at the facility. The use of respiratory protection equipment requires implementing and maintaining a respiratory protection program in accordance with the requirements of 10 CFR 20, Subpart H. If a respiratory protection program will be maintained, that program should be described as it relates to the minimum program requirements of 10 CFR 20.1703.

The bases and values for the expected annual radiation exposure for all locations of the facility should be discussed, including the exposure estimates for applicant-employed personnel, non-applicant-employed research and service personnel, and visitors. This discussion should include the exposure limits and controls for such groups as embryos, fetuses, declared pregnant women, minors, and students. The plans and procedures for exposure control and dosimetry during the fill range of normal facility operations, potential accident conditions, rescue and recovery, and planned special personnel exposures (non-emergency) should also be discussed. The applicant should describe the dosimetry used for assessing external radiation exposures (e.g., whole body, extremities), including the frequency of dosimeter readings, administrative dose action levels, and the suitability of the dosimetry chosen with respect to the radiation sources anticipated and observed. The same factors for how internal exposures and doses are assessed, evaluated, and controlled should be described.

The applicant should describe the type of records retained to document the conditions under which individuals were exposed to radiation. The applicant should discuss the historical and current exposures to personnel and the associated trends.

11.1.6 Contamination Control

The applicant should discuss the plans and bases of procedures for identifying and controlling radioactive contamination, including methods established to assess the effectiveness of the contamination control program. The discussion should include information on the following topics, showing their relationship to regulatory requirements and ALARA concepts:

- Program for routine monitoring to detect and identify fixed and loose contamination
- Programs to control access to contaminated areas, avoid further spread of contamination, and remedy contaminated areas
- Personal monitoring and assessment of internal and external doses to personnel occupying or entering contaminated areas, and methods for appropriate surveying and “frisking” upon exit
- Use of anti-contamination techniques to protect workers, and control and disposition of possibly contaminated clothing and materials
- Procedures for monitoring and handling contaminated equipment and components outside of contaminated areas that have not been decontaminated
- Criteria for classifying contaminated material, equipment, and working areas, and managing, controlling, storing, and disposing of identified contamination
- Training programs for staff and visitors on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination

- Recordkeeping for contamination events, both for personnel and for locations, including records to be available for facility maintenance and for eventual decommissioning
- The bases of TS, if needed, applicable to contamination control: for example, limits on storage and handling of radioactive sources, especially unsealed ones; limitations on encapsulation of irradiated materials; and use of fume hoods and hot-waste drains

11.1.7 Environmental Monitoring

The applicant should describe the environmental monitoring program, including information relating to the following:

- Verification of compliance with commitments made in environmental reports, or other documents, if applicable; discussion of any standards used in the environmental monitoring program
- For established programs, evaluation of the effectiveness of the program
- Identification of potential facility impacts on the environment and the evaluation of the need for remedial action or mitigation measures
- Establishment of baselines for environmental quality, including data comparing preconstruction or preoperational with operational environmental monitoring results

The applicant should describe the written plans and the bases of procedures for implementing the environmental monitoring program, and should discuss the document control measures employed to ensure that the plans and procedures, including changes, are reviewed for adequacy and approved by authorized personnel, and are distributed to and used at the appropriate locations throughout the facility. The environmental surveillance program and its bases should be described. Air, water, and land environments should be specifically discussed. These discussions should include information on at least the following topics:

- Probable facility-related contaminants and pathways to people
- Selection of sampling materials and locations
- Sample analyses (analytical techniques) and sensitivities (detection limits)
- Records of results and trends

11.2 RADIOACTIVE WASTE MANAGEMENT

The applicant should address the five factors for analyzing the safety of auxiliary systems stated in the beginning of Chapter 9, applying those factors to stored waste and fuel, assuring chemical and nuclear stability for long term storage. Phenomena such as heat-induced recombination resulting in positive reactivity addition or the release and required management of tritium should be addressed. For reference, an Oak Ridge National Laboratory conference report documents a criticality concern that occurred in the Molten Salt Reactor Experiment after long-term storage of the solidified fuel salt when the fuel was heated to promote recombination of fluorine in the fuel salt drain tank. Over time, it was discovered that significant amounts of uranium and fluorine had migrated out of the drain tanks and into the off-gas system creating a criticality concern in the off-gas system.

In this section of the SAR, the applicant should propose and justify TS that constitute important design features, LCOs, and SRs discussed in Chapter 14 of this format and content guide.

Each facility that is licensed to operate or utilize a non-power reactor should establish a program and procedures that are designed to ensure that radioactive waste materials are identified, assessed, controlled, and disposed of in conformance with all applicable regulations and in a manner to protect the health and safety of the public and the environment. The magnitude and nature of the effort required should depend upon the size and complexity of both the reactor facility and its utilization programs. Therefore, the nature and details of the radioactive waste management program should also be commensurate with those factors. As noted previously, management of radioactive wastes could be an auxiliary function assigned to existing personnel, such as people engaged in radiation protection or operations. Earlier sections of this chapter have addressed the program and procedures for controlling and assessing radiation exposures and doses at the facility due to all radiation and radioactive sources. In this section, the applicant should address the program and procedures for further managing sources classified as radioactive waste.

11.2.1 Radioactive Waste Management Program

In this section, the applicant should discuss the philosophy and objectives of the program for managing radioactive waste. The applicant should describe the organizational structure within which it will administer the reactor-related radioactive waste management program, including the organization and staffing levels, authorities and responsibilities and position qualifications. The working relationships between such facility organizations as radiation protection and operations staff, and the standards, charters, procedures, or other documents that specify the authority, duties, and responsibilities of the personnel in the radioactive waste management organization should be discussed. The policy governing the program, the allocation of policymaking responsibilities, and the administrative plans and procedures that implement the facility policy should be described, including ALARA. The overall program and how the organization, policy, and program lead to effective management of radioactive waste should be evaluated and described.

The applicant should describe the purpose, organization, and functions of any committee’s assigned responsibility for overseeing radioactive waste management. The description should include each committee’s charter, responsibilities, frequency of meetings, audit and review responsibilities, scope of any audits or reviews, and qualifications and requirements for committee members. How each committee’s work relates to the waste management organization and how they work together should be discussed. If this information has already been described, reference that discussion.

The applicant should describe the waste management training program. This description should include the scope of facility waste management training, as well as specific training requirements for personnel associated with the operation and use of the facility.

The applicant should describe the document control measures that ensure that the plans and procedures involving radioactive waste, including changes, are reviewed for applicability, approved by authorized personnel, and distributed to and used at the locations where waste management activities are conducted.

The applicant should describe the scope of waste management reviews and audits. This description should include the authority of waste management review and audit teams, the objectives and purposes of reviews and audits, and the bases for scheduling these reviews and audits.

The applicant should describe the radioactive waste management recordkeeping process, including retention periods, accessibility, review, and archiving, and should discuss any special review of waste management records for accuracy and validity. Records of radioactive wastes stored for the life of the facility or buried on site should be discussed, as well as records for trend analysis.

The bases for any TS related to the radioactive waste management program should be described.

11.2.2 Radioactive Waste Controls

The applicant should discuss the definition of radioactive waste, the point in any process that a radioactive component or material becomes classified as waste, and the criteria for defining such waste. The applicant should describe the waste management program procedures which ensure that radioactive wastes are identified and characterized appropriately, as noted above, and the bases of the procedures which ensure that radioactive wastes are adequately segregated from nonradioactive wastes. The plans and procedures for managing all forms of radioactive wastes generated during operations, research, and utilization of the reactor should be described. Since radioactive wastes are radiation sources, they should be described, along with other such sources, in Section 11.1 of the SAR.

The applicant should be particularly vigilant of terminology that could confuse the process of waste removal with the production of special nuclear material. Some non-power reactor licenses do not allow the “separation of isotopes” or the “separation of byproduct materials” to enforce regulations dealing with the separation of plutonium or the enrichment of ^{233}U or ^{235}U to produce special nuclear material. In the strictest sense of the word that could be interpreted as not allowing the removal of wastes from MSR fuel salt cores. Fission gases do this inherently by simply rising from the liquid. Other undesired fission products are removed from the fuel salt by waste treatment processes with no intention of producing special nuclear material. The applicant should clearly describe waste treatment processes as such. The applicant should include in the safety analysis consideration that when using conventional waste treatment systems for an MSR there may be criticality concerns in the accident analysis, during normal operation, or both. If so, the applicable provisions of 10 CFR Part 70.24 should be met.

The applicant should describe the plans and bases for procedures for managing gaseous and other airborne radioactive wastes generated during operations, research, and utilization of the reactor, and radioactive waste off-gas collection systems designed to be utilized at the facility. The function and the location of each off-gas collection system should be described. At many non-power reactors, the system for removing gaseous radioactive waste is integral to the ventilation system for the facility and may have engineered safety functions. If these systems have been described in other chapters of the SAR, reference may be made here to those discussions. For all off-gas and ventilation systems, the applicant should describe the wastes produced by operation of the systems. Such items as filters and scrubbers, which collect and concentrate wastes, should be discussed to indicate the disposition of the radioactive material upon regeneration or replacement. If the radioactive materials enter other waste treatment systems, the applicant should indicate how such transfers are made and note any possible chemical or radiological effects of the transfer. The operation of any gas-cleaning equipment and its designed performance should be discussed in this section. The bases of any applicable TS that control these functions should be given. Also, the applicant should describe all secondary radioactive residues that are generated during process treatment, their chemical and physical composition, and the modes for handling, controlling, and storing them.

The applicant should describe how liquid radioactive wastes are generated and where they enter the waste control and treatment systems. Such items as laboratory wastes, liquid spills, and cleanup solutions, including detergent wastes, should be discussed. Information about the projected inventory levels, interim and long-term storage, and processing of those streams to achieve volume reduction or solidification should be included. This discussion should include information about fuel salt cleanup systems and resin regeneration solutions and wastes, if applicable.

The objectives of the processes designed to treat radioactive or mixed liquid wastes should be described. Any backup and special safety features designed to ensure that the radioactive waste is contained during treatment should be described. The designed equipment and systems should be described, and appropriate engineering drawings to show the location of the equipment, flow paths, piping, valves, instrumentation, and other physical features, should be included, along with information on all features, systems, or special handling techniques that prevent uncontrolled releases or personnel exposures.

The applicant should describe the plans and procedures for managing solid radioactive wastes generated during operations, research, and utilization of the reactor. This description should include how solid radioactive materials are generated and where they enter the waste control and treatment systems. For solid radioactive wastes retained or stored on site for the life of the facility, the applicant should discuss the control methods used. Integrity and corrosion characteristics and the monitoring of the containment should be discussed, as well as the plan for disposing of these radioactive wastes when the facility is permanently decommissioned.

The applicant should describe the systems and equipment selected for identifying, segregating, and safely managing the solid, liquid, and gaseous radioactive waste that is generated, and should include appropriate engineering drawings showing the location of the equipment and associated features used for volume reduction, containment, and/or packaging, storage, and disposal. The applicant should also discuss the bases of procedures associated with operating treatment equipment, including performance tests, process limits, and the means for monitoring and controlling to meet these limits. The bases of applicable TS that control these procedures and functions should be discussed. The methods and agents planned for all activities involving routine disposal or release to the environment of radioactive wastes generated in the facility should be described, as should methods used for packaging and shipping solid and liquid radioactive wastes to other facilities or other means for processing, storage, or other disposition.

The applicant should describe the program for minimizing radioactive waste for the facility with respect to the following topics: (1) the specific numerical goals for reducing the volume or radioactivity of each waste stream; (2) the periodic assessments of reactor operations and experimental or utilization activities to identify opportunities to reduce or eliminate the generation of wastes; (3) the continuing efforts to identify and, where cost effective, implement waste reduction technologies; and (4) any periodic independent reviews performed to evaluate the effectiveness of programs to minimize radioactive waste.

11.2.3 Release of Radioactive Waste

The applicant should identify all radioactive waste materials for which controlled release to the environment or transfer to other parties for disposal is planned. This discussion should include the projected concentrations, forms, radioactivity, chemical compositions, and annual quantities of radioactive waste released under normal operating conditions.

All points from which radioactive waste effluents are designed to be released from the facility to the environment should be identified, using a site map to locate the effluent release points and effluent monitoring equipment. Discussions and detailed analyses of potential radiological impact of radioactive waste effluents and the bases for continuous or intermittent monitoring should be provided in the earlier sections of Chapter 11. For liquid releases to the sanitary sewerage, the applicant shall ensure that the requirements of 10 CFR 20.2003 are met. The applicant should describe the systems and procedures designed to ensure that doses resulting from releases of radioactive effluents do not exceed applicable regulatory limits and ALARA goals.

11.3 RESPIRATORY PROTECTION PROGRAM

The applicant should describe how it plans to meet the requirements of 10 CFR Part 20 Subpart H, Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas by one of three methods.

- Pursuant to 10 CFR 20.1701 and in conjunction with ventilation equipment described in Chapter 9, “Auxiliary Equipment.”
- Pursuant to 10 CFR 20.1702 and in conjunction with the use of other controls as discussed in this section or as referenced in this section and discussed elsewhere in the application.
- Pursuant to 10 CFR 20.1703 and in conjunction with individual respiratory protection equipment used and maintained under a program described in this section of the application and in compliance with 10 CFR 20.1703.

11.4 REFERENCES

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. (withdrawn)

Code of Federal Regulations, Title 10, “Energy,” and Title 49, “Transportation,” US Government Printing Office, Washington, D.C., revised periodically.
F. J. Peretz, et al, ORNL/CP-98146, “Removal of Uranium and Salt from the Molten Salt Reactor Experiment,” Oak Ridge National Laboratory, 1998.

US Nuclear Regulatory Commission, NUREG-0851, “Nomograms for Evaluation of Doses from Finite Noble Gas Clouds,” January 1983.

US Nuclear Regulatory Commission, NUREG/CR-2260, “Technical Basis for RG 1.145,” 1981.

US Nuclear Regulatory Commission, Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents.”

US Nuclear Regulatory Commission, Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.”

US Nuclear Regulatory Commission, Regulatory Guide 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.”

US Nuclear Regulatory Commission, Regulatory Guide 8.9, Revision 1, “Interpretation of Bioassay Measurements.”

US Nuclear Regulatory Commission, Regulatory Guide 8.10, Revision 2, “Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable.”

US Nuclear Regulatory Commission, Regulatory Guide 8.13, Revision 3, “Instructions Concerning Prenatal Radiation Exposure.”

US Nuclear Regulatory Commission, Regulatory Guide 8.29, Revision 1, “Instruction Concerning Risks from Occupational Radiation Exposure.”

US Nuclear Regulatory Commission, Regulatory Guide 8.37, “ALARA Levels for Effluents from Materials Facilities.”

12 CONDUCT OF OPERATIONS

Chapter 12 of this guide is applicable to providing a description of the conduct of operations for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should describe and discuss the conduct of operations at the MSR. The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator requalification plan, the startup plan, and environmental reports. The administrative aspects of facility operations are the facility organization, review and audit activities, organizational aspects of radiation safety, facility procedures, required actions in case of license or technical specification violations, reporting requirements, and recordkeeping. This chapter of the SAR forms the basis of Section 6 of the technical specifications.

12.1 ORGANIZATION

In this section of the SAR, the applicant should discuss the organizational structure, responsibilities, and staffing, including selection and training of personnel. This discussion should show that the management and staff of the facility are knowledgeable about the technical requirements to operate a safe facility, are responsible for complying with regulations and license conditions, and will implement a meaningful radiation protection program that will protect the health and safety of the public, the facility users, and the staff. Additional information on these topics is given in Chapter 14, "Technical Specifications," of this format and content guide.

Not all owners and operators of non-power reactors have the same management organization or office titles. Regardless of the details of the management organization or the complexity of the facility, the administrative functions that should be in place at a non-power reactor facility are consistent among the various non-power reactor designs.

12.1.1 Structure

The applicant should discuss the organizational structure for the facility and should submit an organization chart. Lines of responsibility and lines of communication between groups should be shown in a multilevel chart. The individual or group with legal responsibility for holding the reactor license (e.g., utility, research group, university provost or dean) should be shown at the top of the organization, and the individuals who operate the reactor should be shown at the bottom. Intermediate levels should show the individuals who are in charge of the reactor facility and reactor operations (e.g., the facility director and the operations manager). The description of the organizational structure should include the radiation safety function and indicate how the staff implementing that function interacts with the staff responsible for reactor operations and the top administrative officials. The multilevel chart should show the relationship of the review and audit function to the organizational structure. The persons implementing the review and audit function should communicate with the management of the reactor facility but should report to an organizational level above this management to ensure independence of the review and audit function.

12.1.2 Responsibility

The applicant should discuss the individuals or groups that appear in the organizational structure and their respective responsibility for the safe operation of the reactor and the reactor facility, the protection of the health and safety of the public and the workers at the facility, and the protection of the environment.

12.1.3 Staffing

The applicant should discuss staffing issues and the minimum staffing of the facility when the reactor is not secured. The applicant should also discuss the availability of senior reactor operators during routine operations and should list the events that require the presence of a senior reactor operator at the facility. Staffing shall meet, at a minimum, the requirements of 10 CFR 50.54(i), (j), (k), (l), and (m)(1). The applicant should show that these requirements are met.

12.1.4 Selection and Training of Personnel

The applicant should discuss the selection and training of personnel. If minimum requirements exist for the facility staff, they should be discussed in this section. For example, the facility director may have to meet certain educational standards. The applicant should discuss the training programs at the facility, including the initial training and the requalification training of reactor operators. The applicant and licensed operators shall comply with 10 CFR Part 55. The applicant shall meet the requirements of 10 CFR Part 19. The applicant should discuss the training necessary to ensure that all workers meet the requirements of 10 CFR Part 19. Specialized training may be needed for researchers who work near neutron beams or whose experiments may cause changes in reactivity in the reactor. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4 contains additional guidance on the selection and training of personnel for research reactors.

12.1.5 Radiation Safety

The applicant should describe the organizational aspects of the radiation safety program. The program itself should be discussed in detail in Chapter 11, “Radiation Protection Program and Waste Management,” of the SAR. The radiation protection staff can be part of the reactor facility staff or may be provided as a service by a corporation-wide group or by an independent group. The radiation protection staff can report to either the managers of the facility or to the management chain above the facility. If this staff reports to the reactor manager, the applicant should discuss the method of raising concerns to the level of the review and audit committee or to the management chain above the facility. The applicant should specifically discuss the authority of the radiation safety staff to interdict or terminate safety-related activities.

Additional guidance for radiation safety programs at non-power reactors is given in ANSI/ANS 15.11.

12.2 REVIEW AND AUDIT ACTIVITIES

In this section of the SAR, the applicant should discuss the review and audit activities at the reactor facility. This independent oversight is very important to the safe operation of the facility and the protection of the health and safety of the public. NRC expects review and audit programs to be viable and to be strongly supported by the licensee. Having independent experts examine certain activities improves the quality of the program. Independent audit allows the licensee to find and correct problems before NRC inspectors discover them. Under its enforcement program, NRC looks favorably on licensees that aggressively seek out and correct problems in their programs.

The committee established for the review function may be assigned approval authority by the facility manager, or the facility manager may retain the authority. The applicant should explicitly state who holds the approval authority and should specify the committee’s authority and how it communicates and interacts with facility management and university or corporate management.

12.2.1 Composition and Qualifications

The applicant should discuss the composition and qualifications of the review and audit committee. Both functions can be performed by one committee or two separate committees. The applicant should discuss the minimum number of committee members required. Committee members can be members of the reactor facility staff as long as they constitute a minority of the quorum for voting. The applicant should also discuss the qualifications of committee members. Because it is sometimes difficult, especially for small facilities, to have all types of engineering and health physics expertise on staff, the review and audit committee provides access to expertise. The committee members should have a variety of backgrounds to provide both additional expertise that the reactor staff may not have and to allow a second opinion in areas in which the reactor staff has expertise. The applicant should discuss the use of committee members from outside the MSR facility organization. It is desirable to have some members on the committee from outside the facility management to increase the independence of the committee.

12.2.2 Charter and Rules

The applicant should discuss the committee's charter and rules, including the number of times the committee meets, the way the committee conducts business, the requirements for a quorum when voting, and the way the committee distributes its reports and reviews to the applicant. It is important that the definition of a quorum ensures that the reactor staff does not form a majority of members present for any vote.

12.2.3 Review Function

The applicant should list and discuss the items that must be reviewed by the committee. Additional information on the review function can be found in Chapter 14 of this format and content guide.

12.2.4 Audit Function

The applicant should list and discuss the items that must be audited by the committee. In addition to audits by the facility committee, the licensee may consider entering into an auditing agreement with other non-power reactor facilities to bring in staff members from those facilities to perform an audit. This approach has been very productive at the facilities that have used it. The applicant should consider all aspects of facility operations for audit, including the radiation protection program (see Section 11.1.2 of this format and content guide) and the laboratory program at the facility if the program is conducted under the reactor license (see Section 9.5 of this format and content guide). The emergency plan, the physical security plan, and the operator requalification plan should be specified for audit, although the requirement for auditing these plans may be contained in the plan itself.

12.3 PROCEDURES

In this section of the SAR, the applicant should discuss the use of procedures at the facility. NRC does not usually review procedures as part of licensing reviews. The applicant should discuss the basic topics that the procedures do or will cover. If laboratory work is conducted under the reactor license, the applicant should reference procedures applicable to this work. The applicant should discuss the methodology used for developing procedures, including the approval process. The applicant should also discuss the process required to make changes to procedures including substantive and minor permanent changes, as defined in ANSI/ANS 15.1, and temporary deviations to deal with special or unusual circumstances during operation. The applicant should note that 10 CFR 50.59 may apply to changes to procedures. See Chapter 14 of this format and content guide for additional information on the minimum acceptable set of procedures.

12.4 REQUIRED ACTIONS

In this section of the SAR, the applicant should discuss actions to be taken in the event of a violation of the facility safety limits or the occurrence of a reportable event. In this case of a safety limit violation, the reactor shall be shut down, the proper facility management notified, the event investigated, and NRC notified. The method and timeliness of all notifications should be addressed in this section. Additional information on required actions for a safety limit violation is given in Chapter 14 of this format and content guide.

The applicant should discuss what occurrences are considered reportable events and the actions to be taken if a reportable event occurs. These actions usually consist of returning the reactor to its normal condition or shutting the reactor down and reporting the event to facility management and NRC. Additional information on reportable events is given in Chapter 14 of this format and content guide.

12.5 REPORTS

In this section of the SAR, the applicant should discuss what information should be reported to NRC, the format of reports, the timing of reports, and the distribution of reports to NRC. This discussion and related technical specifications on reports take the place of the reporting requirements in 10 CFR 50.72 and 73, which are not applicable to non-power reactors. The applicant should discuss the annual operating report and should list the required contents. The annual report may also be used to meet the reporting requirements of 10 CFR 50.59 by discussing changes made to the facility and procedures and the tests and experiments conducted under the authority of 10 CFR 50.59. Additional information on the contents of annual operating reports is given in Chapter 14 of this format and content guide.

The applicant should discuss special reports, which are used to inform NRC of the violation of safety limits or the occurrence of reportable events, as previously discussed. The special reports are also used to inform NRC of changes in facility management and of changes in the transient and accident analyses in the SAR. Additional information on special reports is given in Chapter 14 of this format and content guide.

12.6 RECORDS

In this section of the SAR, the applicant should discuss facility records and describe the recordkeeping system for the facility, the types of records that need to be retained, and the period of retention. Additional information on records is given in Chapter 14 of this format and content guide.

12.7 EMERGENCY PLANNING

In this section of the SAR, the applicant should give a brief overview of the plan. The reader should be referred to the emergency plan for additional details.

The applicant should follow the guidance of ANSI/ANS 15.16, which is endorsed and amplified by Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors". The applicant should also review NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors".

The applicant should ensure that the terminology for releases discussed in the emergency plan match that used in the revision of 10 CFR Part 20. The use of dose values in millisieverts (mSv) (millirem (mrem)) would be a more appropriate protective action guideline because dose is the ultimate criterion specified.

The applicant should ensure that the action levels discussed in the emergency plan for each emergency class follow the guidance. If it is impossible for an event at a particular facility to reach a given action level, that emergency class is not possible, and the plan should state that fact.

12.8 SECURITY PLANNING

In this section of the SAR, the applicant should briefly discuss security planning for the entire facility. The information in the SAR must not contain Safeguards Information -Modified Handling (10 CFR 73.21 and 10 CFR 73.23) or Safeguards Information (SGI) (10 CFR 73.21 and 10 CFR 73.22). The SGI version of the security plan is protected from disclosure by the regulations referenced above. The applicant should refer to the guidance in Regulatory Guide 5.59 when developing the facility security plan. New facilities may be subject to additional security measures in the form of license conditions.

12.9 QUALITY ASSURANCE

Applicants for construction permits to build non-power reactors must describe a quality assurance program for the design and construction of the SSC of the facility in accordance with 10 CFR 50.34(a)(7). Section 50.34(b)(6)(ii) requires a description in the SAR of managerial and administrative controls to be used to ensure safe operation. The applicant should consider the guidance in Regulatory Guide 2.5 and ANSI/ANS 15.8 in developing quality assurance programs for MSR non-power facilities. These guidance documents provide an acceptable method of complying with the program requirements of 10 CFR 50.34.

12.10 REACTOR OPERATOR TRAINING AND REQUALIFICATION

Each reactor operator or senior reactor operator is required to successfully complete a qualification program developed by the licensee that has been approved by the Commission by passing a comprehensive written examination and an operating test as required by 10 CFR Part 55.

Subsequently, each active reactor operator or senior reactor operator is required to successfully complete a requalification program developed by the licensee that has been approved by the Commission by passing a comprehensive written examination and an annual operating test. The operator requalification plan should describe the operator requalification program to be administered to non-power MSR personnel possessing operator and senior operator licenses in order for them to maintain active status. The plan should also discuss the requirements for reactor operators and senior reactor operators to maintain active status and the steps that should be taken to return an inactive operator to active status. The operator requalification plan is normally a separate document. It should include the organizational structure for implementing the program.

Section 50.54(i-1) of 10 CFR requires that within 3 months after an operating license is issued, the licensee have in effect an operator requalification program, which at a minimum meets the requirements of 10 CFR 55.59(c). In accordance with 10 CFR 55.59(a)(1) and 55.59(c)(1), the requalification period should not exceed 24 months.

The operator requalification plan should include information on the following:

- Requalification schedule
- Lectures, reviews, and examinations
- On-the-job training
- Emergency procedures
- Inactive operators
- Evaluation and retraining of operators

- Requalification documentation and records
- Requalification document review and audit

Preplanned lectures should be given on a regular and continuing basis as required by 10 CFR 55.59(c)(2). Some examples of lecture topics are the following:

- Nuclear theory and principles of operation
- Facility design and general and specific operating characteristics
- Instrumentation and control systems
- Reactor protection system
- Engineered safety features
- Normal, abnormal, and emergency operating procedures
- Radiation control and safety
- Technical specifications and applicable portions of 10 CFR
- Other facility specific training

Self-study methods may be considered and adequate and appropriate training method for the lecture program topics when learning objectives are properly measured by examination or documentation of expertise. Self-study programs must be justified by the applicant.

On-the-job training should occur during the requalification period so that each operator (1) is involved in facility manipulations [see 10 CFR 55.59(c)(3)(i)], (2) understands the operation of apparatus and mechanisms associated with control manipulations and knows operating procedures [see 10 CFR 55.59(c)(3)(ii)], (3) is cognizant of changes in facility design, procedures, and license [see 10 CFR 55.59(c)(3)(iii)], and (4) reviews the contents of all abnormal and emergency procedures on a regular basis [see 10 CFR 55.59(c)(3)(iv)]. Reactor operators must perform, and senior reactor operators must either perform or supervise the appropriate items from 10 CFR 55.59(c)(3)(i), such as reactor startup, shutdown, and significant power change, on an annual basis. NRC has approved requalification plans for on-the-job training that are similar to the following:

- Over the 2-year requalification period, each licensed individual should perform at least ten reactivity control manipulations in any combination of reactor startups, shutdowns, or significant reactivity changes, and each licensed operator should perform at least one reactor startup and one daily checkout quarterly at intervals not to exceed 4 months.
- Any changes in procedures, technical specifications, regulations, as well as any change with safety significance to the facility should be reviewed by every licensed operator on an ongoing basis as part of a required reading list.
- Each operator should participate in at least half of the emergency drills per year specified in the technical specifications. Operators who did not meet the minimum drill requirement should receive special training on proper response to emergencies and a documented review of the last drill missed as well as a walkthrough of the facility related to proper emergency responses. The drill and applicable emergency procedures should be reviewed with all licensed operators within 30 days after completion of the drill.

The requalification program should include provisions for evaluating operators [see 10 CFR 55.59(c)(4)] that include written examinations, observation and evaluation of operator performance, and simulation of emergency or abnormal conditions. The applicant should discuss provisions for accelerated requalification if performance evaluations indicate the need [see 10 CFR 55.59(c)(4)(v)].

Examinations should be administered as necessary to assess the progress of the lectures or self-study. A comprehensive requalification written examination is required for all operators at least biennially [see 10 CFR 55.41, 55.43, 55.59(a)(2)(i), and 55.59(c)(4)]. The contents of the comprehensive examination should be listed and should cover all lecture topics and requirements of the regulations.

For example, NRC has accepted requalification plans similar to the following:

- The acceptance criterion for all graded examinations is 80 percent and all operators are required to complete each examination satisfactorily.
- A score on the written or other examination equal to or greater than 80 percent may require no additional training. Nevertheless, the results of all examinations including missed questions should be reviewed with the operator to ensure proper understanding.
- A score on the written or other examination in the range of 65 to 79 percent requires additional training on those areas or topics where weaknesses or deficiencies are indicated. This retraining and retesting are completed within 60 days from the date the examination was administered and before the candidate is considered requalified. In this case the candidate need not be removed from licensed duties subject to the evaluation of the reactor manager or a duty authorized representative.
- A score on the written or other examination of less than 65 percent requires that an evaluation be performed by the facility director or designated representative within 1 month. The evaluation is to determine if the deficiencies require that the individual be removed from licensed duties pending completion of any accelerated retraining. In any case the licensed operator is removed from licensed duties if within 4 months he or she does not achieve a passing grade after reexamination.
- Regardless of the score, if the individual's test indicates a deficiency in a critical area that affects safety, training is promptly administered to correct the deficiency, or the operator is removed from licensed duties in the affected area until the deficiency is corrected.

Each reactor operator and senior reactor operator is required to take an annual operations test to demonstrate operational proficiency and understanding of systems responses [see 10 CFR 55.45(a)(2-13), 55.59(a)(2)(ii), and 55.59(c)(4)]. NRC has approved requalification plans similar to the following:

- Each licensed reactor operator and senior reactor operator demonstrates satisfactory understanding of the operation of the facility systems, operating procedures, and license as well as changes in facility procedures and the license during an annual walkthrough examination administered by a designated senior reactor operator.
- The annual operations test and annual walkthrough examination are key factors in evaluating the continued competence of the licensed operator for demonstrating both (1) operational proficiency and understanding of system responses and (2) overall satisfactory understanding of the operations of the facility, operating procedures, and facility license changes. The results of these two examinations are used as primary input for evaluating operator performance for requalification purposes.
- An in-depth evaluation of the operating performance of each licensed operator is performed and documented biennially, at a minimum, by a summary and judgement statements. The biennial evaluation includes results from the written examinations, the annual operations test, the annual walkthrough examination, and other on-the-job evaluation of operational proficiency as well as any other available indications of the operator's capability to discharge duties in a safe and competent manner, including participation in practical and special training, instructional activities, and other work activities.

The requalification plan should discuss records associated with the program [see 10 CFR 55.59(c)(5)]. NRC has accepted requalification plans similar to the following:

- Operator requalification records are kept to ensure that all requirements of the facility plan are met. Each operator has an individual folder or notebook containing signature blocks for lectures attended, prepared or assigned self-study sessions, reactivity manipulations performed, weekly and daily checkouts performed, and quarterly emergency drills participated in. The notebook also contains copies of written examinations administered, the answers given by the operator, results of any evaluations, and documentation of any additional training administered in areas in which an operator has exhibited deficiencies. The performance of, or participation in, special training such as for fuel handling or use of emergency equipment is also logged in the notebook.
- A master requalification training manual is used to organize training requirements; the manual contains a schedule of all required lectures, reviews, emergency drills, and other exercises. The date the item is performed is indicated in this manual. A section of this manual is designated to contain completed training items, attendance sheets, master copies of tests given, and the lecture outlines if available.
- Required documents and records pertaining to the requalification program are maintained as part of the facility records for at least 6 years. The records including the master training file are retained for each reactor operator or senior reactor operator until the respective operator's license is renewed or surrendered.

To maintain active status, 10 CFR 55.53(e) requires that each licensed reactor operator or senior reactor operator actively perform the functions of a reactor operator or senior reactor operator for minimum of 4 hours each calendar quarter. For senior reactor operators, direct supervision of these operations may be considered equivalent to actual performance. If this requirement is not met, the license becomes inactive; before reactivation of the license, the licensee should verify that the qualifications and status of the operator are current and the operator should perform 6 hours of licensed activities in the position that is being recertified under the direction of a licensed operator or senior reactor operator as appropriate. For example, NRC has approved requalification plans similar to the following:

- An operator who has not been actively performing licensed functions for a period in excess of 4 months is required to demonstrate to the reactor manager or duly authorized representative that his or her knowledge and understanding of the operation and administration of the facility are satisfactory before returning to licensed duties. An interview and evaluation or a written, oral, or operational examination, or a suitable combination, can be used.
- Any deficiencies uncovered are corrected before the individual resumes performance of licensed functions.
- The operator performs 6 hours of operation under the direction of a reactor operator or senior reactor operator as appropriate.

The applicant should discuss audits of the requalification plan in the plan or as part of Section 6 of the technical specifications. The plan should be audited by the facility review and audit committee at least every other calendar year, with the interval between audits not to exceed 30 months. The audit should consist of a sampling of the plan records for completeness and compliance with the requirements of the plan.

12.11 STARTUP PLAN

In this section of the SAR, the applicant should describe the startup plan for the MSR facility. Startup operations involving fuel should be described as processes with homogenous fuel in liquid or solid form, as appropriate. This section is applicable to new facilities or license amendments authorizing modifications that require verification of operability before normal operations are resumed. Startup plans ensure that the operating characteristics are well understood and validate the predicted behavior of the reactor. Measurements of selected parameters of the reactor should be compared to calculated values to verify analytical methods and ensure that meaningful acceptance criteria for the reactor have been established from the calculational methods. The acceptance criteria should ensure that the reactor is functioning within the bounds for which it was designed and analyzed and that the license and the technical specifications are satisfied.

Operations with the fuel or special nuclear material (SNM) that are conducted outside of the reactor may be subject to the requirements of 10 CFR Part 70. Applicable portions of 10 CFR Part 70 may be incorporated into the Part 50 license as license conditions. Examples of such operations include the following:

- Receipt, unpacking, and internal transfer and storage of new fuel or SNM
- Preparation of fuel for use in the reactor
- Processing of fuel for reuse in the reactor or for disposal
- Packing of spent fuel or SNM for transport

The reactor startup plan should include the following:

- A well-planned systematic approach to melting solid fuel or loading liquid fuel into the vessel
- A well-planned systematic set of subcritical multiplication measurements or an inverse multiplication approach to critical measurement during fuel loading, and confirmation that subcritical multiplication or critical fuel loading is within preestablished acceptable limits
- An experimental measurement plan to determine the important operational reactor physics parameters (such as control element worth, excess reactivity, reactor thermal power, coefficients of reactivity, and power peaking factors) and thermal-hydraulic parameters (such as fuel salt temperature, reactor flow rates, and pressure drops, if appropriate); comparisons with predictions and acceptance criteria; and investigation and discussions of any discrepancies that may have arisen
- Measurements of magnitudes of area radiation fields and radioactive effluents, and comparisons with predictions in the SAR and preestablished acceptance criteria
- Measurements of performance of other systems (e.g., off-gas, ventilation, engineered safety features, or emergency power), and comparisons with predictions in the SAR and preestablished acceptance criteria

The applicant should submit a startup report to NRC in accordance with the timeframe required in the technical specifications. A sample outline for a startup report follows (the actual contents of the report will vary by reactor type and scope of the report):

- Measurements and comparisons with the predication of subcritical multiplication for initial fuel loading
- Confirm fuel salt chemistry is within the expected and analyzed limits
- Critical mass and final criticality conditions for the initial active reactor core and operational active reactor core, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Control and regulating element calibration, including measurements of differential and integral element worth for the initial and operational active reactor core, and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Excess (operational) reactivity, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Measured shutdown margin, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses

- Reactor power calibration, including methods and measurements that ensure operation within the license limit; discussion of nuclear instrumentation setpoints, detector positions, and detector output; and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Neutron flux distributions, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Radiation measurements of reactor coolant salt radioactivity or release during the startup test program to detect potential degradation of the fuel system boundary or contamination from other sources; results of radiation measurements to show effectiveness of facility shielding; measurements of airborne effluents released from the facility; and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Reactivity worths of experimental facilities, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Results of determination of temperature coefficients and void coefficients for the active reactor core, and comparisons with acceptance criteria preestablished in the SAR calculations and analyses
- Thermal-hydraulic characteristics such as fuel salt temperature and flow rate, primary cooling system salt temperature and flow rate, and respective pressure drops
- Measurements of performance of engineered safety features and other tested systems, including comparisons with acceptance criteria preestablished in the SAR calculations and analyses

This list is not complete. The components and systems to be tested and the test results reported to NRC will vary with MSR design and the subject of the licensing action.

12.12 MATERIAL CONTROL AND ACCOUNTING PROGRAM

In this section, the applicant should present information about the material control and accounting (MC&A) program. The description should be sufficient to ensure that the program can fulfill its functions. The applicant should consult NUREG-1065 on the standard format and content for the fundamental nuclear material control plan required for low-enriched uranium facilities." The information in this section should include the following:

- MC&A organization;
- Measurements;
- Measurement control program;
- Statistics;
- Physical inventories;
- Item control;
- Shipper-receiver comparisons;
- Assessment and review of the MC&A program;
- Resolving indications of missing uranium or other SNM of significance;
- Informational aid for assisting in the investigation and recovery of missing uranium; and
- Record keeping.

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, 2007 (R2013).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, LaGrange Park, Illinois, 2016.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, “Quality Assurance Program Requirements for Research Reactors,” ANS, LaGrange Park, Illinois, 1995 (R2013).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, “Radiological Protection at Research Reactor Facilities,” ANS, LaGrange Park, Illinois, 2016.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, “Emergency Planning for Research Reactors,” ANS, LaGrange Park, Illinois, 2015.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.5, “Quality Assurance Program Requirements for Research and Test Reactors,” Revision 1, 2010.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.6, “Emergency Planning for Research and Test Reactors,” Revision 2, 2017.

U.S. Nuclear Regulatory Commission, Regulatory Guide 5.59, “Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance,” Revision 1, 1983.

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-1065, “Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Facilities,” Revision 2, 1995.

13 ACCIDENT ANALYSES

Chapter 13 of this guide is applicable to providing a description of the accident analyses for a non-power MSR.

In the other chapters of the SAR, the applicant should discuss and analyze the safety considerations and functional requirements at an MSR for the design bases that ensure safe reactor operation and shutdown and acceptable protection for the public, the operations and user staff, and the environment. In those chapters, the applicant should not only discuss potential equipment malfunctions, deviations of process variables from normal values, and potential effects of external phenomena on the facility, but should also describe how equipment will work when needed in accident situations. In this chapter of the SAR, the applicant should submit information and analyses that show that the health and safety of the public and workers are protected and that the applicant has considered potential radiological consequences in the event of malfunctions and the capability of the facility to accommodate such disturbances. The application should also show that the public and workers are protected from chemical hazards that are under NRC's regulatory jurisdiction (see the Memorandum of Understanding between the U.S. Nuclear Regulatory Commission and the Occupational Safety and Health Administration (ADAMS Accession No. ML11354A432)). The major purpose of this chapter is for the applicant to demonstrate that the facility design features, safety limits, limiting safety system settings, and limiting conditions for operation have been selected to ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

The class of MSR s discussed in this chapter use liquid fuel rather than solid heterogeneous fuel, and the resulting fission products—both liquid and gaseous—must be contained within the facility barriers rather than within heterogeneous fuel cladding. The homogeneous reactor fuel (fuel salt) is contained within the fuel system boundary. The fuel system boundary consists of all structures, systems, components, and coatings that prevent the release of fuel, fission gases, and other fission products that remain in the fuel salt. For an MSR, the fuel system boundary includes, but is not limited to, the vessel, the waste handling tanks, drain tanks or fuel holding tanks, the gas management system, and associated pumps, heat exchangers, valves, and piping. As part of the accident analyses, the MSR applicant should identify all the sources, locations, quantities, and potential release paths for radioactive material.

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 are compared with 10 CFR 20.1001 through 20.2404 and appendices. Occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analyses with results greater than the 10 CFR Part 20 dose limits discussed above.

If the facility conforms to the definition of a test reactor, the results should be compared with 10 CFR Part 100. As discussed in the footnotes to 10 CFR 100.11, the doses given in 10 CFR Part 100 are reference values and are not intended to imply that the dose number constitute acceptable limits for emergency doses to the public under accident conditions. Rather, they are values that can be used for evaluating reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of exposure of the pubic to radiation.

The accidents analyzed should range from such anticipated events as a loss of normal electrical power to a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible. This limiting accident is named the maximum hypothetical accident (MHA) for non-power reactors; the details are reactor specific. Because the MHA may be a non-mechanistic failure assumed to establish an outer limit consequence, the scenario need not be entirely credible. The initiating event and the scenario details need not require analysis, but the potential consequences should be analyzed and evaluated.

The information on credible postulated accidents should achieve the following objectives:

- Ensure that enough events have been considered to include any accident with significant radiological consequences. Rejection of a potential event should be justified in the discussions.
- Categorize the initiating events and scenarios by type and likelihood of occurrence so that only the limiting cases in each group must be quantitatively analyzed.
- Develop and apply consistent, specific acceptance criteria for the consequences of each postulated event.

The selection of accident scenarios to be analyzed should be based on the consideration of phenomena unique to an MSR that could limit safe operation. Limiting phenomena refer to those physical phenomena that could occur during the course of a transient or accident that significantly affects the subsequent likelihood of failure of the fuel system boundary. Identification and understanding of these limiting phenomena are useful in classifying the consequences of potential transients and accidents, as well as in determining appropriate operating limits. Applicants should identify the limiting phenomena for their technology design. For MSRs, the limiting phenomena include fuel salt precipitation, fission-product precipitation, fuel salt chemistry/physical properties, delayed neutron production, active reactor core voiding, criticality accidents, and tritium production. These phenomena are defined as follows:

- Precipitation of fission products / Precipitation of fuel (uranium)
- Precipitation refers to the formation of solids in the MSR fuel salt. Precipitation of the fuel would lead to uranium collecting in low velocity areas where a hot spot could develop. The accumulated fuel could also break loose resulting in a sudden insertion of fissile material in the active reactor core. This would lead to an abrupt temperature increase in the active reactor core. Precipitation of fission products will result in plate out of these precipitates on undesired surfaces adding to corrosion and limiting heat transfer. Subsequent chemical and thermal-mechanical effects could then challenge fuel system boundary integrity.
- Fuel salt chemistry including redox control and fission product polishing
- Corrosion control is essential to continued operation of an MSR. Though corrosion mechanisms are generally slow-acting, eventual failure of the fuel system boundary could occur if fuel salt chemistry is not maintained within an established set of technical specification limits.
- Reactor flow and the location of delayed neutron production
- Reactor flow affects the location of delayed neutron production. Sudden changes in flow will affect the number of delayed neutrons produced in the active reactor core and subsequently impact active reactor core reactivity. Therefore, reactor power can be controlled by fuel salt flow.
- Solidification in an MSR refers to a fuel salt phase change from a liquid to a solid. A phase change could occur as a result of overcooling in the heat dissipation system that causes overcooling to occur at the fuel salt/primary cooling system salt heat exchanger or as a result of shutdown in areas where pipe heat tracing fails or does not exist. Solidification during operation could also affect active reactor core flow and decay heat removal.
- Active reactor core voiding due to gaseous fission product formation
- Many fission products or their daughter products are gaseous, and they are entrained in the fuel salt. This leads to bubbling or voids flowing through the active reactor core with the fuel salt, which impacts reactivity. Since the amount of voiding in the active reactor core is not precisely the same from one time period to the next, there will be a small, constant fluctuation in active reactor core power (heat). However, sudden changes in the amount of voiding could have a more noticeable step change in power (heat).

- Criticality accidents
- Onsite preparation of the fuel salt and subsequent initial loading of the fuel salt into the vessel are operations that have the potential to introduce conditions that may lead to a criticality accident. Likewise, precipitation of actinides during operation of the reactor may also introduce conditions that may lead to a criticality accident.
- Tritium production

Depending on the salt selection for a given MSR technology, tritium production could be a shielding issue. As an MHA consideration, excessive tritium production could lead to excessive hydrogen accumulating in the gas management system leading to an explosion.

Each postulated event should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor:

- MHA
- Reduction in fuel salt inventory from a barrier failure (includes rupture of the vessel, waste-handling tanks, the gas management system, the polishing system, pumps, valves, heat exchangers, and piping)
- Increase in fuel salt inventory
- Reduction in cooling
- Reactivity and power distribution anomalies
- Mishandling or malfunction of fuel (includes fuel salt composition changes)
- Experiment malfunction (if experiment capabilities are included)
- External events (includes natural hazards and manmade events)
- Mishandling or malfunction of equipment (i.e., stuck-open relief valve, dropping a heavy object, fires)
- Loss of normal electrical power

The accident events in each group should be evaluated systematically to identify the limiting event selected for detailed quantitative analysis. Limiting events in each category should have potential consequences that exceed all others in that group. As noted above, the MHA selected should bound all credible potential accidents at the facility yet should be an event that is not likely to occur during the life of the facility. The applicant should provide information on the following fuel salt parameters for each credible accident identified:

- Temperature
- Pressure
- Flow
- Density
- Reactivity
- Radiological inventory
- Chemical/physical property changes

13.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

In this section of the SAR, the applicant should describe potential accident-initiating events and scenarios for non-power MSRs. For documents on general accident scenarios and analysis, radiological consequences, and fuel types, see Section 13.4, “References.” The following sections discuss selecting and categorizing postulated accidents:

13.1.1 Maximum Hypothetical Accident

In general, the escape of fission products past all established MSR barriers and their release to the unrestricted environment would be the most hazardous radiological accident conceivable at a non-power MSR. However, non-power reactors are designed and operated so that a fission product release is not credible for most. Therefore, the MHA could be based on a breach of the fuel system boundary that results in unrestricted dispersal of radioactive material which bounds all credible accidents and can be used to illustrate the analysis of events and consequences during the accidental release of radioactive material. The MHA need not include an active reactor core scenario because other systems may provide the bounding accident conditions (e.g., the gas management system or the fuel handling system).

Possible MHAs for an MSR could be one or a combination of the following events:

- Complete loss of reactor fuel salt inventory (*e.g.*, vessel rupture or drain tank rupture)
- Complete loss of the gas management system inventory (*e.g.*, system rupture)
- Release of used fuel constituents to the environment (*e.g.*, waste holding tank rupture or fuel polishing system failure)
- Energetic dispersal of the contents from the fuel system boundary with bypass of any scrubbing or delay capacity
- External hazards including flooding, high winds, seismic events, volcanism, missile impact, aircraft impact, and fires
- Excessive tritium production
- A fueled experiment fails catastrophically

13.1.2 Reduction in Fuel Salt Inventory from a Barrier Failure

The fuel system boundary consists of all structures and coatings that prevent the exposure of fuel, fission gas, or other fission products to the next credited release barrier. For an MSR, this includes the vessel, drain tanks, waste-handling tanks, drain tanks or fuel holding tanks, the gas management system, pumps, valves, heat exchangers, and piping. Each SSC provides an opportunity to release radioactive material, resulting in a reduction in fuel salt inventory. The applicant should identify accident initiators that result in a decrease in fuel salt inventory such as valve leakage, pipe leakage, and inter-system leakage. The limiting accident progression and consequences should be discussed including the physical system response and any mitigating effects of included ESF systems. Any means of identifying fuel salt leakage should be identified.

13.1.3 Increase in Fuel Salt Inventory

Fuel salt inventory can be increased from several sources. These include the normal fuel and salt makeup tanks (systematic overfill) as well as inadvertent sources such as primary heat exchanger leakage or inter-system leakage. The applicant should identify accident initiators that result in fuel salt inventory increases. The limiting accident progression and consequences should be discussed including the physical system response and any mitigating effects of included ESF systems. Any means of identifying fuel salt increases should be identified.

13.1.4 Reduction in Cooling

The effect of reduction in cooling should be considered for all MSRs. The effects of a loss of cooling accident include high fuel salt temperature, potentially adverse chemical effects, excessive thermal stress on the fuel system boundary, and an induced reactivity insertion. Available MSR cooling systems include all systems and components that remove heat from the reactor vessel and the fission gases as identified in

Section 5.1 of the SAR. These cooling systems can include the normal heat dissipation flow path, emergency system paths, and other diverse cooling systems such as cooling for the off-gas handling system. A reduction in MSR cooling can be caused by several initiators including the following:

- Loss of electrical power
- Loss of forced circulation
- Failure of active components in the primary cooling system or the normal heat dissipation system
- Primary heat exchanger tube rupture
- Flow obstruction in primary heat exchangers
- Full or partial freezing in system piping
- Loss of cooling to the gas management system
- Loss of cooling to the drain tank (if applicable)

The limiting accident progression and consequences should be discussed including the physical system response and any mitigating effects of included ESF systems. Any means of identifying active reactor core cooling changes should be identified.

13.1.5 Reactivity and Power Distribution Anomalies

For MSRs, reactivity anomalies can become initiating events that lead to a challenge to the integrity of the fuel system boundary. The following MSR-specific reactivity insertion events should be considered along-side more traditional reactivity insertion events, such as uncontrolled control element actuation, misaligned control elements, and experiment malfunction (if applicable).

13.1.5.1 Pressurization of the fuel fluid or excessive active reactor core voiding and subsequent bubble collapse.

This event should be considered for an MSR given the large, negative void reactivity coefficient characteristic of this reactor technology. The physical system response and any mitigating effects of included ESF systems should be included in the discussion of system pressurization or excessive active reactor core voiding and subsequent bubble collapse.

13.1.5.2 Excessive cooling

The effect of overcooling the fuel salt should be considered for all MSRs because of the large, negative temperature reactivity coefficient characteristic of this reactor type and the potential for a flow blockage as a result of fuel salt freezing. The physical system response and any mitigating effects of included ESF systems should be included in the discussion of overcooling. MSR fuel salt overcooling can be caused by several initiators including the following:

- Excessive cooling in the primary cooling system or the heat dissipation system
- Loss of heat tracing

13.1.5.3 Fuel or fuel salt injection

MSRs are expected to have systems for defueling and refueling. Therefore, failures in plant systems or control systems could add fuel or fuel salt to a critical reactor, thus increasing reactivity. The physical system response should be included in the discussion of unintended or excess fuel additions.

13.1.5.4 Interfacing Systems

Liquid-fueled reactors are expected to have systems for mixing fuel salt and adjusting active reactor core chemistry, such as a fuel salt cleanup or polishing system. Failures in such plant systems or control systems could add non-fuel material into a critical reactor, thus impacting reactivity. The physical system response should be included in the discussion of unintended interfacing system responses. For example, unintended operation of the cleanup system could affect reactivity through removal of fission product poisons or introduction of a fluid with a low boiling point could cause system pressurization.

13.1.5.5 Adverse active reactor core geometry changes

Failure of the moderator material, such as graphite in an epithermal MSR or sloshing of the fuel salt because of vibration or other mechanisms could affect the active reactor core moderation characteristics and increase reactivity. The physical system response should be included in the discussion of adverse active reactor core geometry changes.

13.1.5.6 Loss of forced circulation including loss of electrical power

Reactor flow affects the location of delayed neutron production. Sudden changes in flow will affect the number of delayed neutrons produced in the active reactor core and subsequently impact active reactor core reactivity. The physical system response and any mitigating effects of included ESF systems should be included in the discussion of loss of flow.

The limiting accident progression and consequences should be identified for each applicable reactivity anomaly. Limiting initial active reactor core parameters should be bounded by technical specifications. Any means of identifying active reactor core reactivity changes should also be identified.

13.1.5.7 Spurious control element actuation

Spurious actuation or manipulation of control elements could lead to an increase in reactivity or a power distribution anomaly. The physical system response and any mitigating reactivity coefficient effects should be included in the discussion of spurious actuation or manipulation of control elements.

13.1.5.8 Misaligned control elements, if applicable

MSR control elements can be solids, liquids, or gases, and they can be passively or actively positioned. If misalignment of a control element is applicable, it could lead to a power distribution anomaly. The physical system response should be included in the discussion of a control element misalignment.

13.1.5.9 Experiment malfunction, if applicable

An experiment malfunction, if applicable, could lead to an increase in reactivity or a power distribution anomaly. The physical system response as it affects reactivity or power distribution anomalies should be included in the discussion of experiment malfunctions in this section. Other experiment malfunction effects are discussed in Section 13.1.7.

13.1.6 Mishandling or Malfunction of Fuel

Since the fuel in the MSR is liquid, fuel mishandling events can be characterized as fuel salt spills or leaks where some amount of this fuel could gather or migrate to unintended locations. One immediate concern for the outcome of such a spill or leak is the accumulation of a sufficient mass of fuel in a geometry leading to an unintended criticality.

In addition to fuel salt inventory decreases on the active reactor core function, fuel leakage or excessive fuel leakage should be considered within the context of this class of accidents.

Fuel malfunction events for an MSR may be thought of as those events where the physical or chemical form of the fuel salt undergoes a change resulting in adverse chemical effects such as fuel precipitation or excessive corrosion. The limiting accident progression and consequences should be discussed. Any means of identifying fuel spills or fuel salt chemistry changes should also be identified.

13.1.7 Experiment Malfunction

The conduct of experiments is one of the important functions of a non-power reactor. Experiments may contain fuel, explosives, and highly reactive materials. Failure or malfunction of experiments may initiate accidents. In some cases, particularly for lower power non-power reactors, failure or malfunction of an experiment may be the MHA, especially if fueled experiments are allowed by the facility license. Initiating events for this class of accidents include the following:

- loss of cooling or other malfunction in a fueled experiment resulting in liquification or volatilization of the fissile component
- loss of cooling capability in a strongly absorbing non-fueled experiment resulting in absorber failure and rapid increase in reactivity
- placement of an experiment component in an unplanned location, causing effects that were not evaluated
- failure of an experiment containing highly reactive contents
- failure of an experiment and release of corrosive materials in the reactor coolant
- detonation of an explosive experiment
- failure of an experiment that affects fuel salt flow, including localized fuel salt solidification due to overcooling by the failed experiment

The limiting accident progression and consequences should be discussed. Experiment technical specifications should identify limiting experiment types and associated reactivities. Any means of identifying experiment malfunctions should also be identified.

13.1.8 External Events

These events include natural phenomena, such as extreme winds, tornadoes, floods, seismic events, or volcanism, as well as manmade events, such as explosions or toxic releases in the vicinity of the reactor building. For example, the impact of seismically induced changes in the geometry of the fuel solution should be considered. The limiting accident progression, barrier challenges, and consequences should be discussed.

13.1.9 Mishandling or Malfunction of Equipment

The applicant should consider the consequences of mishandling or malfunction of equipment that could result in the spillage or leakage of contaminated fluids. Additionally, since fission gases are not retained in the fuel for an MSR, the applicant should consider the leakage or release of fission gases. For example, a stuck-open relief valve or inadvertent opening of a valve in the waste gas storage systems or waste holdup tanks is an equipment malfunction that could allow radioactive gases to leak out of the fuel system boundary at an excessive rate. Such an equipment malfunction would constitute a loss of integrity of the boundary of the waste gas holding tank that allows escape of fission gas and radiolysis gas into the building confinement. All facility and system interaction events should be considered (e.g., dropping a heavy object such as a transfer cask).

13.1.10 Loss of Normal Electrical Power

This accident initiator could result from onsite or offsite power interruptions. Emergency power supplies, if provided, are assumed to operate. However, the applicant may want to analyze the effects of failure of emergency power. This impacts the decay heat removal analysis discussed in Section 13.1.4

13.2 ACCIDENT ANALYSIS AND DETERMINATION OF CONSEQUENCES

In this section of the SAR, the applicant should discuss each event giving information consistently and systematically that will lead to a clear understanding of the specific reactor and facilitate comparisons with similar reactors. Many of the steps used to select the limiting event in each category may be semi-quantitative. However, the analysis and determination of consequences of the selected limiting events should be as quantitative as possible. Applicants should take the following steps when selecting the limiting event in each category:

1. State the initial conditions of the reactor and equipment. Discuss relevant conditions depending on fuel burnup, experiments installed, active reactor core configurations, or other variables. Use the most limiting conditions in the analyses. Inform plant technical specifications as appropriate.
2. Identify the causes that initiate the event; the causes may include equipment malfunction, operator error, solubility, precipitation, chemical accidents, other natural phenomenon, or one caused by humans. Base the scenario on a single initiating malfunction, rather than on multiple causes.
3. List the sequence of events, assumed equipment operation and malfunction, and operator actions until a final stabilized condition is reached. Discuss functions and actions assumed to occur that change the course of the accident or mitigate the consequences, such as reactor scrams or initiation of such ESFs as emergency cooling. If credit is taken for mitigation of the accident consequences, discuss the bases used to determine that the systems are operable and discuss the system functions.
4. Classify damage that might occur to components during the accident until the situation is stabilized. Discuss all components and barriers that could affect the transfer of radiation and radioactivity from the reactor to the public and that ensure continued stability of conditions after the accident.
5. Prepare realistic analyses to demonstrate a detailed, quantitative evaluation of the accident evolution, including the performance of all barriers and the transport of radioactive materials to the unrestricted area. Include the assumptions, approximations, methodology, uncertainties and degree of conservatism, margins of safety, and both intermediate transit and ultimate radiological conditions. Justify the methods used. Further, ensure that the information is sufficiently complete to allow the results to be independently reproduced or confirmed. Demonstrate the validation of the computational models, code, assumptions, and approximations by comparison with measurements and experiments when possible. Describe in detail computer codes that are used; include the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions for similar operating non-power MSRs. Include estimates of the accuracy of the analytical methods. Chapter 11, “Radiation Protection Program and Waste Management,” of the SAR, discusses the methods and assumptions used to analyze the release and dispersion of radioactive materials from normal operations. Adapt those methods as appropriate for accident analyses.

6. Define and derive the radiation source terms, if any are involved. Include in the source terms the quantity and type of radionuclides that could be released, their physical and chemical forms, and the duration of potential releases. Describe potential radiation sources that could cause direct or scattered radiation exposure to the facility staff and the public.
7. Evaluate the potential radiological consequences using realistic methods. Discuss the degree of conservatism in the evaluation. For example, include a discussion of the degree of conservatism introduced by the use of postulated release fractions or assumptions of an infinite hemispherical cloud.
8. Include environmental and meteorological conditions specific for the facility site to illustrate consequences of the accident. Give an account of exposure conditions for the facility staff until the situation is stabilized (including staff evacuation and reentry), the most exposed member of the public in the unrestricted environment until the accident conditions are terminated or the person is moved, and the integrated exposure at the facility boundary and the nearest permanent residence. The radiological consequences should include external and internal exposures. Address contamination of land and water where applicable and include exposure control measures to be initiated.

13.3 SUMMARY AND CONCLUSIONS

In this section of the SAR, the applicant should summarize the important conclusions about the postulated accidents and the potential consequences. The applicant should compare the projected radiological consequences with the acceptance criteria discussed previously in this chapter. The information should demonstrate that all reasonable measures have been incorporated into the facility design bases to prevent undue radiation exposures and contamination of the unrestricted environment. The discussions should show that ESFs have been incorporated where necessary to mitigate accident consequences to acceptable levels.

13.4 REFERENCES

Memorandum of Understanding between the U.S. Nuclear Regulatory Commission and the Occupational Safety and Health Administration, September 2013 (ADAMS Accession No. ML11354A432)

13.4.1 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 of Power Test Reactor Sites,” TID-14844, March 23, 1962.

Baker, L., Jr., and L. C. Just, *Studies of Metal-Water Reactions at High Temperatures,* III, “Experimental and Theoretical Studies of the Zirconium Water Reaction,” ANL 6548, Argonne National Laboratory, Illinois, 1962.

Baker, L., Jr., and R. C. Liimatakinen, “Chemical Reactions” in Volume 2 of *The Technology of Nuclear Reactor Safety,* Thompson and Beckerly (eds.), The MIT Press, Cambridge, Massachusetts, pp.419-523, 1973.

Muhlheim, M. D., “Identification of Initiating Events for aSMRs,” ORNL/TM-2013/513, Oak Ridge National Laboratory, June 2013.Nuclear Energy Institute, “Risk-Informed, Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," Technical Report 18-04, Revision N, September 2018.

13.4.2 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, p.319, 1981.

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, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

U.S. Nuclear Regulatory Commission, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," Regulatory Guide 2.6, September 2017.

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

13.4.3 Molten Salt Reactors

Beall, S. E., et al., "MSRE Design and Operations Report: Part 5, Reactor Safety Analysis Report," ORNL-TM-732, August 1964.

Bettis, E. S., et al., "Design Studies of a Molten-Salt Reactor Demonstration Plant," ORNL-TM-3832, June 1972.

Holcomb, D. E., et al., "Fast Spectrum Molten Salt Reactor Options," ORNL/TM-2011/105, Oak Ridge National Laboratory, July 2011.

Holcomb, D. E., et al., "Molten Salt Reactor Initiating Event and Licensing Basis Event Workshop Summary," ORNL/TM-2019/1246, Oak Ridge National Laboratory, July 2019.

Robertson, R. C., et al., "Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor," ORNL-4541, June 1971.

14 TECHNICAL SPECIFICATIONS

Chapter 14 of this guide is applicable to the development of technical specifications for a non-power MSR. In this chapter of the SAR, the applicant should discuss the development of the facility technical specifications. This chapter of the format and content guide discusses the contents of Chapter 14 of the SAR and presents guidance in Appendix 14.1 on the format and content of technical specifications for MSR s.

NRC requires each applicant for a license to operate a non-power reactor to develop technical specifications that state the limits, operating conditions, and other requirements imposed on facility operation to protect the environment and the health and safety of the facility staff and the public in accordance with 10 CFR 50.36. The technical specifications are typically derived from the facility descriptions and safety considerations contained in the SAR and represent a comprehensive envelope of safe operation.

Applications for construction permits or operating licenses and renewals of operating licenses must contain proposed technical specifications that will be incorporated in the operating license. During its review of the application, the NRC staff will review the SAR and proposed technical specifications to ensure they are complete comprehensive and that the environment and public health and safety will be protected. After final acceptance by the NRC staff, the technical specifications will be included as Appendix A to the operating license.

The format and content of the technical specifications discussed in Appendix 14.1 follow the format of American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1. Examples of the format and content of technical specifications can be found in previously accepted and approved technical specifications for non-power reactors of similar design, operating characteristics, site and environmental conditions, and use.

This chapter of the SAR is normally very short. The applicant should be able to state conclusively that the technical specifications were prepared following an accepted format, that normal operation of the reactor within the limits of the technical specifications will not result in offsite radiation exposure in excess of 10 CFR Part 20 guidelines, and that the technical specifications limit the likelihood and consequences of malfunctions. The reader is referred to the technical specifications, which are in a document separate from the SAR. The technical specifications are neither derived nor justified in this chapter of the SAR. They are determined by the analyses that appear in the other chapters of the SAR. Each of the technical specifications should be supported by the SAR, and it is useful to refer to the supporting SAR analysis in the basis of each technical specification.

In Appendix 14.1, every section of ANSI/ANS 15.1 is addressed. If ANSI/ANS 15.1 should be modified or clarified to provide acceptable technical specifications, additional guidance is given. Sections that provide acceptable guidance as written are noted. The applicant should propose and justify those technical specifications that are applicable to the reactor design and utilization under consideration.
The standard format and content of technical specifications for non-power reactors are presented in Appendix 14.1. The numbering system (Sections 1 through 6.8) corresponds to the numbering system in ANSI/ANS 15.1.

14.1 REFERENCE

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, “The Development of Technical Specifications for Research Reactors,” ANS, LaGrange Park, Illinois, 2007 (R2013).

APPENDIX 14.1 FORMAT AND CONTENT OF TECHNICAL SPECIFICATIONS FOR NON-POWER MSRS

The numbering system in this appendix (Sections 1 through 6.8) corresponds to the numbering system in ANSI/ANS 15.1. Because ANSI/ANS 15.1 is written to also apply to facilities of the Department of Energy, some terminology used in the standard may be more general than that used by NRC (e.g., the standard uses the term “responsible authority” instead of “U.S. Nuclear Regulatory Commission”).

1. INTRODUCTION

1.1 SCOPE

NRC accepts the guidance provided in this section of ANSI/ANS 15.1 This section confirms that the technical specifications for non-power reactors should include all the categories in 10 CFR 50.36 for production and utilization facilities.

1.2 APPLICATION

1.2.1 Purpose

NRC accepts the guidance provided in this section of ANSI/ANS 15.1. Technical specifications represent a set of operating requirements for a reactor that the applicant and NRC have agreed on. The specifications become part of the operating license.

1.2.2 Format

Sections of the technical specifications should be numbered as indicated in Section 1.2.2 of ANSI/ANS 15.1. Subsections may be left out if not applicable for a particular reactor or may be altered if necessary, but the subsections included should be arranged in consecutive numerical order.

For individual specifications in Sections 2, 3, 4, and 5, applicability, objective, specification, and basis information should be included in the specified format. Although the standard does not include basis information for specifications in Section 5 of the technical specifications, the regulations in 10 CFR 50.36 require it. For Section 6 of the technical specifications, the specifications may be stated without providing applicability, objective, or basis.

Technical specifications that use the SAR as a basis should explicitly reference the SAR section number. In addition, any other sources used to support the technical specification should be explicitly referenced.

1.3 DEFINITIONS

NRC and the non-power reactor community have agreed on most of the definitions given in this section of ANSI/ANS 15.1. Those applicable to a particular facility should be included verbatim. Facility-specific definitions may be added to clarify terms referred to in the technical specifications. Modifications and additional definitions presented below help clarify the meaning of terms used in ANSI/ANS 15.1.

The following definitions should be modified as indicated:

- **Core configuration.** The active reactor core configuration includes the fuel salt, neutron moderator, neutron reflector, control elements, neutron startup source, incore cooling components, and any experimental facilities.
- **Experiment.** Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate nonroutine reactor characteristics or that is intended for irradiation within the reactor structure, on or in a beam port, or irradiation facility. Hardware rigidly secured to a vessel or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.
- **License.** The written authorization, by the U.S. Nuclear Regulatory Commission, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.
- **Reactor secured.** A reactor is secured when
  - *Either* there is insufficient fissile material present in the active reactor core to attain criticality under optimum available conditions of moderation and reflection;
  - *Or the following conditions exist:*
    a) The minimum number of control elements and other safety devices are in their shutdown states, as required by technical specifications;
    b) The console key switch is in the off position and removed from the lock;
    c) No work is in progress involving fuel salt within the fuel system boundary, the active reactor core or vessel structure, installed control elements, or means of manipulating control elements unless they are decoupled from the control elements;
    d) No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.
- **Reference core condition.** The reference core condition is the reactivity condition of the active reactor core at a reference set of parameters (e.g., fuel salt temperature, reactivity worth of neutron poisons, etc.)
- **Shutdown margin.** Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive control element in the most reactive state, and the non-scramable control elements in their most reactive states and that the reactor will remain subcritical without further operator action.

The definitions for “rod control” and “responsible authority” should be replaced with the following definitions:

- **Control element(s).** Object(s) employed to adjust reactivity. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases and can be passively or actively positioned.
- **Responsible authority.** The U.S. Nuclear Regulatory Commission.

The following definitions should be added:

- **Coating or Cladding.** Intervening, protective layer of material between the fuel salt and the structural container alloy. Also included are surface modifications of the structural container alloy to enhance its chemical or mechanical performance by altering its microstructure or composition (e.g. carbiding, phosphiding, or nitriding the surface).
- **Functional containment.** A barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions.

- **Secured shutdown.** Secured shutdown is achieved when the reactor meets the requirements of the definition of “reactor secured” and the facility administrative requirements for leaving the facility with no licensed reactor operators present.
- **Shutdown reactivity.** Shutdown reactivity is the value of the reactivity of the reactor with all control elements in their least reactive condition. The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

If a defined term in ANSI/ANS 15.1 is not used in the technical specifications, the applicant should not include it in the definitions. For example, the definitions for “rod regulating” and “rod transient” are not applicable to non-power MSR operation and should be disregarded and omitted from the technical specifications.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

All reactor licensees are required by 10 CFR 50.36(c) to specify safety limits in the technical specifications. The regulations state that “Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.” Therefore, safety limits should be placed on important process variables identified in the facility SAR as necessary to reasonably protect the integrity of the primary barriers against the uncontrolled release of radioactivity. For nonpower MSRs, the radioactivity of concern is generally the fission products in the fuel salt. For homogeneous-core reactors, this primary barrier may be the fuel salt, a cladding on the fuel system boundary, the fuel system boundary, the gas management system, the drain tank, the fuel salt cleanup system, or other components that contain the fuel and the fission products. Chemistry limits should be considered within technical specifications to help maintain the material condition of the fuel system boundary and any associated cladding (redox controls).

Reactor conditions and safety limits should be developed to avoid failure of the fuel system boundary integrity and should be supported by SAR analyses. The applicant should make maximum use of the appropriate references, some of which are listed at the end of this appendix.

The applicant should base SAR analyses on the applicable fuel salt development test results to ensure fuel system boundary integrity under all operating conditions.

2.1.1 Important Process Variables

ANSI/ANS 15.1 proposes a list of parameters that may be acceptable as process variables for non-power reactors and implies that safety limits will be directly measurable parameters. Not all safety limits for non-power reactors must be monitored and actually measurable. Safety limits could be inferred from limitations on other process variables having significant safety functions.

The fuel salt loop in an MSR operates by forced convection within the fuel system boundary. Active reactor core flow directly correlates to delayed neutrons produced in the active reactor core region. Therefore, flow will be a primary means by which MSR power is controlled. Reactor fuel can typically be added online, and fission products can be removed from the fuel salt online through a fuel salt cleanup system and gas management system. Under normal operation, the fuel salt moves through the active reactor core where it is brought into a critical configuration and heat is added to the fuel salt. Subsequently, the fuel salt continues through a fuel salt/primary cooling system salt heat exchanger where heat is transferred from the fuel salt to the primary cooling system, which uses a compatible nonfuel salt. The fuel salt then returns to the active reactor core. The non-fuel primary cooling system salt passes heat to a heat dissipation system through subsequent system interfaces. Heat can also be removed by natural circulation following a loss of power and a reactor scram. Safety limits for non-power MSRs are discussed in Section 2.1.2 below.

2.1.2 Criteria—Reactors with Engineered Cooling Systems

This section of ANSI/ANS 15.1 should be modified for non-power MSRs as follows. Operation of the primary cooling system for MSRs with forced-convection primary cooling maintains fuel salt temperature within acceptable limits to ensure fuel system boundary integrity. Important parameters include fuel salt temperature, fuel salt flow rate, primary cooling system flow rate, fuel salt volume, fuel salt chemistry, and reactor power level. These parameters should be controlled, measured, and maintained within the limits determined by the safety analyses. When all values are jointly maintained within the limits determined by the safety analyses, fuel system boundary integrity will not be lost. These parameters are important process variables on which safety limits should be established and specified in the technical specifications. The analyses should range over all physical and engineering parameters of the fuel salt components, fuel salt composition, the active reactor core configurations, and the coolant systems, and should also include consideration for uncertainties.

All reactors having forced-convection fuel salt systems should have specifications ensuring operability of systems and reactor configurations for fail-safe changeover from normal forced-convection to emergency forced-convection fuel salt flow or natural-convection fuel salt flow. For reactors that will operate with both natural-convection fuel salt flow and forced convection fuel salt flow, safety limits should be specified for appropriate process variables in both modes of operation. All non-power reactors should be designed so that both fission heat and decay heat can be dissipated without fuel system boundary damage. The analyses also should show that the safety limits are not exceeded during all anticipated modes of operation.

2.1.3 Criteria—Reactors without Engineered Cooling Systems

This section applies to any non-power MSR design proposed with natural convection flow in the fuel salt loop. In this case, only Section 2.1.3(2) of ANSI/ANS 15.1 is acceptable to meet the requirements of 10 CFR 50.36(c). Section 2.1.3(2) of ANSI/ANS 15.1 should be modified as follows. For reactors that circulate fuel salt by natural thermal convection, thermal-hydraulic coolant parameters are not separately controllable. The applicant should ensure fuel system boundary integrity and should note appropriate parameters chosen for safety limits in the technical specifications.

High fuel salt temperature and lack of adequate fuel salt chemistry parameter control are the likely precursors of fuel system boundary failure. Therefore, a maximum allowable fuel salt temperature safety limit and appropriate fuel salt chemistry parameters should be established within which fuel system boundary integrity is ensured. On the basis of this fuel salt temperature, a power level should be calculated using an appropriate margin which ensures that the fuel salt remains below the fuel temperature safety limit.

If the license will contain a provision to measure fuel salt temperature, the maximum fuel salt temperature in the vessel would be the parameter on which a safety limit is established. The SAR should show the relationship between the measured fuel salt temperature and the maximum fuel salt temperature for the proposed reactor conditions. If there is no provision for measuring fuel salt temperature directly, the calculated power level should be selected as the safety limit, on the basis of the maximum allowable fuel salt temperature and appropriate margin. However, the basis for the safety limit still should be the maximum allowable fuel salt temperature.

2.2 LIMITING SAFETY SYSTEM SETTINGS

The NRC accepts the guidance of this section of ANSI/ANS 15.1 for limiting safety system settings. The SAR should address normal operating conditions, off-normal operations, and all pertinent postulated accident scenarios. For each parameter on which a safety limit is established by the SAR, a protective channel should be identified that prevents the value of that parameter from exceeding the safety limit. The calculated setpoint for this protective action, providing the minimum acceptable safety margin considering process uncertainty, overall measurement uncertainty, and the transient phenomena of the process instrumentation, is defined as the “limiting safety system setting (LSSS). Because the LSSSs are analytical limits, the protective channels may be set to actuate at more conservative values. The more conservative values may be established as limiting conditions for operation (LCOs).

Such LCOs may be determined on the basis of experience, which has shown that safety system channels can be set readily within 20 percent of the normal operating value for a measured parameter, if the LSSS is not exceeded, without undue interference to operations. In many cases, the LCO can even be within 10 percent of the operating value. The SAR justification for LSSSs and LCOs should be referenced.

2.2.1 Criteria-Reactors with or without Engineered Cooling Systems

NRC accepts the guidance of this section of ANSI/ANS 15.1 for the forced convection primary cooling mode of operation. For reactors licensed to operate with forced-convection primary cooling, this specification should list the LSSS derived in the SAR for each reactor parameter for which a safety limit was established. The bases part of this specification should indicate the SAR assumptions and limits of uncertainty for each analyzed LSSS. For reactors licensed to operate in forced and natural-convection cooling modes, appropriate LSSSs should be listed for both modes.

NRC substitutes the following guidance for reactors that will be licensed to operate without forced-convection primary cooling. Section 2.1.3 (above) requires that safety limits be established by SAR Analysis for all licensed reactors; therefore, channels should be established on the basis of SAR analysis to not violate each of these safety limits. Calculated LSSSs defined in Section 2.2 of ANSI/ANS 15.1 and in this appendix should be provided as technical specifications.

3. LIMITING CONDITIONS FOR OPERATIONS

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. LCOs are derived from the safety analyses in the SAR, which provide the bases for the LCOs. LCOs are implemented administratively or by control and monitoring. circuitry to ensure that the reactor is not damaged, that the reactor is capable of performing its intended function, and that no one suffers undue radiological exposures because of reactor operations.

NRC accepts the guidance of this section of ANSI/ANS 15.1 as amplified in the sections that follow. Many of the LCOs have evolved from experience. Many are facility specific, depending on reactor type, operating characteristics, and site location. NRC accepts the LCOs discussed in this section provided that the applicant justifies them and shows the applicability to the specific facility. Additional specifications may be appropriate for unique facility designs or experimental features or for additional conservatism in operations required by the applicant or NRC. As noted above, LCOs can in many cases be set within 10 percent of the normal operating level of a parameter. Specifications on surveillance intervals for LCOs and other parameters and facility design features are given in Sections 4 and 5, respectively, of ANSI/ANS 15.1 and this appendix. LCOs should be provided as outlined in the remainder of this section.

3.1 FUEL SALT AND FUEL SYSTEM BOUNDARY PARAMETERS

(1) Excess Reactivity

The upper limit for allowed excess reactivity should be specified. The referenced SAR analyses should discuss all operations that require excess reactivity and the safety implications for the excess reactivity proposed. The discussions should include operational flexibility, potential accidents, and relationship to shutdown margin. The SAR (Chapter 4, "Molten Salt Reactor Description," and Chapter 13, "Accident Analyses") should contain a discussion of the safety implications of the excess reactivity, including the following:

- resultant shutdown reactivity with all control elements in their least reactive condition
- online addition of fuel or fuel salt
- operation of the fuel salt cleanup system on removal of fission product poisons
- effects on the reactor of any credible rapid removal of a control or safety element
- effects of fuel salt pump speed changes on reactivity
- potential effects of other maximum credible rapid additions of excess reactivity
- possible reactivity changes caused by experiment failure or displacement
- interrelationship between shutdown margin and excess reactivity

If none of the postulated events would lead to loss of fuel integrity or to uncontrolled release of radioactivity, the proposed excess reactivity would be acceptable.

(2) Shutdown Margin

A single value for the shutdown margin, as defined in Section 1.3 (above), should be specified. The specification should state that compliance with the shutdown margin takes precedence over the excess reactivity specification. In addition, other reactor parameters that apply to the shutdown margin should be stated. These should include reference core condition, active reactor core configuration (e.g., liquid volume and solid volume), and the status of experiments (e.g., movable experiments in their most reactive state). The value of the shutdown margin should be large enough to be readily determined experimentally, for example, $\geq 0.5\%$ $\Delta k/k$ or ≥ 0.50 dollar.

(3) Core Configurations

The applicant should specify information on the active reactor core geometry, the neutron moderator, materials, reflectors, any cooling requirements, and configurations. The limiting active reactor core configurations for an MSR are the conditions that would yield the highest power density, the highest excess reactivity, and other possible limiting parameters that are of safety interest using the fuel salt specified for the reactor. All other active reactor core configurations should be demonstrated to be encompassed by the safety analysis of the limiting active reactor core configuration.

The applicant should specify special active reactor core configurations, experimental facilities internal to the active reactor core, special neutron reflectors, burnable poisons, or mixed fuel types assumed in the SAR. The following specifications should be included in the LCO for active reactor core configurations:

- The use of in-core experimental facilities is permissible.

- Conditions under which fuel or fuel salt can be added or removed from the vessel. MSRs typically operate with minimal excess reactivity and fuel is typically added online.
- If physical control elements need to be removed from the active reactor core for inspection, an LCO should state the negative reactivity necessary in the active reactor core before a control element can be removed.

Non-power MSRs should be designed with reactivity and void coefficients and a power defect sufficiently negative that many reactor transients are inherently counteracted to avoid loss of fuel system boundary integrity. Although the individual reactivity coefficients and power defect are addressed in the specification below, this LCO should be used to develop specifications on allowed active reactor core configurations to ensure the assumptions used in the development of limits on those parameters are met.

The specified conditions of active reactor core configuration are acceptable to NRC if the SAR shows that none of the conditions analyzed could lead to loss of fuel system boundary integrity, uncontrolled release of radioactivity, or potential exposures exceeding 10 CFR Part20.

(4) Fuel Composition Changes

A heterogeneous core life cycle is limited by fuel burnup. However, a homogenous core does not have the same fuel burnup limitations. The fuel composition in a homogenous core will change over time, impacting active reactor core reactivity. The applicant should propose a chemistry specification to account for operational changes in active reactor core reactivity with fuel composition changes, plutonium buildup, and poisons, both fission products and those added by design. The reactivity impacts of fission gas and void formation, fission product gas removal, and fuel salt addition should be considered.

The TS should include LCOs for operational limits or ranges on parameters such as fission gas evolution, power oscillations, precipitation from fuel salt, fuel salt density, viscosity, actinide-salt ratio, fuel salt redox condition, free fluorine or chlorine, oxygen level, temperature and pressure extremes or distributions, and contaminants such as oil or metal oxide precipitants. There should be surveillance requirements specified in Section 4 of the TSs for sampling fuel salt chemistry during both operating and shutdown periods. The bases should clearly address the appropriate parameter ranges and give meaningful references including chemistry limits to mitigate corrosion. The SAR should justify the values for the chemistry requirements for the particular reactor.

LCOs are acceptable if they are analyzed in the SAR. The analyses should verify for these fuel parameter conditions that the fuel will not exceed safety limits for normal and off-normal operations.

(5) Reactivity Coefficients

Non-power reactors should specify reactivity coefficients for fuel temperature, moderator temperature, and void volume and a power defect if these parameters could vary unacceptably with reactor operation. In many cases, these measurements need only be made during the startup testing program. The net effect of the coefficients and the power defect should be negative over most of the range of reactor operations. The SAR analyses of both routine operation and potential accident scenarios should show that the net negative effect of these active reactor core characteristics is sufficient to mitigate any anticipated event or postulated accident scenario. Reasonable values should be designed into the reactor (e.g., by under moderation of the neutron spectrum). Values for surveillance should be specified for those negative reactivity coefficients and the negative power defect that can be measured. The values of the coefficients and the power defect are acceptable if they ensure that the assumptions and initial conditions of the analyses are enveloped to prevent compromise of the fuel integrity during reactor transients and other applicable accident scenarios.

(6) Fuel Salt Level Limits

Both the fuel salt flow and adequate operating parameters such as pressure depend on the available volume of fuel salt. Operation of the gas management system is also impacted by high fuel salt levels. Fuel salt volume and any associated fuel salt makeup system should be encompassed by the safety analysis in the SAR as the basis.

(7) Detection of Leakage or Loss of Fuel Salt

If fuel salt leakage could lead to an uncontrolled release of radioactive material (see item 8) to the environment, an LCO should state the need for operability of leakage detection systems. Examples of this type of system would be instrumentation that monitors the fuel salt level or the pressure difference between the fuel salt and primary cooling systems at the heat exchanger to detect conditions that would allow loss of fuel salt in the event of a heat exchanger leak.

(8) Detection of Fission Product Activity

The technical specifications should provide for prompt detection of fission products escaping from the fuel system boundary, the gas management system, the fuel handling system, or the fuel cleanup system. The method could be a strategically located continuous air monitor in the reactor room or in a ventilation duct. Temporary substitutions, in case the fission product monitor is inoperable, should follow guidance in Section 3.7.1 of ANSI/ANS 15.1. This specification may be combined with the specification discussed in Section 3.7.1(2) on fission product monitors.

The specified fission product monitor should be able to initiate action, such as a reactor scram, reactor room isolation, or an alarm, as appropriate. The SAR should provide the bases and describe how fission products are distinguished from other airborne radioactivity.

(9) Hydrogen Concentration (Gas Management System) Limits

If the SAR has shown any of the isotopes of hydrogen (hydrogen, deuterium, and tritium) to be a significant risk to personnel or the facility, an LCO should provide for detection or adequate control, as discussed in the SAR.

The technical specifications for the gas management system should provide for prompt detection of a malfunction or failure of components that could cause excessive system pressure, positive reactivity feedback to the fuel solution, and operating instability. All of the process variables controlling the gas management system must be within appropriate limits as analyzed in the SAR.

(10) Emergency Cooling Systems

If the SAR indicates a need for supplemental vessel cooling to mitigate a loss of integrity of the fuel system boundary including a fuel salt drain tank (if applicable), the technical specifications should contain an LCO requiring an operable and adequate system. The system should satisfy the cooling requirements for the SAR scenario and should not depend on continued availability of normal electrical service.

A table should specify any required emergency cooling system (ECS) channels and setpoints, the minimum number of channels, other functions performed by the channel and reactor operating mode. The safety limits that the ECS protects and accident mitigation provided should be discussed in the basis for the table. The SAR should justify whether ECS is required or not.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

(1) Operable Control Elements:

The number and type of operable control elements should be specified. No minimum number of operable control elements is prescribed for non-power reactors. The specification regarding the number of operable control elements is acceptable if the excess reactivity and shutdown margin specifications required by the SAR analyses can be ensured for all operating conditions. The individual or total reactivity worths need not be specifically listed. Although MSR power level is generally controlled by fuel salt pump speed, an element of lesser worth might be designated the “regulating element” and be used as a fine power adjustment mechanism. In some cases, the worth of a control element(s) connected to an automatic control system (which can add reactivity) may be limited to a maximum amount that was assumed in the SAR in this LCO. This regulating element need not have scram capability, but elements without scram capability should not be used when showing compliance with shutdown margin requirements. (Other elements of greater worth, with an automatic protective (scram) function, should be capable of achieving the specified shutdown margin.

The maximum scram time should be specified for each scrammable element. The specification should ensure that the insertion times are consistent with the SAR analysis of reactivity required as a function of time to terminate a reactivity addition event accounting for measurement and calculational uncertainties. In most non-power reactors, full element insertion time in the absence of excess mechanical friction or interference is less than 1 second. If a specification proposes a longer scram time, it requires appropriate SAR analysis. NRC finds it acceptable to shut down a non-power reactor by intentionally scramming the control elements.

(2) Reactivity Insertion Rates

The maximum rates of adding positive reactivity should be specified for the control elements and changes in fuel salt pump speed. The specification should explicitly state that gang or multiple element manipulation is allowed. Control elements(s) connected to an automatic control system may have maximum rates of reactivity addition that differ from the rest of the control elements, The acceptable rates should be based on the SAR, including inadvertent addition of ramp reactivity at the maximum rate for the most conservative power, element position, and reactor conditions.

(3) Scram Channels

A table should specify all required scram channels and setpoints, the minimum number of channels, other functions performed by the channel and reactor operating mode, such as steady-state power, fuel salt circulation method, and primary cooling method, such as forced- or natural-convection primary cooling system flow. The safety limits that the scram protects should be discussed in the basis for the table. Table 14.1 shows how the information could be displayed. Reactor scrams should be based on the SAR. There should be at least two completely independent power level scram channels and they should provide diversity and redundancy.

Table 14.1. Typical required scrams and power reverses*

Channel	Setpoint** and Function	Minimum Number Required
Period safety	Scram if period ≤ 3 sec	1
Period reverse	Element run in if period ≤ 10 sec	1
Power level safety (linear and safety)	Scram if power > 100%	2
High power/no fuel salt flow	Scram if flow < analyzed value and power > 100kW	1
High power/no primary cooling system flow	Scram if flow < analyzed value and power > 100kW	1
Loss of high voltage to detectors	Scram if voltage is lost	1
Containment radiation level and building exhaust air radiation level	Scram if radiation ≥ 50 mrem/hr and concentration ≥ 2 x 10^{-4} μCi/ml	1
Manual scram switch	Scram when switch is depressed	1
Fuel salt temperature	Scram if temperature ≥ high setpoint or ≤ low setpoint based on analysis	2
Primary cooling system temperature	Scram if temperature ≥ high setpoint or ≤ low setpoint based on analysis	1
Automatic control system out of limit	Element run in if out of specification	1
Experiment	Scram if setpoint is violated	1
Console key switch	Scram if key switch in “off” position	1
Loss of site power	Scram if power is lost	1

* As illustrative values, the setpoints and channels listed do not apply to any one reactor.
** Values listed are notional limiting setpoints. For operational convenience, setpoints may be changed to more conservative values.

Historically, there have been cases in which NRC has accepted power level scrams higher than the licensed power (1.2 times licensed power level is common) if supported by the safety analysis. This power level is the only non-power reactor scram setpoint that, if reached, violates the license (maximum power level). Some licensees have incorrectly interpreted this scram setpoint at higher than the licensed power level as allowing limited operation above the license power level. Although this operation is generally not a safety concern, the NRC staff recognizes it as a regulatory problem. For example, if the reactor power measuring channels are out of calibration, it is possible that the reactor has been operated at several percent above the maximum licensed power level for a period of time. To ensure that licensed power levels are not exceeded for non-power reactor operation, the applicant may consider the option of having an LCO which has a power level scram set below the licensed power level. The NRC staff has accepted another option and, upon submittal of a license amendment request and supporting safety analysis, has approved license amendments for non-power reactors that raise the licensed power 10 percent above the power level at which the reactor will be operated. The applicant can then set the reactor scram in this 10-percent power band. This allows the operating power to remain the same while retaining the scram setpoints within the license power limit. Safety limits and associated LSSS based on power are still determined by the results of the analysis in the SAR and should not change.

The applicant can choose to continue to have power level scram setpoints above the licensed power level and can meet the license power level requirements by such other means such as administrative limits.

(4) Control Systems and Instrumentation Requirements for Operation

Technical specifications for non-power reactors should have redundant and accurate power level monitors that cover the range from subcritical source multiplication to above the full power level. Not all monitors are required to include scram capability (see Table 14.2 for a typical minimum set). These include a startup channel, linear power monitor, logarithmic power monitor, and safety channel(s). In addition, most non-power reactors have a period channel (meter), including a period scram. One should be specified as analyzed in the reactor transient response section of the SAR.

Some non-power reactors with forced-convection primary cooling have a channel that displays the radiation level in the primary flow.

Table 14.2. Typical required minimum measuring channels*

Channel	Minimum Number Required	Function
Startup	1	Monitor subcritical multiplication for startup
Power level	2	Input for safety power level scram
Fuel salt temperature	2	Input for fuel salt temperature scram
Primary cooling system temperature	2	Monitor heat removal and input for primary cooling system temperature scram
Fuel salt flow	2	Power indication and input for power/flow scram
Primary cooling system flow	2	Monitor heat removal and input for power/flow scram
Fuel salt level	1	Monitor in-leakage and gas management system operability
Fuel salt pressure	1	Monitor in-leakage
Primary cooling system pressure	1	Monitor out-leakage
Log N/period	1	Wide range power level and input for period meter and period scram
Linear power level	1	Display power for control
*As illustrative values, these channels do not apply to any one reactor. Minimum channels for a particular facility are determined from the SAR analysis.		

In past cases where digital control and safety instrumentation was used, an analog reactor protection system was specified in the technical specifications in addition to the digital system to provide diversity and redundancy. The technical specifications for digital systems (including the degree of diversity and redundancy needed) are based on the analysis in Chapter 7 of the SAR, “Instrumentation and Control Systems.”

Specifications in this section should cover the entire channel, including readout meters and recorders and the protective functions they perform, such as to prevent an LSSS from being exceeded.

Each non-power reactor should have more than one power level channel indication in the control room when operating at full power. However, because sensors and channel electronics might not be identical, the channels may indicate slightly different power levels. Power level is a principal license condition, and each applicant may consider designating a primary channel for power level monitoring. That channel should be calibrated for thermal power in the region of maximum licensed power and should be recorded in a way that allows auditing for later proof of authorized operation within the license condition. Facility procedures should identify this designated channel and allow for alternative designations using analytic comparisons to achieve operational flexibility, if necessary. Technical specifications or facility procedures that do not include this concept are acceptable to NRC.

(5) Interlocks

Required interlocks that inhibit or prevent control element manipulation or reactor startup should be specified by a table (see Table 14.3 as an example). Interlocks should be specific to the facility and should be based on the SAR. Potential interlocks include the following:

- operability of area or other radiation monitors
- experimental facilities
- confinement and ventilation systems
- detected neutrons for startup
- operability of measuring channel components, such as ion chamber power supplies and recorders as discussed in the SAR

Table 14.3. Typical required interlocks*

Channel	Minimum Number Required	Function
Recorders not operating	3	Prevent element manipulation (startup inhibit)
Neutron count rate (startup)	1	Prevent element manipulation (startup inhibit) if count rate ≤ 2 cps
Simultaneous control element manipulation	5	Prevent manipulation of 2 or more control elements
System fuel salt fill	1	Prevent fuel salt addition if fuel salt temperature is too low; fuel salt chemistry is not within specification; or control elements not manipulated as expected
Online fuel or fuel salt addition	1	Limit volume of fuel or fuel salt to be added at one time
Power/flow	1	Prevent fuel salt pump start unless control elements are at or near their most reactive condition

* Values listed are limiting setpoints. For operational convenience, setpoints may be changed to more conservative values.

If the reactor will be licensed to operate in more than one mode, the specification should include the mode for which the interlock is required. If permanent interlocks are established for special experiments, shields, or access control, they should be included in the technical specifications, as described in the SAR.

(6) Backup Shutdown Mechanisms

Most non-power reactors are required to use only control elements for shutdown. If the SAR identifies a need for backup mechanisms (e.g., fuel salt dump), they should be specified with appropriate requirements placed on their operability (LCOs).

(7) Bypassing Channels

Any individual channels identified in items 4, 5, or 6 (above) for which bypassing is allowed during reactor operation should be justified in the SAR and specified under this item. Only minimal bypassing should be permitted in safety systems and never in a system that could compromise scram capability of the other channels. Bypassing temporary scrams or interlocks associated with experiments need not be included in the technical specifications but should be addressed in specific experiment protocol.

3.3 PRIMARY COOLING AND HEAT DISSIPATION SYSTEMS

Heat from the fuel salt is normally transferred through one or more fuel salt/primary cooling system salt heat exchangers to the primary cooling system. The non-fuel primary cooling system salt passes heat to the heat dissipation system through subsequent system interfaces. On reactor shutdown, decay heat can be removed through an auxiliary heat exchanger instead of the fuel salt/primary cooling system salt heat exchanger. Some nonpower MSR designs might be capable of dissipating decay heat in the natural convection mode. Other MSR designs might use a fuel salt drain tank or other irradiated fuel salt storage tanks to allow for safe storage of the fuel salt in the event of a design-basis accident or for maintenance. Decay heat in a fuel salt drain tank is cooled separately and is discussed as a separate system. Supplemental cooling systems might also be necessary to remove heat from the cover gas space above the fuel salt and from any fuel salt cleanup system.

The basic systems required for cooling the fuel salt, the gas management system, the fuel salt cleanup system, and other components for limiting corrosion, and for monitoring primary cooling system radioactivity in non-power MSRs should be specified in this section. All non-power MSRs should have the capability to remove both fission and decay heat to ensure fuel system boundary integrity under all potential conditions. All reactors having forced-convection primary cooling systems should have specifications ensuring operability of systems and reactor configurations for fail-safe changeover from normal forced-convection to emergency forced-convection primary cooling or natural-convection primary cooling. An adequate heat sink, as described in the SAR, is a necessary component of such a system.

For reactors licensed to operate in both forced- and natural-convection primary cooling modes, the appropriate coolant system configurations and the relevant power levels for both modes should be specified as analyzed in the SAR.

Not all of the following items apply to all types of non-power MSRs. However, when applicable, they should be limited by technical specifications on the basis of the analyses and justifications in the SAR.

(1) Shutdown Cooling or Pump Requirements

At a minimum, the requirements for natural-convection primary cooling, including the interfacing heat dissipation system, and the operability and status of related systems required for shutdown should be specified as LCOs. If additional requirements are necessary for temporary forced-convection primary cooling following reactor shutdown from extended high-power operation, the technical specifications should state them, using the SAR as the basis.

(2) Isolation Valves

The existence, location, operability, and status of any valves required to isolate subsystems or components for operational needs, including removal of decay heat, should be specified as LCOs.

(3) Primary Cooling System Level Limits

Both the primary cooling system pressure and adequate natural convection flow depend on the available volume of non-fuel salt. Primary cooling system salt volume should be specified in the LCO for this reason, using the SAR as the basis.

(4) Detection of Leakage or Loss of Primary Cooling System Salt

If primary cooling system leakage or other loss of coolant could lead overheating of the fuel salt, fuel salt system boundary integrity failure, and an uncontrolled release of radioactive material (see items 5 and 8 below) to the environment, an LCO should state the need for operability of coolant leakage detection systems. An example of this type of system would be instrumentation that monitors the pressure difference between the fuel salt and primary cooling systems at the heat exchanger to detect conditions that would allow loss of primary cooling system salt in the event of a heat exchanger leak.

(5) Detection of Fission Product Activity in the Primary Cooling System

The technical specifications should provide for prompt detection of fission products escaping from the fuel system boundary into the primary cooling system. The method could be a radiation detector placed in the primary cooling system flow loop or a strategically located continuous air monitor in the reactor room or in a ventilation duct. Temporary substitutions, in case the fission product monitor is inoperable, should follow guidance in Section 3.7.1 of ANSI/ANS 15.1. This specification may be combined with the specification discussed in Section 3.7.1(2) on fission product monitors.

The specified radiation monitor should be able to initiate action, such as a reactor scram, reactor room isolation, or an alarm, as appropriate. The SAR should provide the bases and describe how the presence of fission products in the primary cooling system are distinguished from other airborne radioactivity.

(6) Primary Cooling System and Heat Dissipation System Radioactivity Limits

In addition to the prompt detection of fission products from failed fuel or experiment malfunctions [see Section 3.3(5)], LCOs should limit radioactivity in the primary cooling system. The technical specifications should require periodic sampling and appropriate analyses to detect and quantify radioactivity in both the primary cooling system salt and heat dissipation system coolant. The coolants should be sampled for gross activity at a short interval, for example, weekly, and sampled for isotope identification at a longer interval, for example, quarterly. The purpose of this LCO is to detect deterioration of components in the fuel salt loop, such as leakage in the fuel salt/primary cooling system salt heat exchanger. These specifications should be stated in such a way that significant changes in radioactivity, as defined in the SAR, trigger remedial action.

A table should specify primary cooling system and heat dissipation system activity channels and setpoints, the minimum number of channels, and the functions performed by the channel. The safety limits should be discussed in the basis for the table.

(7) Primary Cooling System Chemistry Requirements

To control corrosion of such components as the fuel salt/primary cooling system salt heat exchanger, there should be LCOs on primary cooling system chemistry. Parameters such as density, viscosity, oxygen level, salt activation products, and contaminants such as oil or metal oxide precipitants should have operational limits or ranges. There should be a definite schedule for sampling chemistry during both operating and shutdown periods. The bases should clearly address the appropriate ranges and give meaningful references. The SAR should justify the values for the chemistry requirements for the particular reactor.

3.4 CONTAINMENT OR CONFINEMENT

MSRs operate at near atmospheric pressure; therefore, accidents that result in building overpressure are unlikely. As a result, most non-power MSRs are housed in a confinement that is one barrier in a series of barriers comprising a functional containment and are not housed in a near leak-tight containment. There should be an LCO requiring that the system specifically described in the SAR exist as stated. The system should be operable during operation and for other applicable times such as before operation and following shutdown, as noted in Sections 3.4.1 and 3.4.2 of ANSI/ANS 15.1. If interlocks or administrative controls to ensure confinement operability are required, there should be appropriate specifications. Whether the facility has a confinement, or a containment depends on the reactor design, operating characteristics, and facility location. Specifications should require nominal exhaust rates for air under the operating and accident conditions analyzed in the SAR. Specifications should limit building leak rates to those described in the SAR.

3.5 VENTILATION SYSTEMS

Ventilation and exhaust flow rates and the systems to achieve the controlled release of effluents, as analyzed in the SAR, should be specified as LCOs. These LCOs should be established to achieve controlled release of effluents. Automatic failsafe closure of vents should be specified for confinement systems. Provisions to initiate controlled, filtered, and monitored exhaust and ventilation for radiological accidents should be included. In some cases, depending on the results of the analysis, minimum airflow rates may be LCOs.

The ventilation system should maintain a lower air pressure in the reactor room than in adjacent spaces. Air in the reactor room should not be distributed to other occupied spaces within buildings. The location and height of the air exhaust system stack or release point should be specified as an LCO here or as a design feature in Section 5 of the technical specifications. The dimensions of the stack should be consistent with the assumptions used in the SAR to predict potential radiation doses in the unrestricted environment. It is acceptable that the concentration of airborne radioactivity at the point of exhaust for normal operation be higher than the regulatory limit for restricted areas, provided that this point is not readily accessible to the public, the analyzed doses to the public are well below regulatory limits for unrestricted areas, and the potential doses to the facility staff are within regulatory limits. The as low as is reasonably achievable (ALARA) program should be applied in all analyses (see Section 3.7 below).

3.6 EMERGENCY POWER

Any requirement for emergency electrical power for non-power reactor facilities should be analyzed in the SAR on a case-by-case basis. Any necessary facility functions, such as radiation monitoring, emergency vessel cooling, or isolating the containment or the confinement, that need to be maintained if normal electrical power is lost should be described in the SAR. If emergency power is required, an LCO should ensure operability of the system. The technical specification should specify automatic startup of emergency electrical power if automatic startup is indicated in the SAR.

3.7 RADIATION MONITORING SYSTEMS AND EFFLUENTS

Monitoring systems and effluents may be addressed in the technical specifications under separate principal headings. The following discussion is consistent with the corresponding sections of ANSI/ANS 15.I.

3.7.1 Monitoring Systems

A separate table in the technical specifications (see Table 14.4) should list the required radiation monitors, the function each performs (e.g., scram or containment isolation), the approximate location of each, the type of radiation detected, and the alarm and/or automatic action setting, as analyzed in the SAR. The setpoints and calibrations should be listed in terms of radiation exposure rates and concentrations rather than as count rates that can change with calibration. Specific count rates for alarms and action settings can be presented in a facility procedure that can be amended in accord with the procedures section in the technical specifications. For specified monitors that become inoperable, the specification should state that reactor operations may continue only if the monitor is replaced by a substitute or portable monitor. The replacement monitor should perform essentially the same function until the original monitor is repaired or replaced (generally not to exceed 1 work week unless justified in the SAR). The specification also should state that if the specified monitor was displayed in the control room, the operator on duty should also be able to observe the temporary monitor. The applicant should provide a table applicable to the specific facility on the basis of the SAR.

(1) Air Monitors (Gas and Particulate)

Monitors should be specified for both radioactive gas and those radioactive particulates that might be airborne in the reactor room. There should be at least one continuous air monitor (CAM) with an audible alarm and data recorders. These monitors should be capable of alerting facility personnel to the presence of radioactivity. They should be calibrated for anticipated radioactive species. Potential sources of airborne radioactivity should be analyzed in the SAR. There should be specifications requiring operability of properly calibrated effluent monitors, preferably with recorded outputs for long-term records that provide documentation of the concentration and total quantity of radioactive effluents, as discussed in Section 3.7.2 below.

For reactors operated at power levels below a few hundred kilowatts, the concentrations of airborne radionuclides may be too low to measure during normal operation. For these, calculated concentrations of released quantities are acceptable as specifications, using the SAR as the basis.

Table 14.2. Typical required radiation measuring channels

Channel	Minimum Number Required	Function
Area radiation monitors	4	Alarm
Hot cell monitor	1	Alarm and door interlock
Fuel salt		Alarm
Primary cooling system salt	1	Alarm
Building particulate	1	Alarm (isolates containment with building particulate) and potential scram
Stack particulate	1	Alarm and potential scram
Stack gas	1	Alarm

(2) Fission Product Monitors

The specified fission product monitor could be the CAM or the fuel salt monitor, depending on the release scenarios analyzed in the SAR. Release of fission products from both fuel and fueled experiments should be included. This specification may be combined with the specification discussed in Section 3.3(5) above.

(3) Area Monitors

There should be a specification requiring operable area monitors in and near the reactor room. The type of radiation detected, such as gamma rays or neutrons, should be specified. Brand names, efficiencies, and specific designs should be avoided as specifications, but the range of exposure rates monitored may be specified. These area monitors should give information on the potential exposure rates from reactor-related radiation. Alarm and automatic action setpoints should be specified to ensure that personnel exposures and potential doses remain well below limits of 10 CFR 20 and are consistent with the facility ALARA program.

(4) Environmental Monitors

There should be at least one environmental monitoring station near the facility, preferably at the site boundary or at other areas of concern, such as at population centers or student dormitories. These monitors should be specified to match the types of radiation anticipated and should be either in the line of sight from the air exhaust point or downwind in the prevailing wind, as appropriate. The types of monitors should be specified [see Section 3.7.1(4) of ANSI/ANS 15.1]. The location and method of determining background readings should be discussed in the SAR and in the basis of the specification. The specification should state that environmental monitors are used to verify that the potential maximum dose, anal or other, in the unrestricted environment is within the values analyzed in the SAR. The specification should address both potential accident scenarios and normal operations.

3.7.2 Effluents

All radioactive species listed in Section 3.7.2 of ANSI/ANS 15.1 that are released by the facility should be addressed for normal operations, and the releases should be limited by technical specifications. NRC accepts the proposed concentration limits, provided the SAR shows that potential doses from these concentrations, comply with 10 CFR Part20 for the maximum exposed member of the public on a facility-specific basis. If the applicant proposes to limit release to 10 CFR Part20 limits at the point of release, then the analysis of effluents in the safety analysis report is sufficient and no technical specifications need be proposed.

3.8 EXPERIMENTS

Experimental facilities should be described in the SAR, and their basic features should be included in Section 5 (“Design Features”) of the technical specifications. The experiments to be performed in the experimental facilities need only be noted briefly, if at all, in the SAR, unless they could present a hazard to the reactor facility, the public, or facility staff. Any LCOs for experiments should be performance based to ensure that no regulations are violated, that experiment safety analysis limits are not exceeded, and that the reactor is not damaged by experiment failure or malfunction.

Regulatory Guide (RG) 2.2, provides detailed guidance to applicants on the scope of the discussions for experiments to be included in the SAR. The RG also provides guidance on the technical specifications needed to govern the experiments performed. The technical specifications should follow the guidance of Section 3.8 of ANSI/ANS 15.1 and Section C of RG 2.2, as supplemented by the guidance that follows.

3.8.1 Reactivity Limits

Limits should be specified on absolute values of reactivity associated with each type of experiment: secured, unsecured, and movable (see ANSI/ANS 15.1 and RG 2.2 for definitions). Generally, the limits on secured experiments should be approximately twice the limits on unsecured and movable experiments, where the latter should be no more than 1 dollar. The 1-dollar limit is such that inadvertent prompt criticality is avoided even if the experiment were to fail. Movable experiments must be clearly defined to include those to be inserted or removed while the reactor is operating. Unsecured experiments include those installed before reactor startup that change position or change other conditions while the reactor continues to operate. Reactivity limits of experiments that change position while the reactor is operating should not exceed the ability of the reactor operator or automatic servo system to maintain control of the reactor. The specified reactivity limits on movable experiments should not permit the violation of the shutdown margin specification.

The specified sum of the absolute values of the reactivity worths of all experiments should not be more than twice the limit on individual secured experiments. The value should be consistent with the SAR analysis of inadvertent reactivity insertions, as explained in Section C.1.a. of RG 2.2.

There should be a specification requiring that the reactor be shut down during the changing or moving of any secured experiment.

3.8.2 Materials

For fissile materials in experiments, limits should be specified on the allowed thermal power and on the equilibrium or maximum inventory of specific fission products, such as iodine and strontium. Specifications such as those indicated in Section C.2.a of RG 2.2 are acceptable.

A specification should require double encapsulation of potentially corrosive materials. All liquid and gas samples should be analyzed to determine if they require double encapsulation considering such factors as (1) the effect of failure on the reactor, (2) the radiological consequences of failure on the facility staff the public, and the environment, and (3) the possibility that the failure would result in an industrial hazard that could affect safe reactor operation. The failure of an encapsulation of material that could damage the reactor should require removal and physical inspection of potentially damaged components.

Specifications should limit the quantity of explosive material permitted in the experimental facilities and elsewhere in the reactor facility. For experimental facilities, the upper limit should be 25 mg TNT or its equivalent, as indicated in Section C.2.d of RG 2.2. For the overall reactor facility, the upper limit should be no higher than 100 mg TNT or its equivalent, unless a larger quantity is analyzed in the SAR and approved by NRC. An additional specification should require prior testing or analyses of explosive material encapsulations to ensure no reactor damage in the event of detonation, regardless of the limit.
A specification should limit the quantities of unknown materials that could be placed in certain experimental facilities for exploratory studies. Conformance with Section C.2.i of RG 2.2 would be acceptable.

3.8.3 Failure and Malfunctions

Specifications that address the failure and malfunction of an experiment and limit the experiment parameters should be included on a case-by-case basis, as discussed in the SAR. The guidance of

Section 3.8.3(2) of ANSI/ANS 15.1 should be followed, but specifications that require compliance with regulations are redundant and are unnecessary [see Section 3.8.3(1) of ANSI/ANS 15.1].

For experiments that may off-gas, sublime, volatilize, or produce aerosols, standard assumptions are often specified for calculating the activity that could be released under normal operating conditions, accident conditions in the reactor, and accident conditions in the experiment. Such specifications ensure conservatism in the safety analysis of the experiment. These specifications have contained such assumptions as the following: (1) if an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100 percent of the radioactive gases or aerosols escape; (2) if an effluent holdup tank isolates on a high radiation signal, at least 10 percent of the radioactive gases or aerosols escape; (3) if the effluent exhausts through a filter with 99-percent efficiency for 0.3-micron particles, at least 10 percent of the vapors escape; and (4) if an experiment fails that contains materials with a boiling point above 130 'F (54 C), the vapors of at least 10 percent of the materials escape. Any particular assumptions used should be derived from the SAR. Applicable limits for specific experiments are normally not part of the technical specifications and should be derived from the experiment safety review discussed in Section 6.5 below.

3.9 FACILITY-SPECIFIC LCOS

The LCOs discussed above apply to most non-power reactors. Each reactor may also have technical specifications containing facility unique LCOs. These should be based on the SAR and facility design.

For an MSR, facility unique systems could include fuel receipt, handling, mixing, storage, bulk batching and addition to the vessel, daily batching and addition to the vessel, and sample storage. Other unique SSC could include remote handling systems, aspects of the gas management system, and other support tank systems that could contain fissile materials not specified in other LCOs in Section 3 of the technical specifications.

4. SURVEILLANCE REQUIREMENTS

Certain LCOs established in Section 3 of the technical specifications should be accompanied by a surveillance requirement in Section 4. These surveillance-related specifications should clearly identify the parameter or function to be measured or tested, the method, the frequency, and the acceptable deviation or error. Acceptable deviations could be limited by license conditions (such as thermal power level) or by regulations (such as 10 CFR Part20).

NRC accepts the surveillance frequencies stated in this section of ANSI/ANS 15.1 as amplified in the following sections. The actual wording of the specifications should not be ambiguous. Wording in ANSI/ANS 15.1 has been interpreted incorrectly by some licensees to allow the extended interval (interval not to exceed statement) as the average. If the extended interval is used for a particular surveillance test, a shorter interval should be used as soon afterwards as possible to adhere to the average. The intervals of ANSI/ANS 15.1 should be listed in the applicant's technical specifications.
In addition to surveillance verification of LCOs, other surveillance activities should be specified. These include such specifications as thermal power level calibration, preventive maintenance and inspection of control/safety element drive systems, fuel system boundary inspections, preventive maintenance on other important components to provide assurance of operability, and calibration of effluent monitoring systems.

If a surveillance is not required for safety while the reactor is shut down, it may be deferred, but must be performed before reactor startup. If the reactor is not to be operated in a particular mode for an interval that exceeds the surveillance intervals for that particular mode, surveillances not required for safety while the reactor is operated in other modes may be deferred, but must be performed before the reactor is considered operational in the mode in which surveillances were deferred. Scheduled surveillances that cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown. Surveillances that may be deferred and the reasons for deferment should be clearly stated in the technical specifications, justified in the SAR, and noted in the basis of the specification.

In general, any time that a reactor system or component is modified or repaired, the surveillance for that system should be performed as part of the operability check of the system or component. This should be done regardless of when the surveillance was last performed or when it is next due. This special surveillance may change the due date of the next regularly scheduled surveillance of that type.

4.1 FUEL SALT AND FUEL SYSTEM BOUNDARY PARAMETERS

The excess reactivity and shutdown margin LCOs specified in Section 3 of the technical specifications are applicable for all authorized operating conditions. As an example, for a movable experiment, the specifications for excess reactivity and shutdown margin surveillance measurements should be based on that experiment being in its most reactive location. In addition, other reactor parameters that affect reactivity during operation should be explicitly specified. For the following specified surveillance requirements, the parameters may be determined by an appropriate combination of measurements and calculations.

(1) Excess Reactivity

Excess reactivity should be determined after changes in either the active reactor core geometry, core configuration as defined in Section 1.3, other active reactor core materials, or in-core experiments for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

(2) Shutdown Margin

The shutdown margin should be determined at least annually and after changes in either the active reactor core geometry, core configuration, other active reactor core materials, or in-core experiments.

(3) Active Reactor Core Configuration

Limitations on active reactor core configurations are intended to ensure that reactor physics and thermal-hydraulic parameters specific to the active reactor core are within the limits analyzed in the SAR. Active reactor core configuration parameters specified in Section 3.1(4) or in Section 5 of the technical specifications should be met during reactor operations. Therefore, an acceptable surveillance specification is to verify compliance with all applicable specifications in those sections when any change occurs in the active reactor core configuration.

(4) Fuel Composition Changes

Fuel salt chemistry parameters should be evaluated periodically to ensure values are consistent with maintaining fuel system boundary integrity. The surveillance interval should be justified in the SAR.

(5) Reactivity Coefficients

Section 3.1(5) of the technical specifications limits reactivity coefficients, which are largely determined by reactor design and fuel type. Measuring and verifying reactivity coefficients can be a difficult task. An acceptable schedule for surveillance of reactivity coefficients is at initial reactor startup and when any change in the active reactor core configuration or fuel type requires changes in the specifications of Section 5.

(6) Hydrogen Concentration in Gas Management System

If applicable, the gas management system should be tested for isotopes of hydrogen (hydrogen, deuterium, and tritium). An acceptable schedule for surveillance of radiation detectors should be established from engineering judgment and similar component inservice inspection requirements and operating experience.

(7) Detection of Fission Product Activity

Section 3.1(8) of the technical specifications requires prompt detection of fission products escaping from the fuel system boundary, the gas management system, the fuel handling system, or the fuel cleanup system. Detection systems specified in Section 3.1(8) or in Section 5 of the technical specifications should be met during reactor operations. An acceptable schedule for surveillance of radiation detectors should be established from engineering judgment and similar component inservice inspection requirements and operating experience.

(8) Emergency Cooling Systems

NRC accepts the guidance provided in this section of ANSI/ANS 15.1 for the test of emergency cooling sources and systems.

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

(1) Reactivity Worth of Control Elements

The integral and differential worths of all control elements should be determined at initial fuel loading. Integral and differential worths should be determined at least annually and after changes of the active reactor core geometry, core configuration, other active reactor core materials, or in-core experiments, as noted in Section 4.1(1) above.

(2) Control Element Manipulation Speeds

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(3) Scram Times of Control Elements

A specific interval should be stated for the surveillance intervals given in Section 4.2(4) of ANSI/ANS 15.1.

(4) Scram and Power Measuring Channels

Channel tests of all scram and power measuring channels required by technical specifications, including scram actions with control element release, fuel salt pump operation, and interlocks, should be performed before each reactor startup following a shutdown of more than 24 hours or following each secured shutdown. If the reactor operating schedule calls for no secured shutdowns, the channel tests should be performed at least quarterly. Many facilities perform these tests before each reactor startup and NRC recommends this practice.

(5) Operability Tests

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

(6) Thermal Power Calibration for Reactors Cooled by Forced Convection

Thermal power should be calibrated at least annually; heat balance should be verified at least monthly.

(7) Thermal Power Calibration for Reactors Not Cooled by Forced Convection

Thermal power should be calibrated at least annually. The basis should indicate the method to be used.

(8) Element Inspection

The operability of all control elements should be inspected annually. The action of the elements to displace fuel, absorb neutrons, reflect neutrons, spectrally adjust neutrons, or a combination of these methods should be inspected biennially for indications of deterioration or damage. This can be a visual inspection or an inspection that requires the control elements to pass through a measuring device which detects deterioration.

4.3 PRIMARY COOLING AND HEAT DISSIPATION SYSTEMS

Not all of the following items apply to all types of non-power MSRs. However, when applicable, they should be limited by technical specifications on the basis of the analyses and justifications in the SAR. The applicant should discuss the applicability and identify the parameters that should be tested. The applicable parameters should be specified, and the functions should be explicitly stated in the specification.

(1) Inservice Inspections

If any inservice inspections of primary cooling system components are identified in the SAR, they should be performed according to the manufacturers recommendations. If the manufacturer’s recommendation is not available, the frequency should be as established in the SAR from engineering judgment and similar component inservice inspection requirements and operating experience.

(2) Analysis of Coolants for Radioactivity

Analyses for isotope identification of the primary cooling system and, if applicable, the heat dissipation system should be performed by sampling at an appropriate frequency consistent with the analysis in the SAR. Sampling weekly for gross analysis should be considered to establish trends to quickly identify heat exchanger failures.

(3) Primary Cooling System Chemistry

When the reactor is operating on a routine schedule, primary cooling system chemistry should be measured at an appropriate frequency consistent with the analysis in the SAR. This requirement could be met by a system that monitors chemistry parameters continuously while the reactor is operating.

If the reactor is not operated for long periods, the interval between primary cooling system chemistry measurements may be increased if reasonable justification is provided in the SAR.

(4) Primary Cooling System Sensors and Channels

Channel tests of sensor operability and channels not included elsewhere in the technical specifications that are identified in the SAR should be performed quarterly and before startup after maintenance. All channels should be calibrated annually and before startup after major modification or component replacement.

4.4 CONTAINMENT OR CONFINEMENT

4.4.1 Containment

Few licensed non-power reactors are required to have a containment. For those required by the SAR, the surveillance intervals given in ANSI/ANS 15.1 are acceptable.

4.4.2 Confinement

MSRs operate at near atmospheric pressure; therefore, accidents that result in building overpressure are unlikely. As a result, most non-power MSRs are housed in a confinement that is one barrier in a series of barriers comprising a functional containment and are not housed in a near leak-tight containment. Confinement is a system that provides a temporary holdup or controlled release of radioactive effluents to the environment. Most non-power reactors are equipped with confinements and should have a functional test of the overall system described in the SAR quarterly. In addition, an efficiency test of the filters should be performed annually or in accordance with manufacturer recommendations and acceptance criteria.

4.5 VENTILATION SYSTEMS

Ventilation systems at most licensed non-power reactors are an integral part of the containment or confinement system, and surveillance activities may be interrelated. An operability check, including dampers and blowers, should be performed quarterly and following repair or maintenance to declare the system operable. The function and efficiency of filters should be tested annually or in accordance with manufacturer’s recommendations and acceptance criteria and following repair or maintenance to declare the system operable.

4.6 EMERGENCY ELECTRICAL POWER SYSTEMS

For all safety-related emergency electrical power systems, channel checks or other operability checks should be performed before reactor startup and after maintenance. Maintenance should be performed according to the manufacturer’s recommendations. If the manufacturer's recommendations are not available, the frequency should be as determined in the SAR.

4.6.1 Diesels and Other Devices

The shorter of the surveillance intervals given in this section of ANSI/ANS 15.1 is acceptable.

4.6.2 Emergency Batteries

The shorter of the surveillance intervals given in this section of ANSI/ANS 15.1 is acceptable.

4.7 RADIATION MONITORING SYSTEMS AND EFFLUENTS

4.7.1 Monitoring Systems

A channel check should be performed daily before reactor startup. Where physically possible, a channel test using a radiation source should be performed at least monthly. The SAR should describe such capability.

All required radiation monitoring systems, including effluent monitors, should be calibrated at least annually according to the manufacturer's recommendations. Individual systems should have separate specifications.

4.7.2 Effluents

Quantities of radioactive effluents released to the environment are LCOs. If the SAR states that it is not feasible to monitor such effluents from low power reactors in real time at the point of release, calculated releases may be substituted. The SAR should specify surveillance methods and intervals for confirming these releases or for verifying upper limits.

For gaseous airborne radioactive effluent, it is acceptable to confirm annual upper limits by integrating dosimeters such as thermoluminescence dosimeters (TLDs) or film.

For particulate airborne or waterborne radioactive effluent, it is acceptable to confirm annual upper limits by surveillance of environmental factors given in this section of ANSI/ANS 15.1.

4.8 EXPERIMENTS

If any experiment discussed in the SAR is designed to operate with emergency systems or with connections to the reactor protective systems, a channel check should be specified both daily and before reactor startup when the particular experiment is being performed. Surveillance activities for experiments that are included in the experiment protocol and the review and approval process need not be included explicitly in the technical specifications.

4.9 FACILITY-SPECIFIC SURVEILLANCE

There should be applicable surveillance specifications for any facility-specific LCOs in Section 3.9 of the technical specifications not explicitly included in Section 4. These surveillances should be performed to verify significant safety features from the SAR

5. DESIGN FEATURES

The SAR forms the basis for NRC to issue an operating license for a non-power reactor. Essential information includes the type and enrichment of fuel, active reactor core and fuel salt configurations, fuel salt storage facilities, thermal power level, potential accident scenarios and mitigating features, environmental conditions at the site, and other factors. To ensure that the issued license remains valid, design features should not be changed without prior NRC review and approval. These major design features are noted in Section 5 of the technical specifications, if they have not already been specified in Section 2 or 3. The NRC accepts the guidance in this section of ANSI/AS 15.1. The applicant should provide concise but explicit information on all noted features.

6. ADMINISTRATIVE CONTROLS

The specified information and controls on staffing and operations of the reactor facility will ensure that the management and staff of the facility are acceptably knowledgeable and aware of the technical requirements to operate a safe facility, to comply with regulations and the license conditions, and to practice a meaningful ALARA program, which will protect the environment and the health and safety of the public, the facility users, and the staff.

Not all owners and operators of non-power reactors will have the same management organization or office titles. Regardless of the details of the management organization, or of the complexity of the facility, the administrative functions presented in this section of ANSI/ANS 15.1 should be established and specified. The NRC accepts the ANSI/ANS 15 1 position as modified in the sections that follow.

6.1 ORGANIZATION

6.1.1 Structure

The information recommended by ANSI/ANS 15.1 should be clearly stated, including how and when the radiation safety staff communicates with the facility manager and level 1 management to resolve safety issues.

6.1.2 Responsibility

NRC accepts the guidance provided in this section of ANSI/ANS 15 1.

6.1.3 Staffing

NRC accepts the guidance provided in this section of ANSI/ANS 15 1.

6.1.4 Selection and Training of Personnel

Compliance with 10 CFR Part55 is required of the licensee and licensed operators, unless NRC has issued an exemption. ANSI/ANS 15.4 provides additional guidance for non-power reactors.

6.2 REVIEW AND AUDIT

The committee established to perform the review function may be assigned approval authority by the facility manager or the facility manager may retain that authority. Section 6.2 of the technical specifications should explicitly state who holds the approval authority and should specify the committee's authority and how it communicates and interacts with management levels 1 and 2.

6.2.1 Composition and Qualifications

One or more voting members of the committee should be from organizations other than the one operating the reactor.

6.2.2 Charter and Rules

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

6.2.3 Review Function

The fact that this section of ANSI/ANS 15.1 addresses the review function required by 10 CFR 50.59 should be explicitly stated in the technical specifications.

6.2.4 Audit Function

In addition to the emergency plan, all other required plans, such as physical security and operator requalification, should be specified for auditing. The requirement to audit these plans may be part of the plan itself. If that is the case, the requirement to audit does not need to be repeated in the technical specifications.

6.3 RADIATION SAFETY

The technical specifications should state that 10 CFR Part20 establishes requirements that the radiation safety program must achieve. Additional guidance for radiation safety programs at non-power reactors may be found in ANS/ANS 15.11.

The authority of the radiation safety staff to interdict or terminate safety-related activities should be stated. The technical specifications should state management's commitment to practice an effective ALARA program. This program should apply to facility staff, facility users, the general public, and the environment.

6.4 PROCEDURES

Procedures in addition to those in ANSI/ANS 15.1 should be proposed at facilities to address operational situations recognized in the SAR. For example, if byproduct material whose possession is authorized under the reactor license is used in facility laboratories that are part of the reactor license and/or transferred to other licensees, procedures for control and transfer of this byproduct material should be part of the set of minimum procedures required by the technical specifications. The specifications should be written to ensure a minimum necessary set of procedures but should allow for future additions as necessary.

The minor modifications and temporary deviations allowed by ANSI/ANS 15.1 should not be spelled out in the technical specifications. However, the methodology for establishing and changing procedures should be stated in the specifications.

6.5 EXPERIMENTS REVIEW AND APPROVAL

In addition to the guidance of ANSI/ANS 15.1, the review and approval of experiments should be consistent with the guidance provided in Section C.3 of RG 2.2 and RG 2.4. The specifications should make clear that "established and approved procedures" means written procedures, properly reviewed and approved. Any changes made to these procedures should conform to Section 6.4 of ANSI/ANS 15.1.

6.6 REQUIRED ACTIONS

6.6.1 Action to be Taken in Case of Safety Limit Validation

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

6.6.2 Action to be Taken in the Event of an Occurrence of the Type Identified in Sections 6.7.2(1)(b) and 6.7.2(1)(c)

The first sentence of this section of ANSI/ANS 15.1 states that “reactor conditions shall be returned to normal or the reactor shall be shut down.” The specification should be written to provide that the applicant establish, in advance, specific criteria for the two alternative actions: shutdown or return to normal. For example, a return-to-normal event is a reactor scram resulting from a known cause, such as an electric transient.

6.7 REPORTS

6.7.1 Operating Reports

The technical specifications should state that operating reports should be sent to the NRC Document Control Desk and that a copy should be sent to the appropriate regional administrator.

Section 6.7.1(4) of ANSI/ANS 15.1 refers to the reporting required by 10 CFR 50.59. The specification should make reference to the rule.

6.7.2 Special Reports

The technical specifications should state that (1) special written reports of events should be sent to the NRC Document Control Desk and that a copy should be sent to the appropriate regional administrator and (2) special telephone reports of events should be made to the NRC Operations Center and the regional staff.

6.8 RECORDS

NRC accepts the guidance provided in this section of ANSI/ANS 15.1.

7. REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, “The Development of Technical Specifications for Research Reactors,” ANS, LaGrange Park, Illinois, 2007 (R2013).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, “Selection and Training of Personnel for Research Reactors.” ANS, LaGrange Park, Illinois, 2016.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, “Radiological Protection at Research Reactor Facilities,” ANS, LaGrange Park, Illinois, 2016.

Stahl, D., *Fuels for Research and Test Reactors, Status Review,* ANL-85-5, Argonne National Laboratory, Argonne, Illinois, December 1982.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” April 2018.

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, NUREG-0851, "Nomograms for Evaluation of Doses from Finite Noble Gas Clouds," January 1983.

16 OTHER LICENSE CONSIDERATIONS

Chapter 16 of this guide is applicable to providing a description of other license considerations for a non-power MSR. In this chapter of the SAR, the applicant should discuss license considerations that do not belong elsewhere in the SAR. One of these considerations is the way reactor components were used in the past. A more recent consideration discussed in this chapter is the medical use of non-power reactors. Other topics that should appear in this chapter should be determined by the applicant on a case-by-case basis and added as separate sections.

16.1 PRIOR USE OF REACTOR COMPONENTS

This section applies primarily to applications for license renewal, since the facility has a history of operation. However, new facilities at which components are being used that came from other reactor facilities may have to consider prior use of components. Fuel provided by the Department of Energy (DOE) for a new facility or for a facility already in operation could come from DOE storage and have a history of prior use that must be considered. Prior use need not have taken place at the reactor for which the applicant is seeking a license, but the applicant should consider how the component was used in the past.

In SAR Chapter 13, "Accident Analyses," the applicant discusses various accident events considered for the facility and the components or systems that prevent or limit the release of radioactive material to the facility confinement or containment and to the environment. The applicant should consider whether prior use of components or systems could significantly degrade their capability to continue to perform their safety functions.

Some components and systems for which prior use should be considered are fuel salt, the fuel system boundary, reactivity control systems, and engineered safety features. This list is not all inclusive, and the applicant should review the design of the facility to ensure that all prior use of important components and systems has been considered.

For components or systems that are identified for analysis of prior use, the applicant should discuss the possible mechanisms that could lead to deterioration, along with the potential effect of the mechanism on the component or system. Some prominent deterioration mechanisms are the following:

- Radiation
- High temperature or temperature cycling
- Corrosion
- Erosion
- Mechanical damage

The applicant may show by analysis that the deterioration mechanisms are not sufficient to cause damage that would interfere with the performance of safety functions. Components should be demonstrated to be within design-life requirements. For example, manufacturer design specifications, recommendations, and requirements could be used to establish acceptable performance standards for use. Analysis could show that the stress in a component caused by temperature cycling is not sufficient to damage the component. As part of the analysis process, the applicant may consider the performance of similar components or systems in other non-power reactors that have longer operating histories. Component attributes or system performance may be measured to show that deterioration mechanisms have not damaged a component or system to a point that impairs its safety function. Two examples are the ultrasound measurement of pipe thickness or the measurement of control element actuation and response time. In some cases, periodic measurements may be needed to determine trends in component or system deterioration to ensure acceptable performance. The timing of periodic measurements and component or system performance limits may form the basis of limiting conditions for operation in the technical specifications.

The applicant should describe the regular preventive and corrective maintenance program that provides for replacement or repair of important components or systems as necessary. The description should include a discussion of how components are chosen for maintenance, how maintenance intervals are chosen for some key components, and if the manufacturer's suggestions are heeded in the maintenance program. The success of the program in preventing malfunctions and other failures of equipment should be discussed. If components and systems in the maintenance program have malfunctioned or otherwise failed, the applicant should show that these problems do not indicate that the program has broken down.

APPENDIX B. PART 2, *GUIDELINES FOR PREPARING AND REVIEWING APPLICATIONS FOR THE LICENSING OF NON-POWER MSRS: STANDARD REVIEW PLAN*

ABSTRACT

Part 2 of this guide provides guidance on the conduct of licensing action reviews to NRC staff who review non-power liquid-fueled molten salt reactor (MSR) licensing applications. These licensing actions include construction permits and initial operating licenses, license renewals, amendments, decommissioning, and license termination.

INTRODUCTION

BACKGROUND

This document gives guidance to the staff of the U.S. Nuclear Regulatory Commission (NRC) and reviewers under contract to the NRC for performing safety reviews of applications to construct, modify, or operate a non-power liquid-fueled molten salt reactor (MSR)[3]. MSRs are a class of reactors in which a molten salt performs a significant function in the active reactor core. 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 licensing an MSR. This document also makes information about regulatory matters widely available and helps interested members of the public and the MSR community better understand the review process. A companion document, Part 1 of this guide (Format and Content), provides guidance to assist applicants in preparing and submitting high-quality licensing applications for nonpower MSRs.

Unless otherwise noted, the terms *MSR* and *reactor* as used in this guidance are understood to mean a liquid fueled non-power MSR. The use of terminology in this guidance is consistent with the following:

For the purpose of this guide, *non-power production or utilization facility* (NPUF) means a production facility or a utilization facility, licensed under 10 CFR 50.21(a), 50.21(c), or 50.22, as applicable, that is not a nuclear power reactor or a production facility as defined under paragraphs (1) and (2) of the definition of *production facility* in 10 CFR 50.2.

Use of the term *NPUF* in this guide means that the guidance is applicable to both utilization facilities (e.g., MSRs) and production facilities (such as fuel cleanup systems or other facilities used for the processing of irradiated materials containing special nuclear material). If the guidance is applicable to only MSRs or only production facilities, then the term *NPUF* is not used, and instead the term *MSR* or *production facility* (or *fuel cleanup facility*, if appropriate) is used.

The term *MSR NPUF* is used when the intention is to provide guidance that is applicable to both production facilities and utilization facilities that use molten salt technology. MSR NPUF is not used to refer to a single facility, and instead MSR, production facility, or facility is used, as appropriate. If a single site contains both a production facility and a utilization facility that are interconnected or collocated in a single building, then the term *facility* could be used to refer to them as a whole.

Non-power reactor means:

(1) A testing facility; or
(2) A research reactor, which is an NPUF that is a nuclear reactor licensed under 10 CFR 50.21(c) and is not a testing facility, or
(3) A commercial or industrial non-power reactor, which is an NPUF that is a nuclear reactor licensed under 10 CFR 50.22 and is not a testing facility.

Therefore, this guide uses the terms *non-power MSR* or *reactor* to mean that the guidance is applicable to all types of non-power reactors (i.e., testing facility, research reactor, or commercial or industrial reactor) that use MSR technology. The term *MSR* should be considered synonymous with the term *non-power MSR* because this guidance document is specifically directed at non-power facilities. If the guidance intends to point out a distinction between non-power reactors using MSR technology and non-power

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[3] There are also salt-cooled reactor designs that propose using fixed-position, coated-particle ceramic fuel. The discussion in this guide is focused on MSRs operating with liquid fuel.

reactors that use other technologies, then the term *non-power MSR* will be used specifically to highlight the distinction. The term *testing facility* will be used when the guidance is applicable to only testing facilities and the term *research reactor* will be used when the guidance is applicable to only research reactors.

The MSR guidance provided here is based on NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." However, unlike MSRs, NUREG-1537 focuses on reactors operating with heterogeneous fuel elements. Part 2 of this guide is intended to provide license application reviewers a definitive base from which to evaluate an application for a non-power MSR operating with homogenous fuel. The guidance herein is based on the Code of Federal Regulations, Title 10, Section 50.34 (10 CFR 50.34) which describes the information to be supplied in a SAR.

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 MSR 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* of 1954, as amended (AEA or Act). The AEA was written to promote the development and use of atomic energy for peaceful purposes and to control and limit its radiological hazards to the public. These purposes are expressed in paragraph 104 of the Act, 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 nonpower reactors is proportionally less and requires less stringent and less prescriptive measures to give equivalent protection to the health and safety of the public. In MSRs, fission products are generated and entrained in the fuel salt because there is no fuel cladding. Gaseous fission products, such as xenon and krypton bubble off continuously and are typically removed from the cover gas space through a gas management system without any significant impact on reactor operation. Soluble and insoluble fission products remain in the fuel salt. Insoluble fission products tend to plate out on reactor surfaces, while soluble fission products are typically removed from the fuel salt by chemical processing, polishing, or filtration. 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 MSRs apply to both test and research reactors. The guidance herein for reviewing submittals and the criteria for acceptability should be followed for MSR non-power reactors. Differences for test reactors will be discussed in the applicable chapters.

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 are generally compared with the 10 CFR 20.1001 through 20.2402 and Appendices. Occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301.

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 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.

Part 2 of this guide 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 may be applicable to every non-power MSR 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. 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 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 section for every application. Rationale for each decision should be documented in the appropriate section of the SAR.

The regulations in 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” specify environmental reviews that shall be completed as part of various licensing actions for NPUFs. Guidance for applicants and the NRC staff related to these regulations is provided in “Interim Staff Guidance Augmenting NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors,” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12156A069) and “Interim Staff Guidance Augmenting NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors,” (ADAMS Accession

No. ML12156A075). Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance for environmental reviews associated with MSR NPUF licensing.

Guidance for applicants and the NRC staff related to financial qualifications is provided in NUREG-1537 Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content” (ADAMS Accession No. ML042430055), NUREG-1537 Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria” (ADAMS Accession No. ML042430048), “Interim Staff Guidance Augmenting NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors,” and “Interim Staff Guidance Augmenting NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors.” Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance on NRC regulations related to financial qualifications that must be met as part of MSR NPUF licensing.

Guidance for applicants and the NRC staff related to decommissioning is provided in NUREG-1537 Part 1, NUREG-1537 Part 2, “Interim Staff Guidance Augmenting NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors,” and “Interim Staff Guidance Augmenting NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors”. Although these documents were not written specifically for MSR NPUFs, they are general in nature and provide an adequate level of guidance on NRC regulations related to decommissioning that must be met as part of MSR NPUF licensing.

DOCUMENT STRUCTURE

Parts 1 and 2 of this document are complementary, titles and numbers of sections correspond to the SAR sections. Part 2 of 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.
- *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:

    a. stating the applicable requirements or standards, with specific citation to the source of those requirements of standards

b. summarizing the applicant's proposed method for satisfying the requirements or standards
c. summarizing the staff's analysis of whether the applicant's proposal does indeed satisfy the requirements or standards

The documented analysis must provide 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 guide, 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 guide.

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 guide, and which may also list other useful material.

This guide and the format and content guide were developed for all designs and generally apply to non-power MSRs 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.

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 MSR applying for an initial license 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 a release of fuel salt, a release of gaseous fission products from the reactor vessel or inter-connected systems, 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 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.

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 active reactor core configuration changes, and other significant changes to a non-power reactor 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.

REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21, "Format and Content for Safety Analysis Reports for Research Reactors," 2012 (R2018).

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

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

U.S. Nuclear Regulatory Commission, Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 2012.

U.S. Nuclear Regulatory Commission, Interim Staff Guidance Augmenting NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 2012.

U.S. Nuclear Regulatory Commission, "Guidance for Electronic Submissions to the NRC," Revision 8, May 2017.

MSR DEFINITIONS

The following glossary contains terms often used when discussing an MSR.
Where terminology in this guidance does not properly characterize new technology, vendors and applicants should introduce appropriate substitute terminology and provide definitions.

Active Reactor Core: In an MSR, the vessel region occupied by the fuel salt, where the majority of prompt neutrons are generated and where most fissions occur. In an MSR, the active reactor core geometry might change with time as a result of changes in density and voiding of the solution.

Coating or Cladding: Intervening protective layer of material between the fuel salt and the structural container alloy. Also included are surface modifications of the structural alloy to enhance its chemical or mechanical performance by altering its microstructure or composition (e.g., carbiding, phosphiding, or nitriding the surface).

Control Element(s): Object(s) employed to adjust reactivity. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned.

Emergency Cooling System: A system that provides decay heat removal from the reactor fuel following an accident (e.g., a direct reactor auxiliary cooling system or a reactor vessel auxiliary cooling system). Similar cooling systems for fuel drain tanks are also included, as are systems such as in-floor heat pipes, to provide cooling to fuel located in outer containment layers in the event of fuel system breach type accidents.

Fuel Barrier: Portion of the fuel system boundary in contact with the liquid fuel after addition to the fuel circuit and prior to transfer to waste handling (principally the vessel, chemical processing system boundary, drain tank [if used], heat exchanger, cooling thimbles, control element thimbles, instrumentation thimbles, piping, tanks, and valves).

Fuel System Boundary: The material that mitigates the release of radionuclides from the reactor fuel, including volatile fission products (e.g., krypton, xenon, iodine). For an MSR, this includes the vessel, drain tank (if used), cooling thimbles, heat exchangers, chemical processing system boundary, waste-handling tank, pumps, valves, and piping—it essentially includes the radionuclide barrier and the fuel barrier.

Gas Management System: The cover gas system provided to capture volatile fission products (e.g., krypton, xenon, iodine) until ultimate discharge and to provide venting of any pressure/density transients that could result in damage to the vessel or the fuel salt/ primary cooling system salt heat exchanger, thus resulting in loss of the fuel system boundary.

Heat Dissipation System: A set of components or system(s) that interface with the primary cooling system to provide the principal means of transferring the heat from the active reactor core to an ultimate heat sink. The heat dissipation system might use a variety of coolants (e.g., salt, liquid metal, gas, or water) but does not contain fuel.

Neutron Moderator: In an MSR, materials in or near the active reactor core that consist of light elements (e.g., H, B, C). Moderators are generally solid form.

Primary Cooling System: The system that directly interfaces with the fuel system boundary at the fuel salt/primary cooling system salt heat exchanger(s) to provide the principal means of removing heat from the fuel salt during operation by transferring the heat to the heat dissipation system. The primary cooling system may employ a variety of coolants (e.g., nitrate salt, halide salt, etc.), but it does not contain fuel.

Radionuclide Barrier: The portion of the fuel system boundary that serves as the innermost low-leakage barrier to the radionuclides within the gas management system.

Reactor Fuel: In an MSR, fuel salt consisting of fissionable and possibly fertile halide salts, fission products, and generally solvent halide salt(s).

Vessel: For an MSR, the structure containing the active reactor core. In certain design configurations, other components such as heat exchangers might reside in the vessel but outside the active reactor core.

1 THE FACILITY

Chapter 1 of this guide is applicable to reviewing a description of the facility for the licensing of a non-power MSR. This chapter of the 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

This section should provide an overview of the facility being licensed.

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 facility;
- geographical location;
- reactor type, salt description, and power level of the reactor;
- inherent or passive safety features; and
- unique design features that are 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 of 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. 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 MSR facility.

Areas of review should include the following:

- potential radiological consequences of operation and the method of providing protection;
- safety criteria proposed by the applicant;
- principal safety considerations of the facility design;
- description of safety of unique design features; and
- 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 MSR 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 system boundary, 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 the fuel system boundary. 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)* 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 MSR 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 geographic location of the facility and principal characteristics of the site;
- the basic design features, operating characteristics, and safety systems of the MSR and its instrumentation and control and electrical systems;
- engineered safety features designed to control radiation releases;
- the thermal power level of the reactor and the system that removes and disperses the power;
- the design features of the radioactive waste management system or provisions and radiation protection; and
- the basic experimental features and capabilities in the design.

Acceptance Criteria

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

- The applicant should briefly describe:

  a. geographical location of the MSR facility;
  b. principal characteristics of the site;
  c. principal design criteria, operating characteristics, and safety systems;
  d. fuel system boundary;
  e. any engineered safety features;
  f. instrumentation, control, and electrical systems;
  g. other auxiliary systems;
  h. radioactive waste management provisions or system and radiation protection; and
  i. 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 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 MSR 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 MSR facility should be designed to accommodate all uses or malfunctions of the shared facilities without degradation of the non-power MSR safety features.
- The non-power MSR 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 MSR 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' 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 MSR, nor would it damage the non-power MSR 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); and
- 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, "Molten Salt Reactor Description," and Chapter 13 for the bases of NRC's acceptance of fuel and fuel system boundary performance [e.g., the Oak Ridge Research Reactor (ORRR), the system for nuclear auxiliary power (SNAP), the MTR, the homogeneous reactor test (HTR) at Oak Ridge National Laboratory (ORNL), the aircraft reactor experiment (ARE) at ORNL, the molten salt reactor experiment (MSRE) at ORNL, and the advanced test reactor (ATR) at Idaho National Laboratory]
- Chapters 4 and 13 for the bases for active reactor core critical size and geometry [e.g., BSF, CP-5, the Argonne nuclear assembly for CP-1 I (Argonaut), and MSRE]
- Chapter 6, "Engineered Safety Features," for the bases of accident mitigation systems
- Chapter 7, "Instrumentation and Control Systems," for the bases of redundancy and diversity in instruments and controls, including scram (reactor shutdown) systems [e.g., MSRE, BSF, 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:

- As experience with MSR facilities grows, 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 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.

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, as amended, 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:

- Section 302(b)(1)(B) of the *Nuclear Waste Policy Act* of 1982, as amended, 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 contact 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, as amended, have been satisfied.

2 SITE CHARACTERISTICS

Chapter 2 of this guide is applicable to reviewing a description of the facility for the licensing of a non-power MSR. 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. The regulations in 10 CFR 100.10 specify factors to consider in selecting sites for testing facilities, while there are no regulations that specify factors to consider for siting research reactors. However, IAEA-TECDOC-403, "Siting of Research Reactors," provides guidance for siting research 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 distribution 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 population density up to 5 miles from the reactor should be given. The current and projected population distributions within 0.5, 1, 2, 3, and 5 miles 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 MSR. 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 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:

    a. the maximum snow and ice load that the roofs of safety-related structures must be capable of withstanding during reactor operation
    b. the maximum wind speed that safety-related structures must be capable of withstanding during reactor operation

c. 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 5 miles.
- 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 fuel salt or primary cooling system salt 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 fuel salt or primary cooling system salt, 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 fuel salt or primary cooling system salt, 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 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, “Emergency Planning for Research Reactors,” 2015.

International Atomic Energy Agency, IAEA-TECDOC-1347, Consideration of External Events in the Design of Nuclear Facilities Other Than Nuclear Power Plants, with Emphasis on Earthquakes," 2003.

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 RG. 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," Revision 2, 2017.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Chapter 3 of this guide is applicable to reviewing a description of the design of structures, systems, and components (SSC) for a non-power MSR. This chapter gives guidance for reviewing and evaluating the principal architectural and engineering design criteria for the SSC 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., barriers, confinement, or 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 SSC that are required to ensure the following:

- Safe reactor operation
- Safe reactor shutdown and continued safe conditions
- Response to anticipated transients
- A 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 SSC 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:

    a. design for the complete range of normal reactor operating conditions
    b. 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
    c. design to facilitate inspection, testing, and maintenance
    d. design to limit the likelihood and consequences of fires, explosions, and other potential man-made conditions
    e. quality standards commensurate with the safety function and potential risks
    f. design bases to withstand or mitigate wind, water, and seismic damage to reactor systems and structures
    g. analysis of function, reliability, and maintainability of systems and components

In this section the applicant should identify the SSC 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 SSC.

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 SSC 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 SSC 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 SSC 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 SSC 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 safety-related SSC 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 SSC 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 SSC 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 SSC 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 SSC 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 SSC 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 SSC 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 SSC 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. If local building codes are insufficient to ensure operability of safety related SSCs such that dose limits could potentially be exceeded, more stringent acceptance criteria will be necessary.
- 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 SSC discussed in the SAR. For any SSC 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 reviewer should ensure that appropriate detail on the scope and complexity of the seismic models used to determine potential seismic damage to SSCs is provided in the SAR. In addition to SSC, consideration should be given to movement of the liquid fuel and to any shutdown mechanisms that are susceptible to changes in shape or dimensions. 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.

The seismic event considered in the analyses 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 seismic guidance. In addition, IAEA-TECDOC-1347 contains guidance on the seismic design of SSC.

The above guidance is applicable to research reactors licensed by NRC. For test reactors the requirements of 10 CFR 100 must be applied. The guidance and criteria of 10 CFR 100 is complete and is 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 SSC 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 SSC 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 REFERENCES

American National Standards Institute, ANSI N323, 'Radiation Protection Instrumentation Test and Calibration," ANS, LaGrange Park, Illinois, 1978 (R1996).

American National Standards Institute, ANSI N323c, “Radiation Protection Instrumentation Test and Calibration – Air Monitoring Instruments,” ANS, LaGrange Park, Illinois, 2009.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.2, "Quality Control for Plate-Type Uranium-Aluminum Fuel Elements," ANS, LaGrange Park, Illinois, 1999 (R2016).

American National Standards Institute/American Nuclear Society, ANSIIANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1995 (R2018).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiological Protection at Research Reactors," ANS, LaGrange Park, Illinois, 2016.

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 (R1986). (withdrawn)

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

International Atomic Energy Agency, IAEA-TECDOC-1347, " Consideration of External Events in the Design of Nuclear Facilities Other Than Nuclear Power Plants with Emphasis on Earthquakes," 1985.

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

4 MOLTEN SALT REACTOR DESCRIPTION

Chapter 4 of this guide is applicable to reviewing a description of the design and functional characteristics of the reactor for the licensing of a non-power MSR. In this chapter of the SAR, the applicant should discuss and describe the principal features, operating characteristics, and parameters of the MSR. The analysis in this chapter should support the conclusion that the reactor design provides reasonable assurance of adequate protection of public health and safety through safe operation and shutdown under all credible operating conditions. Information in this chapter of the SAR should provide the bases for many systems, subsystems, and functions discussed elsewhere in the SAR and for many technical specifications (TS).

In following the instructions in this chapter for the MSR, it should be noted that the fuel salt performs the function of the fuel and coolant. In the following sections, any direct reference to a moderator applies to designs that might use a moderator. It should also be noted that no fuel cladding is used in the MSR design; consequently, the concept of radionuclide barrier performed by the cladding is no longer valid. The cladding's role is now performed by the fuel system boundary, including the reactor vessel and the boundaries of any penetrations (control element channels, instrumentation systems thimbles, fuel salt/primary cooling system salt heat exchangers) and fuel transfer pipes and tanks in and outside the MSR vessel.

This chapter gives guidance for evaluating the SAR description 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 TS. The systems that should be discussed in this chapter of the SAR include the active reactor core, vessel, gas management system, and biological shield. The nuclear design of the reactor and the way systems work together should also be addressed. In this chapter the applicant should explain how the design and proper operation of an MSR 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 way the facility design principles achieve the principal safety considerations. For the items requested, this section should include summaries of the format and content guide and descriptive text, summary tables, drawings, and schematic diagrams.

4.2 ACTIVE REACTOR CORE

This section of the SAR should contain the design information on all components of the active 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 active reactor core components and how they constitute a functional MSR that could be operated and shut down safely.

By reviewing this section, the reviewer gains an overview of the active reactor core design and assurance that the SAR describes a complete operable MSR. Subsequent sections should contain a description and analysis of the specifications, operating characteristics, and safety features of the reactor components. Although cooling systems should be discussed in Chapter 5, “Molten Salt Reactor Coolant Systems,” of the SAR, 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, including the use of the vessel as fuel barrier and radionuclide barrier
- Control elements
- Neutron moderator (if any) and neutron reflector
- Neutron startup source
- Vessel
- Gas management system

The information in the SAR for each active reactor 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, as well as a reference to Chapter 7
- TS requirements and their bases, including testing and surveillance, or a reference to Chapter 14, “Technical Specifications”

4.2.1 Reactor Fuel

Areas of Review

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 an MSR should be the maintenance of fuel system boundary integrity under any conditions assumed in the safety analysis. Loss of integrity is defined as the escape of any fuel and fission products from the fuel system boundary beyond those releases identified in the design basis. Since the fuel in an MSR is a liquid salt without cladding or encapsulation, the fuel system boundary is the interface surface between the fuel salt, including fission products, and any egress point. During operation, this interface includes the vessel, the gas management system, the fuel salt/ primary cooling system salt heat exchanger(s), the control element thimbles, and any pipes used for transferring fuel from and to the active reactor core and tanks that store the radioactive fuel salt. Therefore, the fuel salt must be shown to be compatible with the materials of construction for the fuel barrier and the coating (including fission products) for any normal or upset condition. The reviewer should be able to conclude that the applicant has included all information necessary to establish the limiting characteristics beyond which fuel system boundary 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 barrier. Analyses in this section of the SAR should address mechanical forces and stresses; corrosion and erosion of the fuel barrier, or collection of fission products, decay daughters, or fuel precipitates on the fuel barrier, whether caused by changes in salt chemistry (such as redox, density, pressure, and temperature) or from normal operation; hydraulic forces, including natural convection in the fuel salt; thermal changes and temperature gradients; and internal and external pressures from the production of fission gas, including pneumatic pressures related to fuel transfer. 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 TS.

Acceptance Criteria

The acceptance criteria for 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 to the bases. Maintaining fuel system boundary integrity should be the most important design objective.
- The chemical and physical characteristics of the fuel constituents, including the salt and any stabilizing additives, should be chosen for compatibility with each other and the anticipated environment, including interaction with the fuel barrier. Consideration should be given to fission product buildup in or precipitation from the homogeneous fuel salt.
- Fuel enrichment in MSRs using uranium should be consistent with the requirements of 10 CFR 50.64, "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Non-Power Reactors."
- The fuel operating parameters should take into account characteristics that could limit fuel barrier integrity, such as heat capacity and conductivity, melting, and softening temperatures of the vessel and cooling interface materials; corrosion and erosion caused by coolant or fuel salt; chemical compatibility of the fuel salt with the Fuel barrier; physical stresses from mechanical or hydraulic forces (internal pressures, vibration, and Bernoulli forces); fuel composition changes; radiation damage to the fuel barrier; and retention of fission products.
- The fuel design should include the nuclear features of the active reactor core, such as structural materials with small neutron absorption cross-sections and minimum impurities, neutron reflectors, and burnable poisons, if used.
- The various phenomena that result in changes to the initial fuel composition and properties should be considered. The submittal should include information on fission product gas formation; sparging gas if used; the transport, changes in void fraction, and removal of gas; associated redox changes, potential fuel and fission product precipitation, and the addition of fuel and salt, along with the reactivity implications of these items.
- The discussion of the fuel should include a summary of the fuel development, qualification, and production program.
- The applicant should propose TS 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 TS in this section of the SAR.

Technical Rationale

The parameters included in the technical review have been identified as important, based on experience with previous operating MSRs, as discussed in References 1 through 4.

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 active reactor 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 NRC staff's safety evaluation report:

- The applicant has described in detail the fuel salt to be used in the reactor. The discussion includes the design limits (chemical and physical) and clearly gives the technological and safety-related bases for these limits.
- The applicant has discussed the constituents, materials, components, and preparation specifications for the fuel. Compliance with these specifications for all fuel used in the reactor 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 TS.
- Information on the design and development program for the fuel offers reasonable assurance that the fuel can function safely in the reactor without adversely affecting the health and safety of the public.

4.2.2 Control Elements

Areas of Review

The control elements in an MSR are designed to change reactivity by changing the amount of neutron absorber (or fuel) or reflection in or near the active reactor core. Control elements can be designated by their material, phase, and their intended function in the reactor. To trip the reactor, the negative reactivity of the control elements is usually added passively and quickly. Because the control elements can 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 reactor trip does not challenge the safety of the reactor or cause any undue strain on any systems or components associated with the reactor. Other systems may also act to reduce reactivity on a reactor trip such as adjusting reactor fuel flow, dumping fuel salt to a drain tank, securing the addition of fresh fuel, and securing operation of the fuel salt cleanup system.

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

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

- The control elements, blades, followers, movable reflectors (if used), and support systems should be designed with adequate margin to withstand all anticipated stresses and challenges from mechanical, hydraulic, and thermal forces and the effects of their chemical and radiation environment. If the control elements are susceptible to dimensional changes, then the seismic analysis acceptance criteria should reflect the potential failure mechanisms and consequences.
- The control elements should be sufficient in number and reactivity worth to demonstrate that it is possible to shut down the reactor and comply with the requirement of minimum shutdown margin if the highest worth scrammable control element fails to insert negative reactivity. The control elements 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 elements 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 elements should be designed so that tripping them does not challenge their integrity or operation or the integrity or operation of other reactor systems.
- The control element 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 element should be independent from other control elements to prevent a malfunction in one from affecting the actuation of any other.
- The drive speeds and scram times of the control elements should be consistent with reactor kinetics requirements, considering mechanical friction, hydraulic resistance, and the electrical or magnetic system.
- The control elements should allow replacement and inspection, as required by operational requirements and the TS.
- The action of the control element (manual or automatic) should be such that it does not affect the stability of the active reactor core, but a return to a stable state following small perturbations (including physical ones from fission product gas formation, sparging gas if used, and changes in void fraction), if the active reactor core is designed within an acceptable power density limit.
- TS 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 (LCOs) and surveillance requirements, and they should be justified in this section 4.2.2 of the SAR.

Review Procedures

The reviewer should confirm that the design bases for the control elements 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 element 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 elements designs.
- The applicant has included information on the materials, components, and fabrication specifications of the control element systems. These descriptions offer reasonable assurance that the control elements conform to the design bases and can control and shut down the reactor safely from any operating condition.
- Information on scram design for the control elements has been compared with designs at other non-power reactors having similar operating characteristics. Reasonable assurance exists that the reactor trip features designed for this facility will perform as necessary to ensure fuel system boundary integrity and to protect the health and safety of the public.
- The control element 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 TS.
- Changes in reactivity caused by control element dynamic characteristics are acceptable. The staff evaluations include 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, LCOs, and surveillance requirements for the control elements and included them in the TS.

4.2.3 Neutron Moderator and Neutron Reflector

Areas of Review

In this section of the SAR, the applicant should describe neutron moderators (if applicable) and reflectors and their special features. The information pertinent to this section, therefore, applies to any moderator that might be added to the MHR design. The reflectors are chosen primarily for favorable nuclear properties and physical characteristics. Section 4.2.1 of the SAR should contain a description of the relationship of all moderators (if applicable) to the active reactor core. Buildup of contaminating radioactive material in the moderator (if applicable) or reflector during reactor operation should be discussed in Chapter 1, "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
- Provisions for cooling

Section 4.5 of the SAR should discuss the nuclear characteristics of the moderator.

Acceptance Criteria

The acceptance criteria for information on neutron moderators (if applicable) and reflectors include the following:

- The nonnuclear design bases, such as reflector encapsulations, should be clearly presented, and the nuclear bases should be briefly summarized. Nonnuclear design considerations should ensure that the moderator (*if applicable*) and reflector can provide the necessary nuclear functions.
- The design should ensure that the moderator (*if applicable*) and reflector are compatible with their chemical, thermal, mechanical, and radiation environments.
- The design should allow for dimensional changes from radiation damage and thermal expansion to avoid malfunctions of the moderator (*if applicable*) or reflector.
- The design should provide for removal or replacement of moderator (*if applicable*) or reflector components and systems, if required by operational considerations.
- TS, 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 LCOs and surveillance requirements. The proposed TS should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the information on the neutron moderator (if applicable) 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 (*if applicable*) and reflector are integral constituents of an active reactor core; the staff's evaluation of the nuclear features appears in Section 4.5. The designs take into account interactions between the moderator (if applicable) or reflector and the reactor environment. Reasonable assurance exists that degradation rates of the moderator (*if applicable*) or reflector will not affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the unrestricted environment.
- Moderators (*if applicable*) or reflectors are clad in [*insert name of cladding material*] if they are located in an environment where fuel salt infiltration could cause changes in neutron scattering and absorption, thereby changing active reactor core reactivity. Reasonable assurance exists that leakage will not occur. In the unlikely event fuel salt infiltration occurs, the applicant has shown that this infiltration will not interfere with safe reactor operation or prevent safe reactor shutdown.
- The moderator (*if applicable*) or reflector is composed of 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, LCOs, and surveillance requirements for the moderator and reflector and included them in the TS.

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
- Source material phase (e.g., solid material stored in a holder or liquid dissolved in the fuel salt)
- Interaction of the source and holder (if applicable), while in use, with the chemical, thermal, and radiation environment
- Design features that ensure the function, integrity, and availability of the source
- TS

Acceptance Criteria

Acceptance criteria for 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 active 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 significant decay over a 24-hour period 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 radiation 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.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs 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 the NRC (or the design has been fully described and analyzed). The staff concludes that 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 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 LCOs and surveillance requirements for the source and included them in the TS.
- The source and holder operate safely and reliably.

4.2.5 Reactor Internals Support Structures

Areas of Review

An MSR active reactor core is composed of the homogeneous fuel salt and off-gas inside the vessel; the active reactor core does not require a support structure beyond the MSR vessel. However, all other active 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 elements. Predictable fuel barrier integrity depends on stable and reproducible control element action and fuel salt flow patterns. Generally, the control elements of non-power reactors are suspended from a superstructure, which allows gravity to rapidly change MSR reactivity to shut down the reactor.

Areas of review include the design of the support structure for the active reactor core components and vessel, including a demonstration that the design loads and forces have adequate margin compared with all expected loads and hydraulic forces and that relative positions of components can be maintained within tolerances.

This section of the format and content guide discusses additional areas of review.

Acceptance Criteria

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

- The design should show that the support structure will hold the weight of all active reactor core-related components with adequate margin.
- The design should show that the support structure will conservatively withstand all hydraulic forces from anticipated flow with negligible deflection or motion with adequate margin.
- The design should consider the methods by which active reactor core components (e.g., reflector pieces, control elements, non-coolant moderator, any associated cooling systems, and the fuel transport pipe) are attached to the active reactor core support structure. The information should include tolerances for motion and reproducible positioning. These tolerances should ensure that variations will not cause reactivity design bases, cooling design bases, safety limits, or LCOs in the TS to be exceeded.
- The design should account for the effect of the local environment on the material of the support structure. The impact of radiation damage, mechanical stresses, chemical compatibility with the fuel salt and active reactor core components, and reactivity effects should not degrade the performance of the supports sufficiently to prevent safe reactor operation for the design life of the reactor.
- The design should show that stresses or forces from reactor components other than the active reactor core could not cause malfunctions, interfere with safe reactor operation or shutdown, or cause other active reactor core-related components to malfunction.
- The active reactor core of an MSR could vary in dimension, based on the purpose of the facility. Fuel could be transferred to and from the active reactor core during planned operations; consequently, there are devices to ensure that such operations do not occur inadvertently. The design for a changing active reactor core configuration should contain features such as 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 active reactor core configuration from changing while the reactor is critical or while forced cooling is required, if applicable. The design should show how the reactor would be shut down if unwanted action occurs.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs 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 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 active reactor core, including the design bases, which are derived from the planned operational characteristics of the reactor and the active reactor core design. All functional and safety-related design bases can be achieved by the design.
- The support structure includes acceptable guides and supports for other essential active reactor core components, such as control elements, nuclear detectors, and neutron reflectors.
- The support structure provides sufficient heat removal to conform to the design criteria and to prevent loss of fuel system boundary integrity from overheating.
- The support structure is composed of materials shown to be resistant to radiation damage, fuel salt erosion and corrosion, thermal softening or yielding, and excessive neutron absorption.
- The active reactor core support structure is designed to ensure a stable and reproducible active reactor core configuration for all anticipated conditions (e.g., reactor trips, fuel salt flow change, and active reactor core motion) through the reactor life cycle.
- The applicant has justified appropriate LCOs and surveillance requirements for the active reactor core support structure and included them in the TS.

4.3 VESSEL

Areas of Review

The vessel of an MSR is an essential part of the fuel system boundary and is the primary fuel barrier (including fission products). The vessel can also provide some support for components and systems mounted to the active reactor core supports.

The areas of review are the design bases of the vessel and the design details needed to achieve those bases. This section of the format and content guide discusses the information that the applicant should submit for review.

Acceptance Criteria

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

- The vessel dimensions should include thickness and structural supports, and fabrication methods should be discussed. The vessel should be designed with adequate margin to withstand all hydrodynamic, hydrostatic, mechanical, seismic, chemical, and radiation forces or stresses that could cause failure or loss of integrity of the vessel during its projected lifetime over the range of design characteristics.
- The construction materials and vessel treatment should resist chemical interaction with the fuel salt and be chemically compatible with other reactor components in the fuel system boundary. The compatibility between the vessel material and fuel salt should be addressed to prevent fuel salt leakage.
- The dimensions of the vessel and the materials used to fabricate it, as well as the position of the active reactor core with respect to the vessel, should ensure that radiation damage to the vessel is minimized, so that the vessel will remain intact for its projected lifetime.
- The construction materials and vessel treatment should be appropriate for preventing fuel salt from corroding the vessel interior.

- A plan should be in place to assess irradiation of and chemical damage to the vessel materials. Remedies for damage or a replacement plan should be discussed.
- All penetrations and attachments to the vessel below the fuel salt level should be designed to avoid malfunction and unintended loss of fuel salt.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Technical Rationale

Fuel salt chemistry has been shown to affect corrosion and result in possible loss of vessel integrity, based on experience from operation of previous reactors, as described in References 1 through 4.

Review Procedures

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

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:

- Information has been provided on gas composition (sparging and fission gases) and gas handling.
- The vessel system can withstand all anticipated mechanical and hydraulic forces, stresses, and irradiation damage effects to prevent loss of integrity, which could lead to a loss of fuel salt or fission products or other malfunctions that could interfere with safe reactor operation or shutdown.
- The penetrations and attachments to the vessel are designed to ensure safe reactor operation. Safety and design considerations of any penetrations below the fuel salt level include analyses of potential malfunction and loss of fuel salt. The applicant discusses credible fuel spill and leak scenarios in Chapter 13, of the SAR, Section 13.1.4.
- The construction materials, treatment, and methods of attaching penetrations and components are designed to prevent chemical interactions among the vessel and the fuel salt and other components.
- The inner surfaces of the vessel are designed and treated to avoid corrosion in locations that are inaccessible for the life of the vessel. Vessel surfaces will be inspected in accessible locations.
- The applicant has considered the possibility that fuel salt might leak into unrestricted areas, including groundwater, and has included precautions to avoid the uncontrolled release of radioactive material.
- Design considerations include the shape and dimensions of the vessel to ensure sufficient radiation shielding to protect personnel and components. Exposures have been analyzed, and acceptable shielding factors are included in the vessel design.
- The applicant has justified appropriate LCOs and surveillance requirements for the vessel and included them in the TS.
- The design features of the vessel offer reasonable assurance of its reliability and integrity for its anticipated life. The design of the vessel 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 and safety objective is to protect employees and the public. The second design objective is to make the shield as thin as possible, consistent with acceptable protection factors. In some MSRs, fuel salt might need to be transported to locations outside the vessel as part of normal operations, and this should be addressed in the shield design. 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 of these.

This section of the format and content guide discusses areas of review.

Acceptance Criteria

The acceptance criteria for 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, "Standards for Protection Against Radiation," and the guidelines of the facility's ALARA (as-low-as-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 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 possibility 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.
- 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.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements. The applicant should justify the proposed TS 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 employees and the public 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 radiation so as not to exceed the limits of 10 CFR Part 20 and the guidelines of the facility's 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's ALARA program.
- Reactor components are sufficiently shielded to avoid significant radiation-related degradation or malfunction.
- The applicant has justified appropriate LCOs and surveillance requirements for the shield and included them in the TS.

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 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 TS. The text, drawings, and tables should completely describe the reactor's 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.

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

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

- The information should show a complete, operable active reactor core. Control elements should be sufficiently redundant and diverse to control all proposed excess reactivity safely and to safely support shutting down the reactor and maintain it in a shutdown condition. Reactivity analyses should include individual and total control element effects.
- The information should describe anticipated power oscillations and their effects on safety-related equipment and systems. These oscillations should be shown to be self-damping and controllable.
- Anticipated fuel salt composition evolution should account for ; actinide generation and depletion (including fissile material); fission product buildup; changes in fuel salt chemical stability caused by changes in redox, temperature, pressure, density, and specific heat capacity; viscosity; poisons, both from fission products and those added by design; and changes to fuel salt composition from online fuel salt polishing or processing and changes to fuel salt composition from online addition

or removal of fuel salt, for the life of the reactor. The information should also include an analysis of the total fuel salt volume as a function of fuel composition changes.

- The analyses should show initial and changing reactivity conditions, control element reactivity worths, and reactivity worths of reflector units, as well as active reactor core components for all anticipated configurations. There should be a discussion of administrative and physical constraints that would prevent inadvertent reactivity changes that could suddenly introduce sufficient reactivity to cause prompt criticality or an analyzed safe amount, whichever was larger. These analyses should address movement, overfilling of fuel, and voiding of active reactor core components, including fission gas generation.
- The reactor kinetic parameters and behavior should be shown, along with the dynamic reactivity changes caused by the instrumentation and control systems. Analyses should prove that the control systems will prevent nuclear transients from causing the loss of fuel system boundary integrity or an uncontrolled addition of reactivity (e.g., the reactivity control system shall be designed with appropriate limits on the rate and amount of reactivity increase that could occur during a reactivity insertion accident so as to prevent compromise of the fuel system boundary).
- The information should include the magnitude and impact of the distribution of delayed neutrons, in fuel-containing systems outside the active reactor core, on the ability to control the reactor. The magnitude and rate of reactivity addition associated with sudden reduced fuel salt flow out of the active reactor core and the impact on the ability to control of the reactor should also be included.
- The information should include calculated active reactor core reactivities for possible and planned configurations of the control elements. This should include the reactivity impacts of fission gas, sparging gas, void formation and collapse, fission product gas removal, fuel salt polishing or processing, and fuel salt addition/removal. If only one active reactor core configuration will be used over the life of the reactor, the applicant should clearly indicate this. The limiting active reactor core configuration during reactor life should be indicated. This information should be used for the analyses in Section 4.6 of the SAR. The information should also include reactivities for fuel salt storage and handling outside the reactor, fuel transport to and from the vessel, and the effects of fuel salt recycling after cleanup operations (*if any*).
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Technical Rationale

Power oscillations in MSRs are expected and usually are self-limiting because of the large negative reactivity feedback coefficients. It is necessary to ensure that oscillations are bounded for proper operation of the reactor, based on the operation of previous MSRs found in References 1 through 4.

Review Procedures

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

Evaluation Findings

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

- The applicant has described the proposed initial active reactor core configuration and analyzed all reactivity conditions. These analyses also include other possible active reactor 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 fuel composition changes; fissile material buildup; the buildup and removal of fission products, both in solution and in the gas management system; fuel salt polishing or processing; redox control additives (e.g., Be); fuel salt addition or removal; and the use of poisons, as applicable.
- The reactivity analyses include the reactivity values for the components that impact reactivity such as control elements, neutron reflector and non-coolant moderator (*if applicable*). The assumptions and methods 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 elements and instrumentation is designed to prevent reactor transients that cause the reactor to exceed operating limits. The analyses address the effect of distributed delayed neutrons associated with fuel outside the active reactor core and the impact on the ability to control the reactor. The analyses also indicate the magnitude and rate of reactivity addition associated with a rapid reduction or total loss of fuel salt flow. This event should be discussed in chapter 13 of the SAR.
- The analyses show that any active reactor core components that could be flooded or voided could not cause reactor transients beyond the capabilities of the instrumentation and control systems to prevent fuel system boundary damage. This includes any impacts on reactivity associated with failures associated with pneumatic fuel transfer.
- The analyses address a limiting active reactor core that is the minimum size possible with the planned fuel considering the interacting effects of fuel selection, neutron moderators (if any) and neutron reflectors, control elements, expected voiding, and any experimental facilities. 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 an active reactor core that could be designed, built, and operated without unacceptable risk to the health and safety of the public.
- The applicant has justified appropriate LCOs and surveillance requirements for minimal operating conditions and included them in the TS. The applicant has also justified the proposed TS.

4.5.2 Active Reactor Core Physics Parameters

In this section of the SAR, the applicant should present information on active reactor core physics parameters that determine reactor operating characteristics and are influenced by the reactor design.

Areas of Review

Areas of review should include the design features of the active reactor core that determine the operating characteristics and analytical methods for important contributing parameters. The results presented in this section of the SAR should be used in other sections of this chapter.

This section of the format and content guide further discusses the areas of review.

Acceptance Criteria

The acceptance criteria for information on active 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 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 (accounting for distributed delayed neutrons), 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.
- Net coefficients of reactivity (temperature, void, and power) should 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 active reactor core configurations, power level, and fuel composition changes should not change the conclusions about reactor protection and safety, nor should they void the validity of the analyses of normal reactor operations.
- The methods and assumptions for calculating the various neutron flux densities should be validated by comparisons with results for similar reactors or previous experimental measurements. Uncertainties and ranges of accuracy should be given for other analyses requiring neutron flux densities, such as fuel composition changes, thermal power densities, fission gas production, control element reactivity worths, and reactivity coefficients. This should include a description of the method of calculating and verifying the composition changes and the fuel composition after any isotope removal. It should also include methods to analyze fission gas evolution and the generation of void spaces or collapse of voids and predict their reactivity effects.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs 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 consider 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 should 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 composition changes and reactor operating characteristics for the life of the reactor are considered in the analyses of the active reactor core physics parameters.
- The numerical values for the active 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 LCOs and surveillance requirements for the active reactor core physics parameters and has included them in the TS. The applicant has also justified the TS.

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 active reactor core and safe shutdown from any operating condition. The information should demonstrate a balance between fuel loading, control element worths, and number of control elements. The applicant should discuss and analyze potential accident scenarios, as distinct from normal operation, in Chapter 13 of the SAR.

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

The acceptance criteria for information on 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:

  a. Temperature coefficients of reactivity
  b. Fuel composition changes between reloads or shutdowns
  c. Void coefficients
  d. Fission product poison (e.g., samarium and or dissolved xenon [if applicable])
  e. Overall power coefficient of reactivity if not accounted for in the items listed previously
  f. Fuel processing/cleanup, handling, and reuse
  g. Fuel salt flow rate
  h. Experiments

- Credible inadvertent insertion of excess reactivity should not damage the reactor or fuel system boundary; this event should be analyzed in Sections 4.5 and 4.6 and Chapter 13 of the SAR.
- The minimum amount of total control element reactivity worth to ensure reactor subcriticality should be stated.
- A transient analysis should be performed that assumes that an instrumentation malfunction actuates the most reactive control element in a way that it causes a continuous ramp reactivity insertion in its most reactive region. This analysis can also be based on a credible failure of a movable experiment. The analysis should show that the reactor would not be damaged, and that fuel system boundary integrity would not be lost. Chapter 13 of the SAR should analyze reactivity additions under accident conditions.
- An analysis should be performed that examines reactivity, assuming that the reactor is operating under its maximum licensed conditions, normal electrical power is lost, and the control element of maximum reactivity worth and any non-scrammable control elements remain fully withdrawn. The analysis should show how much negative reactivity must be available in the remaining scrammable control elements so that, without operator intervention, the reactor can be shut down safely and remain subcritical without risk of fuel system boundary damage, even after temperature equilibrium is attained and all transient poisons, such as xenon, are reduced, with consideration for the most reactive active reactor core loading.
- 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 active reactor 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 (LSSSs) in the thermal-hydraulic analyses. The active reactor core configuration should be compared with other configurations to ensure that a limiting configuration is established for steady power.
- The effects of fuel salt surface phenomena on reactivity should be considered, if applicable.
- Analysis should show that power oscillations will not exceed the operational or safety limits and might include operational limits on parameters such as power density.
- The applicant should propose and justify TS for safety limits, LSSSs, LCOs, 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 activation of the most reactive control element or other credible failure that would add reactivity to the reactor would not lead to loss of fuel system boundary integrity. Therefore, the information demonstrates that the proposed amount of reactivity is available for normal operations but that it would not cause unacceptable risk to the public from a transient.
- The definition of the shutdown margin is negative reactivity obtainable by control elements to ensure reactor shutdown from any reactor condition, including a loss of normal electrical power. With the assumption that the most reactive control element is inadvertently stuck in its most reactive position, and non-scrammable control elements are in the position of maximum reactivity addition, the analysis derives the minimum negative reactivity necessary to ensure safe reactor shutdown. The applicant proposes a shutdown margin of [*xx reviewer should insert the margin specified in the SAR*] in the TS that provides reasonable assurance that the reactor can be shut down during any operating condition and remain in a shutdown state. 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 active reactor core configuration. This value is used in the calculations in the thermal-hydraulic section of the SAR to derive reactor safety limits and LSSSs, 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 the cooling conditions necessary to ensure that fuel system boundary integrity will not be lost under any reactor conditions, including accidents.

Since the fuel salt is free to move in a liquid form, the temperature within the fuel can more readily equalize; however, the power shape might still cause some hot spots, which could have adverse safety impacts in terms of instability and/or fuel and fission product precipitation, a potential that the applicant's safety analysis should address. Because some of the factors in the thermal-hydraulic design are based on experimental measurements and correlations that are a function of fuel salt conditions, the analyses should confirm that the values of such parameters are applicable to the reactor conditions analyzed.

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

The acceptance criteria for 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 system boundary integrity under all analyzed conditions. The discussion should include the consequences of these conditions and justification for the alternatives selected. It should also include the limiting power density to offset the onset of instability following perturbation to the system (including from fission gas generation).
- Safety limits, as discussed in Chapter 14 of the format and content guide, should be derived from the analyses described previously, the analyses in Section 4.5.3 of the SAR, and any other necessary conditions that provide reasonable assurance of adequate protection of public health and safety. The safety limits should include consideration of the effects of uncertainties or tolerances and should be included in the TS.
- LSSSs, as discussed in Chapter 14 of the format and content guide of the SAR, should be derived from the analyses described previously, the analyses in Section 4.5.3 of the SAR, and any other necessary conditions that provide reasonable assurance of adequate protection of public health and safety. These settings should be chosen to maintain fuel system boundary integrity when safety system protective actions are initiated at the LSSSs.
- A forced-flow reactor should be capable of switching to natural-convection flow without jeopardizing safe reactor shutdown. Loss of normal electrical power should not change this criterion. These limits should be based on the thermal-hydraulic analyses and appear in the TS.
- For MSRs, changes in the redox of the fuel salt could result in fuel or fission product plate-out or precipitation; this should be considered in the thermal-hydraulic design.
- The gas treatment system will contain fission product gas. In addition, there might be storage or drain systems and associated piping that contain fuel salt. Since these form part of the fuel system boundary, this section should consider any associated cooling systems and show their ability to maintain their functions and fuel system boundary integrity under normal and abnormal operations. Fuel salts have high melting temperatures; thus, the thermal-hydraulic analysis should consider the effects of partial or complete loss of flow in some piping as a result of freezing from loss of heaters. Additionally, the thermal-hydraulic analysis needs to address the possible effects of overcooling.

Technical Rationale

Previous experience with MSRs has indicated the importance of the interrelationship between the temperature of the fuel salt and redox.

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, redox, and peaking). The basic approach is an audit of the SAR analyses, but the reviewer may also 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. This includes fission gas generation, changes in void fraction including void collapse, and phenomena that could affect stability on the fuel salt surface. The applicant has justified the assumptions and methods and validated their results.
- All necessary information on the fuel salt hydraulics and thermal conditions of the fuel salt is specified for this reactor. The analysis has considered the various approaches and systems for heat removal, such as the gas management system, fuel transfer piping and drain or storage tanks containing radioactive fuel. The analyses give the limiting conditions of the features that ensure fuel system boundary integrity.
- Safety limits and LSSSs are derived from the thermal-hydraulic analyses. The values have been justified and appear in the TS. The thermal-hydraulic analyses on which these parameters are based ensure that overheating or overcooling during any operation or credible event will cause neither a loss of fuel system boundary integrity and unacceptable radiological risk to the health and safety of the public nor fuel or fission product plate-out or precipitation that could lead to a loss of radionuclide barrier integrity. The analysis includes methods for calculating the induced natural convection within the homogeneous fuel salt.

4.7 GAS MANAGEMENT SYSTEM

Areas of Review

This section of the SAR should contain the design information on all components of the gas management system. The design information should be presented in drawings, diagrams, text, and analysis in sufficient detail for the staff to understand the flow of evolved gases and fission products from their generation in the active reactor core to their ultimate release. Using this information, the staff should determine whether there is reasonable assurance that the gas management system can contain hazardous chemicals and volatile fission products until they can be managed safely, in accordance with regulatory requirements and can withstand any pressure transients within the reactor system.

In evaluating the analysis demonstrating these capabilities, the staff should ensure that these criteria can be met for the maximum power density that is considered credible during power oscillations. The applicant should justify the maximum fission product generation rates during power oscillations.

This section of the format and content guide discusses the areas of review.

Technical Rationale

Areas of review, acceptance criteria, and evaluation findings are all dictated by the following hazards: an inadvertent criticality outside the active reactor core, a release of gaseous fission products, and an increase in the pressure in the headspace over the active reactor core. Although the reactor will operate in a steady-state mode, power oscillations could be possible. Therefore, the design must be sufficiently robust to sustain fission product, heat generation, and pressures that will occur at peak power.

Acceptance Criteria

The design of the gas management system should be found acceptable if it meets the following acceptance criteria:

- The geometry of all equipment and piping should be favorable (e.g., subcritical when filled with optimally moderated fuel salt).
- If any portions of the equipment or piping are not in a favorable geometry, the applicant's analysis should demonstrate that no single failure can result in a criticality outside the active reactor core.
- Monitoring should be provided periodically for the long-term accumulation of fissionable material in the system.
- The construction materials must be compatible with the chemical environment such that corrosion cannot lead to a loss of confinement.
- The maximum pressure resulting from heat from fission gases must not exceed the design pressure for the system, unless redundant pressure relief features are described.
- The maximum release of fission gases must not exceed applicable regulatory criteria.
- The maximum release of hazardous chemicals must not exceed applicable regulatory criteria (this should include any potential effect on workers in the facility).
- Monitoring should be provided for concentrations of hazardous chemicals and fission products to detect buildup, clogging and leaks.

Chapter 5 contains acceptance criteria for any credited cooling function of the gas management system.

Technical Rationale

Most of these events can result in release pathways through the loss of confinement (e.g., by corrosion or over pressurization). The exception to this is criticality, which will result in the generation of more fission products (although they will be small compared with those generated during normal reactor operations). Criticality should not be allowed outside the vessel because there are no means to control it or adequately protect personnel outside such an environment. Ideally, all equipment that is connected to the vessel should have favorable geometry (i.e., the contained SNM will always have a subcritical multiplication factor). Maintaining aerosolized fuel within the active reactor core (ideally) or the favorable geometry part of the gas management system (as an anticipated upset) is crucial.

Review Procedures

The reviewer should confirm that the design of the gas management system and the associated analysis are sufficient to provide reasonable assurance of safe operation of the reactor and compliance with all applicable chemical and radiological release 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 applicant has described the system in sufficient detail to prevent criticality outside the vessel, caused by the entrainment of fissile material in the gas, slow accumulation over time, or backflow of fuel salt from the vessel.

- The applicant has designed the system to withstand the maximum pressure that could occur during credible power oscillations, so as to avoid breaching confinement and exceeding applicable regulatory limits.
- The applicant has designed the system to allow for control of the reactor during possible r increases in pressure (e.g., pneumatic fuel transfer system).
- The applicant has designed the system to be compatible with the chemical environment to which it will be exposed, avoiding corrosion that could result in a release of hazardous chemicals or fission products exceeding applicable regulatory limits.
- The applicant has designed sufficient surge capacity to contain hazardous chemicals and allow for the decay of fission products until they can be released in accordance with applicable regulatory limits.

Technical Rationale

These conclusions are driven by the consideration of hazards discussed previously.

4.8 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, “The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 2013.

Ball and Kerlin, “Stability Analysis of the Molten-Salt Reactor Experiment,” ORNL-TM-1070, 1965 explains the response of MSRE to oscillations.

DeVan and Evans, “Corrosion Behavior of Reactor Materials in Fluoride Salt Mixtures,” ORNL-TM-328, 1962 - see figure 9 in particular explains the relationship between redox, temperature, and corrosion.

Keiser, “Compatibility Studies of Potential Molten-Salt Breeder Reactor Materials in Molten Fluoride Salts,” ORNL-TM-5783, 1977 - goes into a bit more detail about measured removal rates for long term applications.

Steffy and Wood, “Theoretical Dynamic Analysis of MSRE with 233U Fuel,” ORNL-TM-2571, 1969 - also provides a good explanation of transient damping in MSRs.

5 MOLTEN SALT REACTOR COOLING SYSTEMS

Chapter 5 of this guide is applicable to reviewing a description of the cooling systems for the licensing of a non-power MSR. This chapter of the SAR contains guidance for evaluating the design bases, descriptions, and functional analyses of the MSR cooling systems. The principal purpose of the cooling systems is to safely remove the fission and decay heat from the fuel salt and dissipate it to the environment. The design of the reactor cooling systems is based on choosing among interdependent parameters, including thermal power level, research capability, available fuel type, active reactor core physics requirements, and radiation shielding.

The class of MSRs discussed in this chapter use liquid fuel rather than solid heterogeneous fuel. The homogeneous reactor fuel (fuel salt) is contained within the fuel system boundary. Reactor fuel can typically be added online, and fission products can be removed from the fuel salt online through a fuel salt cleanup system and gas management system. Under normal operation, the fuel salt moves through the active reactor core where it is brought into a critical configuration and heat is added to the fuel salt as discussed in Chapter 4, “Molten Salt Reactor Description,” of the SAR. Subsequently, the fuel salt continues through a fuel salt/primary cooling system salt heat exchanger where heat is transferred from the fuel salt to the primary cooling system containing a compatible non-fuel salt (coolant salt). The fuel salt then returns to the active reactor core. The coolant salt passes heat to the heat dissipation system through subsequent system interfaces. The subsystems may be forced convection or natural thermal convection.

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

For all other non-power MSRs, the applicant should describe and discuss all systems to remove and dissipate heat from the fuel salt. The design bases of the active reactor 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 fuel salt or primary cooling system should also be described and discussed in this chapter. Any auxiliary systems using cooling from other sources should be discussed in Chapter 9, “Auxiliary Systems,” of the SAR. The design bases of any features of the fuel salt 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,” of the SAR. 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 (TS) that are needed to ensure operability consistent with SAR analyses assumptions.

The fuel salt system of non-power MSRs could be of two basic types: forced convection and natural thermal convection. Facilities using forced-convection cooling might also be licensed to operate in natural-convection mode and should be capable of dissipating decay heat in that mode.

In this chapter, MSR applicants should describe and discuss all systems that remove and dispose of heat from the active reactor core, as well as major components including heat from any experimental facilities. 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

In this section, the applicant should give a brief description of fuel salt cooling systems, including the supplementary vessel heat removal pathways, summarizing the principal features. Information should include the following:

- Description of the fuel salt system
- Type of coolant flow in the fuel salt system: forced convection, natural convection, or both.
- Type of primary cooling system, if one is present, and the method of heat transfer.
- Type of heat dissipation system, if one is present, and the method of heat transfer to the environment.
- Description of the capability to provide sufficient heat removal to support continuous operation at full licensed power.
- Description of any supplementary methods of removing vessel decay heat.
- Description of any methods employed to prevent fuel salt from freezing in the vessel or in any attached piping or tank, such as a drain tank.
- Description of any methods employed to prevent primary cooling system salt from freezing in the piping or in any attached tank, such as a drain tank.
- Description of special or facility-unique features.

The applicant should summarize the principal features of the Primary Cooling System that are unique to the MSR. In addition to fuel salt heat removal through the fuel salt/primary cooling system salt heat exchanger, other means of heat transport from the fuel salt should be described, including the corresponding amount of heat transported from the fuel salt and the fraction of total vessel heat removed. These are the supplementary vessel heat removal pathways.

5.2 FUEL SYSTEM BOUNDARY AND FUEL SALT HEAT TRANSPORT

Areas of Review

For an MSR, the fuel system boundary provides the fuel barrier for the fuel salt. The fuel system boundary includes an interface with the primary cooling system to remove heat from the fuel salt through the fuel salt/primary cooling system salt heat exchanger. The fuel system boundary and the fuel salt/primary cooling system salt heat exchanger should have the capability to do the following:

- Remove the fission and decay heat from the fuel salt during reactor operation and decay heat during reactor shutdown.
- For most non-power MSRs, transfer the heat to the primary cooling system for controlled dissipation to the environment by the heat dissipation system.
- Maintain high fuel salt quality to limit corrosion of the fuel barrier, internal control elements (if any), components in the vessel, and other essential interfacing components.
- Prevent uncontrolled leakage or discharge of fuel salt to the unrestricted environment.

The MSR fuel salt could be corrosive to the inside of the fuel system boundary (or coating) and will contain radioactive fission products and actinides. No fuel cladding barrier exists for the fuel salt, as is characteristic of heterogeneous fuel elements in conventional non-power reactors. Instead, the MSR fuel barrier provides an analogous function to fuel cladding. Likewise, the primary cooling system salt is typically an unfueled version of the fuel salt and could be corrosive to the outside of the fuel system boundary. Because this affects the design of MSR heat removal systems, consideration should be given to the following:

- Construction materials of components and fabrication specifications of safety-related components as they relate to corrosion resistance of the fuel salt and the primary cooling salt.
- Fuel salt quality requirements for operation and shutdown conditions to prevent corrosion on either side of the fuel system boundary for heat removal components.
- Locations, designs, and functions of essential components, because these components ensure that the fuel system boundary is operable and that uncontrolled loss or discharge of fuel salt into the primary cooling system does not occur.

A fuel salt drain tank might be provided to allow for safe storage of the fuel salt in the event of a design-basis accident or for maintenance. The fuel salt drain tank maintains the fuel salt in a noncritical configuration and is cooled separately to remove decay heat. A detailed discussion of any ECS function for the fuel salt drain tank system and its activation should be provided in Chapter 6 of the SAR. The fuel salt drain tank cooling system should be discussed in Chapter 9 of the SAR. Any accident analyses regarding this system should be discussed in Chapter 13 of the SAR. If included in the design, fuel salt drain tank operation should be summarized in this section of the SAR. Specific areas of review for this section are discussed in Section 5.8 of the format and content guide.

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 fuel system boundary provides these functions. Section 5.2 of the format and content guide discusses specific areas of review for this section.

Acceptance Criteria

The acceptance criteria for information on the fuel system boundary include the following:

- Chapter 4 of the SAR should contain analyses of the active reactor core including the fuel salt parameters necessary to ensure removal of heat from the active reactor core to provide fuel system boundary integrity. Safety limits (SLs), limiting safety system settings (LSSSs), and limiting conditions for operation (LCOs) should be derived from those analyses and included in the TS. Examples of fuel salt system variables on which LSSSs and LCOs might be established are maximum thermal power level for operation in natural-convection flow, maximum fuel salt temperature, minimum fuel salt flow rate, fuel salt viscosity, and fuel salt pressure range. The analyses in this section should show that the components and the functional design of the fuel system boundary 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 MSRs 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 MSRs.
- The functional design should show that safe reactor shutdown and decay heat removal are sufficient to ensure fuel system boundary integrity for all possible reactor conditions, including potential accident scenarios. Scenarios that postulate loss of flow or loss of fuel salt should be analyzed in Chapter 13 and the results summarized in this section of the SAR. Emergency cooling system (ECS) interfaces with the fuel salt system, such as any use of an auxiliary heat exchanger or a drain tank for accident mitigation and decay heat removal, should be discussed in Chapter 6 and summarized in this section of the SAR.
- The descriptions and discussions should show that sufficient instrumentation, fuel salt parameter sensors, and control systems are provided to monitor and ensure stable fuel salt flow, respond to changes in reactor power levels, and provide for a rapid reactor shutdown in the event of loss of cooling from the primary cooling system. The fuel salt should provide a chemical environment that limits corrosion of the fuel barrier, control element surfaces, the vessel, and other essential interface components. Chapter 4 of the SAR should contain discussion and analyses of fuel salt

quality and other purity factors. Chemical conditions should be maintained, as discussed in Section 5.4 of this standard review plan.

- The applicant should discuss potential neutron activation and radiation damage in Chapter 4 of the SAR. To ensure that the design of the 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 fission product gases, soluble, and insoluble fission products and actinides will be produced in the fuel salt as a result of reactor operation. Additional radioactivity could occur as a result of neutron activation of fuel salt contaminants. Provisions for limiting personnel radiological hazards should maintain potential exposures from fuel salt 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 these radioactive species are acceptable, potential exposures should not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility ALARA program.
- Because the fuel system boundary provides essential fuel cooling and contains fission products and actinides, the system design should avoid uncontrolled release or loss of fuel salt. The effect of any special shielding features, such as fuel storage facility shielding and experimental facility shielding (e.g., beam tubes), on the function of the fuel salt system boundary and fuel salt system cooling should be discussed. Some design features to limit losses include using a guard boundary outside the fuel system boundary, providing syphon breaks in piping that enters the fuel system boundary, and providing check valves to preclude backflow. The designs and locations of such features should provide reasonable assurance that fuel system boundary failure is unlikely. A potential accident of rapid loss of fuel salt should be analyzed in Chapter 13 and summarized in this section of the SAR.
- If fuel salt were lost from the fuel system boundary, 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 fuel salt release should be discussed in Chapter 11 and summarized in this section of the SAR. Necessary surveillance provisions should be included in the TS.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the fuel system boundary meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

The acceptance criteria for information on the fuel salt drain system, if applicable, include the following:

- The fuel salt drain tank should maintain the fuel salt in a noncritical configuration.
- The fuel salt drain tank should have an independent decay heat removal system as discussed in Chapter 9 of the SAR.
- The fuel salt drain tank should have sufficient headspace to store all anticipated fission product gases and other gases generated from the salt. Alternatively, the drain tank should have its own independent gas management system or allow for gaseous fission products and other gases generated from the salt to be vented to the reactor gas management system.
- Any postulated malfunction of the fuel salt drain tank system should not cause uncontrolled loss of fuel salt or prevent a safe reactor shutdown.
- 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 as discussed in Chapter 11 of the SAR.

- The fuel salt drain tank design should provide for any necessary chemical control to limit corrosion or other degradation of the heat transfer interfaces and prevent chemical contamination of the environment.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the TS.

Review Procedures

The reviewer should compare the functional design and operating characteristics of the fuel system boundary 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 previously while considering the specific facility design under review.

The reviewer should compare the functional design and operating characteristics of the fuel salt drain tank system, if applicable, 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 previously while 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 fuel system boundary is designed in accordance with the design bases derived from all relevant analyses in the SAR.
- Design features of the fuel system boundary and components give reasonable assurance of boundary integrity under all possible reactor conditions, including potential accident scenarios. The fuel system boundary should be designed to remove sufficient fission heat from the fuel salt to allow all licensed operations without exceeding the established LSSSs that are included in the TS.
- The design and location of fuel system boundary components have been specifically selected to avoid fuel salt loss that could lead to fuel system boundary failure, an uncontrolled release of excessive radioactivity, or damage to safety systems or experiments.
- The fuel system boundary is designed to convert in a passive or fail-safe method, to natural-convection flow sufficient to avoid loss of fuel system boundary integrity. *(This feature is evaluated in conjunction with the reviews of the reactor description and accidents. It is applicable to licensing MSRs to operate with forced-convection flow.)*
- The chemical quality of the fuel salt will limit corrosion of the fuel barrier, internal control elements (if any), the inside of the vessel, and other essential components in the fuel system boundary for the duration of the license and for the projected utilization time of the fuel salt.
- Fuel system boundary instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.
- The TS, including testing and surveillance, provide reasonable assurance of necessary fuel system boundary operability for reactor operations as analyzed in the SAR.
- The design bases of the fuel system boundary provide reasonable assurance that the environment and the health and safety of the public will be protected.

In addition, if a fuel salt drain tank is incorporated into the design, 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 fuel salt system drain tank should accommodate any fuel salt heat load required as a result of normal operation or accident conditions as analyzed in Chapters 9 and 13 of the SAR.
- The fuel salt drain tank system is designed in accordance with the design bases derived from all relevant analyses in the SAR
- Design features of the fuel salt drain tank system and components give reasonable assurance of fuel salt system boundary integrity under all possible reactor conditions.
- The fuel salt drain tank system design or procedures will prevent radiation exposures or release of radioactivity to the environment that would exceed the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- Fuel salt drain tank system instrumentation and controls are designed to provide all necessary functions and to transmit information on all important system parameters to the control room.
- The TS, including testing and surveillance, provide reasonable assurance of necessary fuel salt drain tank system operability as analyzed in the SAR.

5.3 COOLING SYSTEMS

Most MSRs include a primary cooling system that interfaces directly with the fuel salt through the fuel salt/primary cooling system salt heat exchanger(s). The guidance for evaluating the design bases, descriptions, and functional analyses of the primary cooling system is detailed in Subsection 5.3.1. Heat is transferred from the primary cooling system to the environment through a heat dissipation system. The guidance for evaluating the design bases, descriptions, and functional analyses of the heat dissipation system is detailed in Subsection 5.3.2.

5.3.1 Primary Cooling System

Areas of Review

A systematic means to remove heat from the fuel salt is required unless the MSR is licensed to operate at such low power levels that no significant fuel salt temperature increases would occur during normal operation, as justified in Chapter 4. The primary cooling system of an MSR should be designed to transfer heat from the fuel salt to the heat dissipation system. Non-power reactors can be designed in three ways: with a continuously operating primary cooling system, with an on-demand primary cooling system, and without a primary cooling system. For most MSRs, the primary cooling system is designed for continuous operation at the licensed power level. Therefore, the primary cooling system should be designed to dissipate heat continuously. In this section of the SAR, the applicant should justify how any necessary heat transfer to the heat dissipation system 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 information on the primary cooling system include the following:

- The analyses and discussions of Section 5.3 should demonstrate that the primary cooling system is designed to allow the fuel system boundary to transfer heat from the fuel salt as necessary to ensure fuel system boundary integrity. The analyses should address the fuel system boundary operating with forced flow, natural-convection flow, or both for reactors licensed for both modes. The design should show that the primary cooling system is capable of transferring all necessary fission and decay heat to the heat dissipation system for all potential reactor conditions as analyzed in the SAR.
- Some MSRs might be designed with primary cooling systems that will not support continuous reactor operation at full licensed power. This is acceptable, provided the capability and such

limiting conditions as maximum fuel salt temperature are analyzed in the SAR and included in the TS.

- The fuel salt will contain radioactive contamination. The design of the primary cooling system should ensure that release of such radioactivity through the heat dissipation 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 fuel salt pressure is lower than the primary cooling system salt pressure across the fuel salt/primary cooling system salt heat exchanger(s) under all anticipated conditions; the primary cooling system is closed; or radiation monitoring and an effective remedial capability are provided. The primary cooling system should prevent or acceptably mitigate uncontrolled release of radioactivity to the unrestricted environment. Periodic samples of primary cooling system salt should be analyzed for radiation. Action levels and required actions should be discussed.
- The primary cooling 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.
- Primary cooling system salt drain tanks, if provided, should accommodate any heat load required during normal operation or during accident conditions as analyzed in Chapters 9 and 13 of the SAR.
- The primary cooling system design should provide for any necessary chemical control to limit corrosion or other degradation of the heat transfer interfaces and prevent chemical contamination of the environment.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the primary cooling system meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should verify that all reactor conditions, including postulated accidents, requiring transfer of fuel salt heat from the fuel system boundary to the primary cooling system have been discussed. The reviewer should verify that the primary cooling system is capable of transferring the amount of heat and the thermal power necessary to the heat dissipation system to ensure fuel system boundary integrity. The reviewer should also confirm analyses of primary cooling system malfunctions, including the effects on fuel system boundary 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 primary cooling system and components allow transfer of the necessary fuel salt heat from the fuel system boundary under all possible reactor conditions.
- Locations and design specifications for primary cooling system components ensure that malfunctions in the system will not lead to reactor damage, fuel system boundary failure or uncontrolled release of radioactivity to the environment.
- The primary cooling system instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.
- The primary cooling system is designed to respond, as necessary, to such postulated events as a significant reduction of forced fuel salt flow or a total loss of forced fuel salt flow. Loss of fuel salt flow could lead to an overcooling situation in the fuel salt/primary cooling system salt heat exchanger(s).

- The TS, including testing and surveillance, provide reasonable assurance of necessary primary cooling system operability for normal reactor operations.

5.3.2 Heat Dissipation System

Areas of Review

A systematic means to remove heat from the fuel salt is required unless the MSR is licensed to operate at such low power levels that no significant fuel salt temperature increases would occur during normal operation, as justified in Chapter 4. The heat dissipation system of an MSR should be designed to transfer heat from the primary cooling system to the environment. The heat dissipation system should be designed to dissipate heat continuously. In this section of the SAR, the applicant should justify how any necessary heat transfer to the environment 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 information on the heat dissipation system include the following:

- The analyses and discussions of Section 5.3 should demonstrate that the heat dissipation system is designed to allow the primary cooling system to transfer the heat from the fuel salt as necessary to the heat dissipation system to ensure fuel system boundary integrity. The design should show that the primary cooling system is capable of transferring all necessary fission and decay heat to the heat dissipation system for all potential reactor conditions as analyzed in the SAR.
- Some MSRs might be designed with heat dissipation systems that will not support continuous reactor operation at full licensed power. This is acceptable, provided the capability and such limiting conditions as maximum fuel salt temperature are analyzed in the SAR and included in the TS.
- The heat dissipation system should accommodate any heat load required in the event of a potential engineered safety feature operation or accident conditions as analyzed in Chapters 6 and 13 of the SAR.
- Heat dissipation system coolant drain tanks or storage tanks, if provided, should accommodate any heat load required during normal operation or during accident conditions as analyzed in Chapters 9 and 13 of the SAR.
- The heat dissipation system design should provide for any necessary chemical control to limit corrosion or other degradation of the heat transfer interfaces and to prevent chemical contamination of the environment.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the heat dissipation system meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should verify that all reactor conditions, including postulated accidents, requiring transfer of fuel salt heat through the primary cooling system to the heat dissipation system have been discussed. The reviewer should verify that the heat dissipation system is capable of transferring the amount of heat and the thermal power necessary to the environment to ensure fuel system boundary integrity. The reviewer should also confirm the analyses of heat dissipation system malfunctions, including the effects on the primary cooling system integrity and any resultant effects on the fuel system boundary integrity, provide reasonable assurance of adequate protection of public health and safety.

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

5.4 FUEL SALT CLEANUP SYSTEM

Areas of Review

In MSRs evaluated under this chapter, fission products are released from the liquid fuel directly into the fuel salt because there is no fuel cladding. Gaseous fission products, such as xenon and krypton, bubble off continuously and are typically collected in a cover gas space and removed through a gas management system without any significant impact on reactor operation. The gas management system is discussed in Chapters 4 and 11 of the SAR. The gas management system cooling is discussed in Chapter 9 of the SAR. Soluble and insoluble fission products remain in the fuel salt and can be removed from the fuel salt by chemical processing, polishing, or filtration. The radiological controls for the fuel salt cleanup system are discussed in Chapter 11 of the SAR. Cooling for soluble and insoluble fission product cleanup systems are discussed in Chapter 9 of the SAR. In some MSR designs, the fuel salt cleanup system might also be used to add additional fuel to the fuel salt. Fuel handling is discussed in Chapter 9 of the SAR; if appropriate, fuel addition is summarized here. The purity of the fuel salt should be maintained as high as reasonably possible for the following reasons.

- To limit the chemical corrosion of the fuel system boundary
- To maintain the thermal-dynamic properties of the fuel salt within the operational limits established for the fuel

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 information on the fuel salt cleanup system include the following:

- The fuel salt quality should be maintained in the ranges established as acceptable in Chapters 4 and 11 of the SAR.
- The geometry of all fuel salt cleanup system equipment and piping should be favorable (e.g., subcritical when filled with optimally moderated fuel salt).
- Radioactive contaminated filters and other materials associated with operation of the fuel salt cleanup system should be disposed of or regenerated in accordance with the 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.

- The location, shielding, and radiation monitoring of the fuel salt 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.
- The location and functional design of the components of the fuel salt cleanup system should ensure the following:

  a. Malfunctions or leaks in the system do not cause uncontrolled loss or release of fuel salt.
  b. Personnel exposure and release of radioactivity do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility ALARA program.
  c. Safe reactor shutdown is not prevented.

- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the fuel salt cleanup system meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should compare the design bases for the fuel salt quality with the design bases by which the fuel salt 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 other materials associated with the fuel salt cleanup system.

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 fuel salt cleanup system give reasonable assurance that the required fuel salt quality can be achieved to maintain fuel salt chemistry parameters within TS limits and to prevent or limit to acceptable levels fission product poisons, heat exchange surface corrosion, film buildup, fouling, and plugging.
- The fuel salt 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 fuel salt 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 TS, including testing and surveillance, provide reasonable assurance of necessary fuel salt cleanup system operability for normal reactor operations.

5.5 SALT MAKEUP SYSTEMS

During MSR operation, there might be a need for salt to be replaced or replenished in the fuel salt system or the primary cooling system. The guidance for evaluating the design bases, descriptions, and functional analyses of the fuel salt makeup system is detailed in Subsection 5.5.1. The guidance for evaluating the design bases, descriptions, and functional analyses of the primary cooling system salt makeup is detailed in Subsection 5.5.2.

5.5.1 Fuel Salt Makeup System

Areas of Review

It could be necessary to replace or replenish the generally solvent halide salt(s) in the reactor fuel because of operational activities or a need to adjust the fuel salt composition. Although each reactor should have a makeup salt system or procedure to meet projected operational needs for fuel chemistry, the salt addition system need not be designed to provide a rapid, total replacement of the salt inventory. In some MSR designs, the fuel salt makeup system might also be used to add additional fuel to the fuel salt. Fuel handling is discussed in Chapter 9 of the SAR; but if appropriate, fuel addition can also be summarized here. Specific areas of review for this section are discussed in Section 5.5.1 of the format and content guide.

Acceptance Criteria

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

- Any projected reduction of salt inventory in the reactor fuel for anticipated reactor operations should be discussed. The design or plan for supplying makeup salt should ensure that those operational requirements are satisfied.
- The geometry of all fuel salt makeup system equipment and piping should be favorable (e.g., subcritical when filled with optimally moderated fuel salt).
- Storage of salt for the reactor fuel should be provided as required by the design bases of the MSR or a plan should ensure that such salt is provided.
- The makeup salt system or plan should include features to prevent release of fuel salt from the fuel system boundary.
- The makeup salt system need not have a functional relationship with any installed ECS. If it does, it should not interfere with the availability and operability of the ECS.
- The applicant should describe, if appropriate, the steps taken to ensure no significant change in fuel salt characteristics occurs (e.g., redox chemistry, temperature) when makeup fuel salt is added.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the fuel salt makeup system meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should compare the design bases and functional requirements for replenishing salt in the reactor fuel, including the quantity and quality of the salt, the activities or functions that remove salt, and the systems or procedures to accomplish salt makeup with the acceptance criteria. The review should focus, as applicable, on safety precautions to preclude overfilling of the fuel system boundary and the release of fuel salt back through the salt makeup system into 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 design bases, functional descriptions, and procedures for salt makeup in the reactor fuel give reasonable assurance that the quantity and quality of salt required will be provided.
- The fuel salt makeup system design or procedures prevent overfilling of the fuel system boundary or malfunction of the makeup salt system and prevent the loss or release of fuel salt that would exceed the requirements of 10 CFR Part 20 and the facility ALARA program guidelines.
- The TS, including testing and surveillance, provide reasonable assurance of necessary makeup salt system operability for normal reactor operations.

5.5.2 Primary Cooling Makeup System

Areas of Review

During MSR operations, it could be necessary to replace or replenish the salt in the primary cooling system. Salt could be lost because of operational activities, or there might be a need to adjust the salt composition. Although each reactor should have a primary cooling makeup system or procedure to meet projected operational needs for heat removal, the system need not be designed to provide a rapid, total replacement of the primary cooling system salt inventory. Specific areas of review for this section are discussed in Section 5.5.2 of the format and content guide.

Acceptance Criteria

The acceptance criteria for information on the salt makeup systems include the following:

- The projected loss of salt inventory in the primary cooling system (or other coolants in the heat dissipation system) for anticipated reactor operations should be discussed. The design or plan for supplying makeup salt should ensure that those operational requirements are satisfied.
- Storage of salt for the primary cooling system should be provided as required by the design bases of the MSR, or a plan should ensure that such salt is provided.
- The primary cooling makeup system or plan should include features to prevent loss or release of salt from the primary cooling system.
- The primary cooling makeup system or plan should include provisions for recording the use of makeup salt to detect changes that indicate leakage or other malfunction of the primary cooling system.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the primary cooling makeup system meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should compare the design bases and functional requirements for replenishing salt in the primary cooling system, including the quantity and quality of the salt, the activities or functions that remove salt, and the systems or procedures to accomplish salt makeup with the acceptance criteria. The review should focus, as applicable, on safety precautions to preclude overfilling of the primary cooling system.

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 cooling makeup system give reasonable assurance that the quantity and quality of salt required will be provided.
- The primary cooling makeup system design or procedures prevent overfilling of the primary cooling system.
- The TS, including testing and surveillance, provide reasonable assurance of necessary primary cooling makeup system operability for normal reactor operations.

5.6 AUXILIARY SYSTEMS

Areas of Review

The fuel salt, primary cooling salt, or other heat dissipation system coolants might serve functions other than cooling the fuel salt. Some of these auxiliary functions involve cooling other heated components, which could affect the heat load of the respective cooling system.

Auxiliary uses of the fuel salt, primary cooling system salt, or other heat dissipation system coolants could affect system availability for fuel salt cooling, which is the principal use. Although the principal discussions of these auxiliary systems should be located in other sections of the SAR, their effects on the respective cooling systems should be summarized in this section. Auxiliary systems that could use primary cooling system salt or heat dissipation system coolants include the following:

- Experiment cooling
- Experimental facility cooling
- Biological shield cooling
- Thermal shield cooling
- Fuel storage cooling and shielding
- Reflector cooling
- Drain tank cooling

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 information on the auxiliary systems using fuel salt, primary cooling salt, or other heat dissipation system coolants include the following:

- The systems should remove sufficient projected heat to avoid damage to the cooled device.
- The system should not interfere with required heat removal from the fuel salt.
- Any postulated malfunction of an auxiliary system should not cause uncontrolled loss of fuel salt or prevent a safe reactor shutdown.
- 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 propose TS as discussed in Chapter 14 of the format and content guide to ensure that auxiliary systems meet the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should verify that auxiliary cooling using fuel salt, primary cooling salt, or other heat dissipation system coolants is described in this section of the SAR for any component in which potentially damaging temperature increases or excessive radiation exposures are predicted. If the potential exists for radiation heating of components near the vessel, the reviewer should verify that the heat source, temperature increases, heat transfer mechanisms, and heat disposal have been discussed and analyzed.

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 (1) described and analyzed auxiliary systems that use fuel salt, primary cooling salt, or other heat dissipation system coolants for functions other than fuel salt cooling; (2) derived the design bases from other chapters of the SAR; (3) analyzed any reactor components located in high-radiation areas near the active reactor core for potential heating that could cause damage to the fuel system boundary or failure of the component; and (4) planned acceptable methods to remove sufficient heat to ensure the integrity of the components. Cooling for these systems is obtained from the fuel salt, primary cooling salt, or other heat dissipation system coolants without decreasing the capability of any system below its acceptable performance criteria to maintain fuel system boundary integrity.
- The TS, including testing and surveillance, provide reasonable assurance of necessary auxiliary cooling system operability for normal reactor operations.

5.7 REFERENCES

MSRE Design and Operations Report, Part 1: Description of Reactor Design, ORNL-TM-728, Robertson, 1965.

The Development of Technical Specifications for Research Reactors, American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, ANS, LaGrange Park, Illinois, 2013.

6 ENGINEERED SAFETY FEATURES

Chapter 6 of this guide is applicable to reviewing a description of the engineered safety features for the licensing of a non-power MSR. This chapter gives the review plan and acceptance criteria for active or passive engineered safety features (ESFs) of the MSR 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 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 MSR is defined as operation with all process variables and other reactor parameters within allowed conditions of the license, technical specifications (TS), applicable regulatory limits, and design requirements for the system. Accidents at non-power MSR facilities assume failure of a major component such as the fuel salt system boundary or a reactivity addition event. Licensees analyze a maximum hypothetical accident that assumes an incredible failure that leads to unacceptable fuel salt system boundary degradation. These postulated accidents are compared with acceptance criteria such as the safety limits from the TS or, where there are radiological consequences, to accepted regulatory limits (10 CFR Part 20 or 100). Consideration must also be given to the fission product decay heat generated in the interfacing gas management system and fuel salt cleanup system (if applicable). Cooling systems described in Chapter 9, "Auxiliary Systems," are designed to provide normal cooling for these systems. However, the maximum hypothetical accident that assumes an incredible failure should also consider these interfacing systems because of the radioactive material content and the potential radiological consequences to accepted regulatory limits (10 CFR Part 20 or 100) resulting from interfacing system boundary degradation. The results of the accident analyses are presented in SAR, Chapter 13, "Accident Analyses." ESF systems must be designed to function for the range of conditions from normal operation through accident conditions.

Accident scenarios that should be discussed by the applicant in Chapter 13 of the SAR include the following:

- Loss of fuel salt
- Loss of fuel salt flow
- Insertion of excess reactivity (rapid or ramp)
- Loss of fuel system boundary integrity or mishandling of fuel salt
- Loss of gas management system integrity or fuel salt cleanup system integrity
- Precipitation of fuel
- Failure or malfunction of an experiment
- Other uncontrolled release of radioactive material
- Loss of electrical power
- External events such as floods and earthquakes

The SAR accident analyses for a non-power MSR may support the conclusion that ESFs are not required, even for the maximum hypothetical accident. In other cases, the accident analyses may conclude that ESFs need to be considered in mitigating the potential release of hazardous quantities of radioactive material to the environment. Note that there could be several systems containing highly radioactive materials within separate boundaries that should be analyzed apart from the fuel system boundary when considering the maximum hypothetical accident.

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 MSRs are designed to ensure adequate public safety, 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 TS. The TS surveillance requirements for system components that ensure the integrity and operational capability of the ESFs should also be reviewed.

MSR research reactors must show that exposures meet the requirements of 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.

For a non-power MSR licensed as a testing facility, the reviewer should compare the results of the accident analyses 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 how the transfer between normal and emergency sources of electricity, 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 direct reactor auxiliary cooling system to the primary cooling system or the confinement to the reactor room ventilation system).

Typical ESFs that might be required for a proposed design are the confinement, containment, and emergency cooling system (ECS), which are discussed in this chapter of the format and content guide. The postulated accident analyses by the applicant determine whether an MSR facility needs confinement, containment, an ECS, or no ESFs. The reviewer will find that heating, ventilation, and air-conditioning (HVAC) and air exhaust systems at MSRs 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 ECS ESFs. Information on any additional ESFs required at non-power MSRs can be evaluated by the reviewer in a similar manner.

Most MSR designs will likely include containment or a set of barriers that act as a functional containment. 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, containment ESF would not be required. Some licensees have chosen to build containments as additional design conservatism.

6.1 SUMMARY DESCRIPTION

In this section of the SAR, the applicant should briefly describe all of the ESFs in the facility design and summarize the postulated accidents for which 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, performance criteria, and 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, containment, and ECS 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 might 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.

During normal operations, the MSR could release small amounts of radioactive material. Specifically, relatively small amounts of fission product gaseous fission products could escape from the reactor primary radionuclide barrier. The applicant should describe how these releases to the environment will be controlled so that they are within the limits for gaseous effluents in 10 CFR Part 20 and neither the public nor the facility’s operating staff will receive radiation doses 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 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

  a. seals, gaskets, filters, and penetrations (e.g., electrical, experimental, pneumatic, and cooling medium);
  b. necessary ESF equipment included as part of the confinement; and
  c. 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-references to other chapters for discussion of normal operations (such as Chapter 4, “Molten Salt 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 TS that ensure operability and availability of the confinement ESFs, when required.

Acceptance Criteria

The acceptance criteria for 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. It should be designed to remain functional for the full range of potential operational conditions, including the environment created by accident scenarios.
- The applicant should propose TS as discussed in Chapter 14, “Technical Specifications,” of the format and content guide to ensure that the confinement meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

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, facility staff, or the environment.
- The staff has reviewed the designs and functional descriptions of the confinement ESF and has reasonably ensured 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, environment, and 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 that are as far below the regulatory limits as can be reasonably achieved.
- The TS, including testing and surveillance, provide reasonable assurance of necessary confinement operability for reactor operations as analyzed in the SAR.

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.

Most MSRs will likely include containment or a collection of barriers that act as a functional containment. In fact, it is possible that multiple individual containments will be included around interfacing systems containing highly radioactive materials such as the gas management system or the fuel salt cleanup system.

Containment is considered necessary for MSR 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 might show that a confinement is an acceptable ESF, but the applicant chooses to construct a containment for additional conservatism.

Most MSRs release a small amount of radioactive material during normal operation. Even though the quantity of radioactive material produced might 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 fission product gaseous radionuclides, which can 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 Chapter 13 of the SAR. The analyses should show whether any postulated accident could cause an unacceptable radiological exposure, as discussed previously, to the public, environment, or 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.

MSRs 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 the 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, pneumatic, and cooling medium);
    - necessary ESF equipment included as part of the containment; and
    - fabrication specifications for essential and safety-related components.
- Discussion and analyses, keyed to drawings, of how the structure provides the necessary containment presented in Chapter 13, with cross-references 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 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 TS that ensure operability and availability of the containment, when required.

Acceptance Criteria

The acceptance criteria for 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. It should be designed to operate in the environment created by the accident scenario.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the containment meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should review the accident scenarios and 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 whether 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 testing facilities, 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 Chapter 13 of the SAR for all analyzed accidents.
- The TS, including testing and surveillance, provide reasonable assurance of necessary containment operability for reactor operations as analyzed in the SAR.
- The design of the containment ESF gives reasonable assurance that it will not interfere with reactor operation or shutdown.

6.2.3 Emergency Cooling System

Areas of Review

For most MSRs, heat must be removed from the fuel salt during normal operations, and decay heat from radioactive fission products in the fuel salt must be removed after the reactor is shut down. Cooling systems described in Chapter 5, "Molten Salt Reactor Cooling Systems," are designed to provide these functions. If cooling is accidentally lost, the decay heat in some non-power MSRs could be high enough to require an emergency fuel salt cooling system to avoid unacceptable fuel system boundary degradation as a result of high system temperatures. In addition, decay heat must be removed from interfacing systems with radioactive fission products such as the gas management system and fuel salt cleanup system. Cooling systems described in Chapter 9 typically provide these functions.

In Chapter 13 of the SAR, each applicant should present analysis of 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 ECS through the progression of postulated accident scenarios, including the maximum hypothetical accident, if relevant.

Acceptance Criteria

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

- The design bases and functional description should be derived from postulated accident scenarios, including the maximum hypothetical accident, if relevant, and be presented in Chapter 13 of the SAR. This includes interfacing systems with radioactive fission products such as the gas management system and fuel salt cleanup system.
- The design features ensure that the ECS will provide decay heat removal for the time interval required by the scenario. The design features ensure that any necessary utility sources, such as normal electricity, emergency power, and cooling, will be available to the ECS.
- The ECS should not interfere with either normal operations or reactor shutdown.
- The consequences of postulated accidents, as mitigated by the ECS, 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.
- The applicant should propose TS as discussed in Chapter 14 of the format and content guide to ensure that the ECS meets the safety-related design requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

The reviewer should evaluate the accidents in Chapter 13 of the SAR to determine the scenario and consequences for each accident and to ascertain if the integrity of the fuel system boundary or any interfacing systems can be compromised. The reviewer should verify that the proposed ECS can prevent or mitigate degradation of the fuel salt and the fuel salt boundary. The reviewer should compare the design details of the proposed ECS with the design and functional requirements of the SAR accidents in Chapter 13 and the mitigated radiological consequences with 10 CFR Part 20 for research reactors, or 10 CFR Part 100 for test reactors to determine whether 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 staff's safety evaluation report:

- The applicant has identified a potential or maximum hypothetical accident that could lead to unacceptable fuel system boundary degradation or loss of fuel system boundary integrity and unacceptable radiological consequences, if relevant to the ECS.
- The applicant has identified a potential or maximum hypothetical accident that could lead to unacceptable interfacing system degradation or loss of integrity and unacceptable radiological consequences, if relevant to the ECS.
- The applicant's analysis of this accident in Chapter 13 includes a proposed ECS whose design and function is to cool the fuel to prevent failure of the fuel system boundary and associated containment.
- The ECS would not interfere with normal operations and would not prevent safe reactor shutdown.
- The design and operation of the ECS would not lead to uncontrolled release of radioactive material.
- The TS, including testing and surveillance, provide reasonable assurance of necessary ECS operability as analyzed in the SAR.
- The design of the ECS 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 provides for alternative sources, if necessary.

- The functioning of the ECS as designed reasonably ensures that the maximum hypothetical accident at the reactor facility, if relevant, would not subject the public, environment, or facility staff to unacceptable radiological exposure.

6.3 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1-R2013, "*The* Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 2013.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.21-2012, "Format and Content for Safety Analysis Reports for Research Reactors," ANS, LaGrange Park, Illinois, 2012.

7 INSTRUMENTATION AND CONTROL SYSTEMS

Part 2, Chapter 7, of the standard review plan and acceptance criteria, is applicable to reviewing a description of the instrumentation and control (I&C) systems for a non-power MSR.
In this chapter of the SAR, the applicant describes the design and operating characteristics of the I&C systems. 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; reactor fuel flow, temperature, and level; primary cooling system 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.

In MSRs, fission product gases build up within the vessel, or at a free surface provided above the liquid fuel. Gases generated during fission also collect in the cover gas interface with the fuel salt and are removed through the gas management system. Therefore, I&C subsystems relevant to the head space and gas management systems should be provided. As described in Chapter 4, “Molten Salt Reactor Description,” I&C subsystems should support the limiting conditions for operation of the homogenous fuel salt (reactor fuel) and the fuel system boundary. These limits are specified to ensure that the integrity of the fuel system boundary will not be impaired by loss of redox control, changes in reactor fuel flow, fuel addition, fission gas evolution, power oscillations, precipitation from the reactor fuel, temperature and pressure extremes or distributions, and materials compatibility. In addition, fuel handling systems as described in Chapter 9, “Auxiliary Systems,” and fuel salt cleanup systems as described in Chapter 5, “Molten Salt Reactor Cooling Systems,” should be monitored by I&C systems.

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 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 testing facilities. 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, such as by adjusting reactor fuel flow, adding fuel, operation of the fuel salt cleanup system, or manipulating control elements. 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 active reactor core neutron flux; reactor fuel flow, temperature, and level at the interface with the cover gas; primary cooling system flow, temperature, and level; heat dissipation system flow and temperature; area radiation intensities in selected areas around the reactor, including the gas management system, cleanup system, and fuel handling system; 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 bases and design criteria
- 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 radiation, and possibly fuel salt temperature within the reactor for operational purposes
- process instruments- instruments designed to measure and display such parameters as fuel salt flow, temperature, or level at the fuel salt interface with the gas management system; fuel salt thermophysical properties; gas management system parameters; fuel salt addition; fuel salt drain; or fuel salt cleanup parameters. (Note that overcooling in associated heat removal systems could interrupt fuel salt flow)
- control elements- types, number, function, design, and operating features of reactivity control devices. Control elements can act through fuel displacement, neutron absorption, neutron reflection, neutron spectral adjustment, or a combination of these methods. Control elements can be solids, liquids, or gases, and they can be passively or actively positioned (coordinate with the review of Chapter 4, "Reactor Description")
- interlocks- circuits or devices to inhibit or prevent an action, such as control element 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 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 and rate of change of power levels during reactor startup, the full range of normal operation, and shutting down the reactor. 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 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 be designed with sufficient control of reactivity for all required reactor operations and ensure compliance with analyzed requirements on shutdown margin.
- 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.
- Regulatory Guide (RG) 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," provides guidance that the NRC staff also deems acceptable for complying with the Commission's regulations for promoting high functional reliability and design quality of hardware and software for computerized systems in NPUFs. Additionally, the guidelines of IEEE 7-4.3.2-2016, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," which serves to amplify criteria in IEEE Std 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" also has guidance applicable to NPUFs. Software testing should be consistent with the applicable guidance of ANSI/ANS 10.4-2008, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," that apply to NPUF systems.

ANSI/ANS 15.15-1978, "Criteria for the Reactor Safety Systems of Research Reactors" provides general guidance for the design, implementation, and evaluation of I&C systems for non-power reactors and, although withdrawn, provides useful guidance, as applicable.

- The RCS should be designed for reliable operation in the normal range of environmental conditions (e.g., temperature, humidity, electromagnetic compatibility and susceptibility, etc.) 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 (TS), 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 and the introduction and emission rate of any neutron startup source should allow a monitored startup with the reactor instrumentation (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 active reactor 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.
- Automatic and manual manipulation of control elements 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. Control element status or state should be clearly indicated for operator or interlock use.
- The applicant should plan and discuss how all control elements, their driver and release devices, and displays 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: control element status; power level or reactor period recorders; startup neutron counter; fuel salt flow, temperature, or level conditions; fuel addition status, operation of the fuel salt cleanup system, experimental facility status, high radiation areas; confinement or containment systems; 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, and safe reactor shutdown cannot 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 and the planned reactor uses. The applicant has also analyzed the functions of the 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 or common cause failure 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 TS 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 a combination of dumping fuel salt to a drain tank, minimizing reactor fuel flow, manipulating control elements, curtailing fuel addition, or shutdown of the fuel salt cleanup system. whenever any of the selected parameters exceeds predetermined limits, in order to prevent damage to the fuel system boundary. The automatic protective function may also be initiated manually by the operator. Parameters typically monitored for this purpose include active reactor core neutron flux; reactor fuel flow, temperature, and flow; primary cooling system flow, temperature, and level, or radioactivity; and reactor area radiation levels in selected areas around the reactor, including the gas management system, fuel salt cleanup system, and fuel handling system. 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 power reduction. The reviewer should examine how the RPS automatically initiates rapid operation of the available means of reactivity control 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 subsystems.
- 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 or common cause failure and to meet requirements for seismic conditions, environmental qualification, redundancy, diversity, independence, and protection of equipment important to safety against other natural phenomena and external events.

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 comprised 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 and design criteria 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.
- TS, 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 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.
- Any automatic reactor power reduction subsystem 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 (e.g., temperature, humidity, electromagnetic compatibility and susceptibility, etc.) 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 TS.
- 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.
- To the extent applicable, hardware and software for computerized systems should meet the guidelines of Regulatory Guide 1.152 and IEEE 7-4.3.2-2016. ANSI/ANS 15.15 may also be useful as a general guide for the design implementation, and evaluation of I&C systems for NPUFs and should be used where applicable.

Review Procedures

The reviewer should compare the design bases for the RPS with SAR analyses of possible hazards to the MSR NPUF 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 design 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 MSR NPUFs 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 RPS for the (*insert name of facility*). The protection channels and protective responses are sufficient to ensure that limiting safety system settings, or RPS-related limiting conditions of operation discussed and analyzed in the SAR will provide for safe operation of the facility. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit 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 that 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 function independently of any 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 or common cause failure 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 included 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 TSs, including surveillance tests and intervals that are designed to ensure operability of the ESF actuation systems.

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 that would result from any postulated accident until the accident has been brought to a stable condition.

- The equipment should be designed to be readily tested and calibrated to ensure operability.
- TS 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.
- To the extent applicable, hardware and software for computerized systems should meet the guidelines of Regulatory Guide (RG) 1.152 and IEEE 7-4.3.2-2016. Software should meet the guidelines of ANSI/ANS 10.4-2008 (R2016) that apply to NPUF systems.
- ANSI/ANS 15.15 may also be useful as a general guide for the design implementation and evaluation of I&C systems for NPUFs and should be used to the extent applicable.

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 environmental conditions 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 TSs, 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 NPUF control room, containing the control console and other status display instruments is the hub for 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. Parameters monitored to safely operate the facility may include the following: fuel system boundary integrity issues; gas management system failures; the need to actuate barrier, containment, or confinement systems; the need for personnel radiation protective actions; and to monitor release of radioactive material to the environment. The console contains most of the circuitry and hardware for organizing and processing the information using either analog or digital systems, 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
- analog 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, equipment, and system 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 TS 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 human-machine interface 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
- A control console instrument system failure should not prevent the RPS from performing its safety function and should not prevent safe reactor shut down
- 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 all normal and transient conditions of operation
- 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 TS, including surveillance tests and intervals for control console devices, should be discussed in this section of the SAR
- The outputs and display devices required for safe operation of the MSR NPUF should be readily observable by the operator while positioned at the control console systems.
- The control elements in an MSR are designed to change reactivity by changing the amount of neutron absorber (or fuel) or reflection in or near the active reactor core. Depending on their

function, control elements can be designated by their material, phase, and their intended function in the reactor. Control element position/status indication and operational 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 active reactor 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 facilities, ESF initiation, radiation field intensity or radioactivity, and confinement or containment status.
- To the extent applicable, hardware and software for computerized systems should meet the guidelines of Regulatory Guide (RG) 1.152 and IEEE 7-4.3.2-2016. Software should meet the guidelines of ANSI/ANS 10.4-2008 (R2016) that apply to NPUF systems.
- ANSI/ANS 15.15 may also be useful as a general guide for the design implementation and evaluation of I&C systems for NPUFs and should be used to the extent applicable.
- 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 facility. The reviewer should compare the design bases and functional requirements of other MSR 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.

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 MSR 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 MSR controls. The control console design and operator interface are sufficient to promote safe MSR 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 are readily testable and 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 MSR operation, even if the console does not include a parameter display
- The locking system on the control console reasonably ensures that the MSR 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 an MSR 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 fuel salt level as observed at the cover gas interface; fuel system boundary integrity issues; gas management system failures; fuel salt cleanup system failures; fuel addition system issues; barrier, 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 TSs 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 or common cause failure protection and to meet requirements for seismic conditions, environmental qualification, redundancy, diversity, independence, and protection of radiation monitoring systems important to safety against other natural phenomena and external events.

Acceptance Criteria

Acceptance criteria for radiation monitoring systems should include the following:

- The systems should be designed to interface with either an analog or digital RCS or RPS as applicable.
- The systems should be designed not to fail or operate in a mode that would prevent the RPS from performing its safety function and should not 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 environmental conditions in which the radiation monitoring systems are expected to operate.
- 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 TS, including surveillance tests and intervals, should be sufficient to ensure that the systems will be operable and will perform their designed functions.
- To the extent applicable, hardware and software for computerized systems should meet the guidelines of Regulatory Guide (RG) 1.152 and IEEE 7-4.3.2-2016. Software should meet the guidelines of ANSI/ANS 10.4-2008 (R2016) that apply to NPUF systems.
- ANSI/ANS 15.15 may also be useful as a general guide for the design implementation and evaluation of I&C systems for NPUFs and should be used to the extent applicable.

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 MSR NPUF facility. The reviewer should establish that the design includes sufficient monitoring points, 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 conclusion 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 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, measured, 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, 2008 (R2016).

American National Standards Institute/American Nuclear Society, ANSI/ANS 10.5, "Accommodating User Needs in Scientific and Engineering Computer Software Development," ANS, LaGrange Park, Illinois, 2006.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, “Quality Assurance Program Requirements for Research Reactors,” ANS, LaGrange Park, Illinois, 1995 (R2018).

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 (R1986). withdrawn

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.

Electric Power Research Institute, “Guideline on Licensing Digital Upgrades,” EPRI TR-102348, Rev. 1, Palo Alto, CA, December 1993.

Electric Power Research Institute, “Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications,” EPRI TR-106439, Palo Alto, CA, October 1996.

Electric Power Research Institute, “Generic Requirements Specification for Qualifying a Commercially Available PLC for Safety-Related Applications in Nuclear Power Plants,” EPRI TR-107339, Palo Alto, CA, December 1996.

Electric Power Research Institute, “Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule,” Palo Alto, CA, 2002. 1002833.

Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2-2016, “IEEE Standard Criteria for Digital Computers Systems in Safety Systems of Nuclear Power Generating Stations,” Piscataway, New Jersey, 2016.

Institute of Electrical and Electronics Engineers, IEEE Standard 603-2009, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Piscataway, New Jersey, 2009.

Institute of Electrical and Electronics Engineers, IEEE Standard 1023-2004, “IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities,” New York, New York, June 2005.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.152, Rev. 2, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” January 2006 (ADAMS Accession No. ML053070150).

U.S. Nuclear Regulatory Commission, NUREG-0700, Rev. 2, “Human-System Interface Design Review Guidelines,” May 2002 (ADAMS Accession Nos. ML021700337, ML021700342, ML021700371).

U.S. Nuclear Regulatory Commission, NUREG 0711, Rev. 3, “Human Factors Engineering Program Review Model,” November 2012 (ADAMS Accession No. ML12324A013).

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. (ADAMS Accession No. ML042430055).

U.S. Nuclear Regulatory Commission, NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” February 1996 (ADAMS Accession No. ML042430048).

8 ELECTRICAL POWER SYSTEMS

Chapter 8 of this guide is applicable to reviewing electrical power systems for a non-power MSR. In this chapter of the SAR, the applicant discusses and describes the electrical power systems at a non-power MSR 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, "Molten Salt Reactor Description"; Chapter 5, "Molten Salt Reactor Cooling 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 MSRs 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 guidance for non-power MSRs (Part 1).

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 maintain adequate heat removal or avoid salt freezing (Chapter 5), 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 system boundary integrity or uncontrolled release of radioactive material.

Areas of Review

Non-power MSRs 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 damage to the fuel system boundary 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

If no emergency power system is proposed by the applicant, 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:

- No emergency electrical system is required at the facility to maintain safe reactor shutdown; to support operation of a required engineered safety feature; or to protect the public from release of radioactive effluents.

Otherwise, 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 required 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

Chapter 9 of this guide is applicable to reviewing a description of the auxiliary systems for the licensing of a non-power MSR. This chapter of the SAR contains guidance for evaluating the information on auxiliary systems in the MSR. 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, facility staff, and 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 an MSR, 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 MSRs 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 should also 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 9.8 of the guide.

9.1 HEATING, VENTILATION, AND AIR-CONDITIONING SYSTEMS

Areas of Review

At MSRs, 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 MSRs, 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.

In a liquid-fueled reactor, gaseous fission products are released directly to the reactor fuel. This gas must be trapped and directed to a radioactive waste treatment system. Review of the HVAC system must consider any intrusion of this highly radioactive waste stream into occupied areas. The review of systems with this potential should be covered in their review as is discussed in depth in Chapter 5, “Molten Salt Reactor Cooling Systems.”

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 (TS) and their bases, including testing and surveillance.

Acceptance Criteria

The acceptance criteria for 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 TS 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 ("Molten Salt 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 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 in 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 TS, 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.

For some MSRs, the handling of reactor fuel is a continuous operation that is more significant during operation than during a shutdown.

Areas of Review

The reviewer should evaluate the handling, protection, and storage of the reactor fuel when it is not in the vessel, 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 reactor 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.
- Systems and methods for storage as the fuel is made ready for use and when ready for use, any purifications systems required, and how it is blended into a useable fuel salt.
- Methods, analyses, and systems for secure storage of new and irradiated reactor fuel that will prevent criticality (keff 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.)

- Systems and methods for the addition of fuel or fuel salt to the vessel and for the removal of fuel salt from the vessel during reactor operation. (This discussion should include physical and administrative methods to ensure that fuel and fuel salt is handled only by authorized persons.)
- Systems and methods for the addition of fuel salt to the vessel and for the removal of fuel salt from the vessel during shutdown. (This discussion should include physical and administrative methods to ensure that fuel and fuel salt is handled only by authorized persons.)
- Systems, components, and methods for radiation shielding and for protecting irradiated reactor fuel during removal from the vessel, movement within the reactor facility, and storage.
- Systems, components, and methods used to prepare and ship reactor 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.)
- Nuclear and chemical stability of materials subject to long-term exposure to irradiated fuel stored on-site (e.g., keff changes resulting from a heat-induced recombination of gas in a highly irradiated liquid fuel storage tank). For reference, an Oak Ridge National Laboratory conference report documents a criticality concern that occurred in the Molten Salt Reactor Experiment after long-term storage of the solidified fuel salt when the fuel was heated to promote recombination of fluorine in the fuel salt drain tank. Over time, it was discovered that significant amounts of uranium and fluorine had migrated out of the drain tanks and into the off-gas system creating a criticality concern in the off-gas system.
- TS that define controls on fuel during handling and storage, including testing and surveillance.

Acceptance Criteria

The acceptance criteria for 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 active reactor core should ensure that the neutron multiplication, keff, will not exceed 0.90 under any possible conditions using conservative assumptions. (Existing usage with keff greater than 0.90 will be acceptable if the usage was previously reviewed and approved by the 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 the addition of fuel or fuel salt to the vessel and for the removal of fuel salt from the vessel, should be designed to prevent release of fission products.
- 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 as discussed in Chapter 12, “Conduct of Operations,” Section 12, “Material Control and Accounting Plan.” 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.
- The TS should contain limitations on the 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 reactor 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 system boundary 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 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. 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).
- Methods for assessing irradiated reactor fuel radioactivity and potential exposure rates are adequate to avoid overexposure of the staff.
- Methods for shielding, cooling, and storing irradiated reactor fuel give reasonable assurance of the following:

    a. Potential personnel doses will not exceed the regulatory limits of 10 CFR Part 20 and are consistent with the facility ALARA program.
    b. Irradiated reactor fuel can be cooled as necessary to avoid loss of integrity and corrosive deterioration during both moving and storage within the facility.

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.
- Fire protection equipment that uses water should consider the following MSR issues:

    a. Fire suppression with water in areas with high temperature components creates a potential for rapid pressurization (i.e., steam hazard).
    b. Water always creates the potential for a criticality excursion around fissile material.
    c. A properly written fire hazards assessment contains a discussion on the control of run off from fire suppression. In the case of MSRs, water is probably the most efficient transport mechanism for fuel salts.

- List of the objectives of the fire protection program, as well as 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 a summary of the more detailed discussions of these personnel and off-site 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 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.
- Fire protection equipment that uses water should state how the following MSR issues are addressed:

    a. Fire suppression with water in areas with high temperature components creates a potential for rapid pressurization (i.e., steam hazard).
    b. Water always creates the potential for a criticality excursion around fissile material.
    c. A properly written fire hazards assessment contains a discussion on the control of run off from fire suppression. In the case of MSRs, water is probably the most efficient transport mechanism for fuel salts.

- The facility should be designed, and protective systems should exist, to ensure safe reactor shutdown and prevent uncontrolled release of radioactive material if a fire were to 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; the specific use of water to suppress and extinguish MSR fires; passive features designed into the facility to limit fire consequences; response organization training and availability to fight fires as detailed in the emergency plan; designs of reactor systems that can ensure safe reactor shutdown in the event of fire; and potential radiological consequences to the public, staff, and environment if 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 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. This includes consideration of the following for MSRs:

    a. Fire suppression with water in areas with high temperature components creates a potential for rapid pressurization (i.e., steam hazard).
    b. Water always creates the potential for a criticality excursion around fissile material.
    c. A properly written FHA contains a discussion on the control of run off from fire suppression. In the case of MSRs, water is probably the most efficient transport mechanism for fuel salts.

- 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 TS related to fire protection have been proposed and justified (if applicable). These TS include acceptable requirements for testing and surveillance to ensure operability of fire detection and protection equipment.

9.4 COMMUNICATION SYSTEMS

Areas of Review

The reviewer should include the following:

- Methods of communication between all necessary locations during the full range of reactor operations.
- Summary of emergency communications that are discussed in detail in the physical security and emergency plans and evaluated by the staff.
- Method for providing two-way communication between the reactor control room and other locations in the reactor facility.

Acceptance Criteria

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

- The communication system should allow the reactor operator on duty to contact the supervisor on duty, health physics staff, and other personnel required by the TS any time the reactor is operating.
- The communication system should allow the operator, or other designated staff member, to announce the existence of an emergency to 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 between 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 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 TS 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 an MSR 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 may also 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.

For a liquid-fueled reactor, the quantities and concentrations of byproduct, source, and special nuclear material is an integral part of fuel management.

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 TS 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 TS covering these auxiliary facilities should ensure 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;" 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 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*).
- TS 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 GAS MANAGEMENT SYSTEM

Areas of Review

The active reactor core of an MSR is a solution of fuel salt within a gas-tight vessel. Therefore, a gas management system is very important for MSR designs since the fission gas produced during operation is released directly into the reactor fuel. Section 4.7, "Gas Management System," covers flow of evolved gases and fission products from their generation in the vessel to their ultimate release. This section of the SAR should contain the design information on control of the cover gas interface with the fuel salt and all decay heat removal components in the gas management system. The design information should be presented in drawings, diagrams, text, and analysis in sufficient detail for the staff to understand system cooling. Using this information, the staff should determine whether there is reasonable assurance that the gas management system has adequate cooling resources to maintain fuel system boundary integrity.

In evaluating the analysis demonstrating these capabilities, the staff should ensure that these criteria can be met for the maximum power density that is considered credible during power oscillations. The applicant should justify the maximum fission product generation rates during power oscillations.

Areas of review should include the following:

- Design bases for the closed systems, addressing the types of gases to be contained, cooled, and controlled in them.
- Systems for assessing and maintaining any required pressure differential between the external atmosphere and the fuel system boundary.
- Systems for assessing the required purity or concentrations of the contained gases.
- Systems for removing heat from the gas management system.
- Methods and systems for circulating, processing, decontaminating, recovering, and storing the contained gases.
- 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 TS and their bases, including testing and surveillance, required to ensure operability of the gas management systems, if applicable.

Acceptance Criteria

The acceptance criteria for information on the gas management system in MSRs include the following:

- The systems should be designed to perform the design-bases functions.
- The systems should be designed to remove decay heat from the gas management system. The analysis should show that the reactor would not be damaged and fuel system boundary integrity would not be lost.
- 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.
- TS 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, cool, and control the gas management system for MSRs. 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 safety evaluation report:

- The reactor is designed to operate with a gas-tight vessel, and the design of the gas management system helps provide that function. The gas management system is designed to prevent the uncontrolled release of radioactive material and interference with safe reactor operation or shutdown.
- The discussion of decay heat removal from the gas management system shows that a credible failure would not lead to loss of fuel system boundary integrity. Therefore, the information demonstrates that adequate heat removal mechanisms are available to the gas management system.
- The gas management system is designed to ensure that the required type of gas, the acceptable concentrations of constituents (including processing, storing, and recombining of reactive gases, as applicable), and the design-basis pressure are maintained.
- TS and their bases that are necessary to protect the health and safety of the public and safe reactor operation have been provided.

9.7 COOLING SYSTEMS

Among the auxiliary systems that should be addressed are any cooling systems that are part of the licensed facility. Chapter 5, "Molten Salt Reactor Cooling Systems," identifies the following cooling systems that might be associated with a liquid fueled MSR:

- Fuel salt drain tank
- Primary cooling system drain tank, if applicable
- Gas management system cooling
- Cooling for chemical processing/polishing loop
- Other cooling systems

For each type of cooling system that is part of the facility being licensed, the following review procedures should be followed:

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 auxiliary cooling systems include the following:

- The design and functional description of the auxiliary cooling system should ensure that it conforms to the design bases.

- The design, functions, and potential malfunctions of the auxiliary cooling 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 cooling 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 cooling system should interfere with or prevent safe shutdown of the reactor.
- The TS and bases applicable to an auxiliary cooling system should be provided.

Review Procedures

The reviewer should compare the design and functional descriptions of the auxiliary cooling system 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 cooling 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 safety evaluation report:

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

9.8 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 MSRs 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 and include the following:

- Demonstrate that the auxiliary system will function under analyzed reactor accident conditions, if required.
- Demonstrate that the auxiliary system and any malfunction could not create conditions or events that could cause an unanalyzed reactor accident or the uncontrolled release of radioactive material beyond those analyzed in Chapter 13 of the SAR.
- Demonstrate that the auxiliary system could not prevent safe reactor shutdown.

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 to 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 TS 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 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 TS and their bases proposed in the SAR give reasonable assurance that the system will be operable, as required by the design bases.

9.9 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 8.24, “Validation of Neutron Transport Methods for Nuclear Criticality Safety,” ANS, LaGrange Park, Illinois, 2017.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, “The Development of Technical Specifications for Research Reactors,” ANS, LaGrange Park, Illinois, 2013.

F. J. Peretz, et al, ORNL/CP-98146, “Removal of Uranium and Salt from the Molten Salt Reactor Experiment,” Oak Ridge National Laboratory, 1998.

U.S. Nuclear Regulatory Commission, NUREG-0849, “Standard Review Plan for Fuel Cycle Facilities License Applications,” Rev. 2, June 2015. (Adams Accession No. ML15176A258)

10 EXPERIMENTAL FACILITIES AND UTILIZATION

Chapter 10 of this guide is applicable to reviewing any experimental facilities associated with a non-power MSR. This chapter of the SAR contains guidance for evaluating the information on the experimental facilities at the MSR, 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 Guide 2.2. The reviewer should be familiar with this document.

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

In addition to traditional experimental purposes, as described above, a non-power MSR may be used to gather information and data that could be useful for the purposes of licensing future prototype facilities and power reactors. In this case, specific experimental facilities may not be included as part of the MSR. However, new or unique SSC could be described by the applicant to be demonstrated by the MSR. The applicant should describe any special safety features and added instrumentation for these SSC in this chapter of the SAR. QA programs for new or unique SSC being demonstrated by the MSR should be referenced here and discussed in Chapter 12, "Conduct of Operations."

In addition, the reactor itself can be considered an experimental facility to demonstrate MSR technology for eventual prototype and commercial scale up. When the reactor is the experiment, the applicant should include safety analysis and reactor technical specifications that demonstrate appropriate operational limitations. The reviewer will find that this section of the SAR and the experimental technical specifications will focus on reactor physics constraints for operating entire reactors as experiments.

Utility, integrity, longevity, versatility, diversity, and safety should be considered for the experimental facilities in the same manner they are considered for the active 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 active reactor 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 active reactor 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.

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 or the reactor vessel, this chapter of the SAR should contain an analysis of the structural design and potential impact on reactor operation. 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 to 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 (active reactor core, thermal column, external beam, etc.)
- Limiting experimental characteristics (e.g., reactivity, contents)
- Monitoring and control of the experiments and the interactions 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 drawing 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 independent 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 coolant leakage, fuel salt 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.
- Active reactor core/in-reflector experimental facilities, including important design and operating parameters and specifications: location of the facility in relation to the active reactor core, safety systems, active reactor core support, neutron detectors, fuel salt components, coolant system components; and reactivity effects of these facilities.
- Features of the experimental facility that prevent interference with safe reactor shutdown or with adequate reactor cooling.
- Radiological considerations associated with the design and use of the experimental facilities: generation of radioactive gases, release of fission products or other radioactive contaminants, and exposure of personnel to neutron and gamma beams, in relations to any facility-related technical specifications; direct radiation streams from 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 interference between this system and the reactor protecting 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 fuel salt or primary cooling system. The reviewer should refer to Chapter 5, "Molten Salt Reactor Cooling 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 salt temperature, fuel salt pressure, and fuel salt chemistry.

- 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 the fuel system boundary, 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 does 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. Equipment to provide for remote access may be necessary. 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.
- 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 cooling systems should not compromise adequate fuel salt cooling or prevent safe reactor shutdown.

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

- The applicant should show that fuel system boundary integrity would not be lost as a result of an experimental facility boundary breach or affected by the design of an experimental facility or by any other 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. 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 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 state 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 Guide 2.2 is 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 Guide 2.2 was considered should be discussed. The assessment of experimental materials should include the following:

    a. Radioactive materials, including fissile materials, and radiological risks from radiation fields or release of radioactive material
    b. Neutron absorption and activation in trace elements and impurities

c. Effects on reactivity, both positive and negative
d. Explosive, corrosive, and highly reactive chemicals
e. Radiation-sensitive materials
f. Flammable or toxic materials
g. Unknown materials
h. 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 REFERENCE

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 2007 (R2013).

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

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

Chapter 11 of this guide is applicable to reviewing a description of radiation protection and the radioactive waste management programs for the licensing of a non-power MSR. 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 MSR. 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 and to the public as low as is reasonably achievable (ALARA). The responsibilities of the radiation protection organization at the reactor facility and of any other onsite radiation safety 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 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.

Both the radiation protection program and waste management for the MSR has significant differences from conventional light water reactors. In an MSR, fission products are released to the liquid salt fuel solution and contained by the fuel barrier. The gaseous or volatile fission products will be gathered and processed within the radionuclide barrier. The fission gas may require holdup for decay or further treatment (such as filtering with activated charcoal) before being released to the environment or disposed of as waste. Residue from the cleanup and polishing of the liquid fuel will be laden with fission products and will likely require treatment as radioactive waste.

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 and associated hazards. 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 the 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, alpha, beta, and gamma), radioactivity (curie or becquerel content) 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 in both restricted, controlled (if present), and unrestricted areas. The applicant should identify (1) locations of specific sources (e.g., fuel salt, coolant salt, 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 total effective dose equivalent (TEDE) 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 dose 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 in curie or becquerel 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 and material transfer data can be reviewed to verify quantities and general locations of special nuclear material (SNM). The reviewer should examine the fuel management program and record keeping of fissile and fissionable material in the MSR cycle and the radiation and radioactive sources inherent in the fuel cycle.

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., fuel salt in the active reactor core and vessel), predicted exposure, access control, provisions for source control and storage, and interim or ultimate disposition.

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 description of potential radiation sources and associated doses and dose rates 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
- The radiation protection program should clearly distinguish between trained radiation workers, who may receive specified occupational dose during an accident, and members of the public, whose consequences and likelihoods should be controlled to more stringent levels.
- The application should identify a controlled area, as defined in 10 CFR 20.1003. This controlled area should be identified in the boundary and area maps provided in Chapter 2, Section 2.1.1.2, of the SAR.

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 radiation protection program should provide for compliance with all applicable regulations.
- The radiation protection program should show clearly that all review areas (discussed above) have been addressed and all expected radiation doses have been addressed.
- The radiation protection 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.
- The radiation protection program should clearly distinguish between trained radiation workers, who may receive specified occupational dose during an accident, and members of the public, whose consequences and likelihoods should be controlled to more stringent levels. The application should identify a controlled area, as defined in 10 CFR 20.1003. This controlled area should be identified in the boundary and area maps provided in Chapter 2, Section 2.1.1.2, of the SAR. The licensee must retain the authority to exclude or remove personnel and property from the area. For the purpose of demonstrating that the operations of the facility meet the criterion of the radiation protection program, individuals who are not workers, may be permitted to perform ongoing activities (e.g., at a facility not related to the licensed activities) in the controlled area, if the licensee:

    (1) Demonstrates and documents, in the radiation protection program, that the risk for those individuals at the location of their activities does not exceed the performance requirements; or
    (2) Provides training that satisfies 10 CFR 19.12(a)(1)–(5) to these individuals and ensures that they are aware of the risks associated with accidents involving the licensed activities as determined by the radiation protection program, and conspicuously posts and maintains notices stating where the information in 10 CFR 19.11(a) may be examined by these individuals.

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 review's 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.
- The staff's evaluation should verify that the license application contains a clear definition of the controlled area, and that the radiation protection program performance requirements clearly distinguish between workers inside the controlled area and members of the public outside the controlled area.

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 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 l0 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 examples:

  a. Continuous air monitors (CAMs), including fixed and moving filter CAMs, and gaseous monitors
  b. Portable radiation survey instruments
  c. Remote area radiation monitors (RAMs)
  d. Effluent radiation monitors
  e. Samplers
  f. Environmental radiation monitors
  g. Contamination-monitoring equipment
  h. Personal dosimeters
  i. Portal monitors
  j. Radwaste storage and release monitors
  k. Criticality detection 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 (TS).

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 radiobioassay 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 air packs) 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 bioassay 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 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
- Radiation 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 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 verity 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, and 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 several 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 “racked” 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:
    a. 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
    b. Provisions for planning both reactor utilization and operation activities to avoid or prevent uncontrolled occurrence and spread of radioactive contamination
    c. Provisions for routine monitoring and access control to identify radioactive contamination and to assess and limit personnel exposures
    d. The bases for TS 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 preoperational 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 TS 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 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 pre-construction or pre-operation 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 organization 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 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 TS

Acceptance Criteria

The 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. TS 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
  a. The applicant has developed the bases for a complete and effective program,
  b. The program includes review, audit, and assessment provisions, and
  c. The program complies with all applicable regulations.
- The applicant has described the waste management program in a manner showing that processes effectively remove undesired materials from the liquid fuel without providing a means that fissile material can be removed and collected in the process.
- 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 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. The reviewer should take particular care of the wording used in all documentation to distinguish between processes used by which waste materials are removed from the liquid fuel of the MSR and processes that might be construed as producing special nuclear material.

Acceptance Criteria

The acceptance criteria for information on radioactive waste controls include the following:

- The applicant should describe how all processes 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.
- Wording used in all licensing documents clearly distinguishes the difference between waste removal processes used to clean up the MSR liquid fuel and processes that may be considered production of special nuclear material.

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

The 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 RESPIRATORY PROTECTION PROGRAM

Areas of Review

The areas of review should include detailed information about the following two areas of the respiratory program:

(1) Establishment, maintenance, and implementation of a respiratory protection program.
(2) Design and implementation of programs to control airborne concentrations of radioactive material by using ventilation systems, containment systems, and respirators.

Acceptance Criteria

The applicant should do the following:

- Install appropriately sized ventilation and containment systems in areas of the plant identified as having potential airborne concentrations of radionuclides that could exceed the occupational derived air concentration values specified in 10 CFR Part 20, "Standards for Protection against Radiation," Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."
- Describe surveillance requirements, including preventive and corrective maintenance and performance testing, to ensure that the ventilation and containment systems operate when required and are within their design specifications.
- Describe the criteria for the ventilation and containment systems, including minimum flow velocity at openings in these systems, maximum differential pressure across filters, and types of filters to be used.

- Describe the frequency and types of tests to measure the performance of ventilation and containment systems, the acceptance criteria, and the actions to be taken when the acceptance criteria are not satisfied.
- Establish a respiratory protection program that meets the requirements of 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."
- Prepare written procedures for the selection, fitting, issuance, maintenance, testing, training of personnel, monitoring, and recordkeeping for individual respiratory protection equipment and for specifying when such equipment is to be used.
- Revise the written procedures for the use of individual respiratory protection equipment, as applicable, when making changes to processing, facility, or equipment.
- Maintain records of the respiratory protection program, including training in respirator use and maintenance.

Review Procedures

The reviewer should determine whether the respiratory protection program provides adequate protection of personnel from airborne concentrations exceeding the limits of Appendix B to 10 CFR Part 20 and the overall adequacy of the program. The methods used for the identification and evaluation of potential hazards and estimated doses should provide realistic and accurate predictions. The applicant should evaluate potential hazards and estimated doses by performing surveys, bioassays, air sampling, or other means as necessary.

As for the respiratory protection to be used, the reviewer should ensure that the equipment has been tested and certified to provide the appropriate degree of personal protection. The applicant must also commit to testing of respirators for operability before usage. The reviewer should also examine the description of respirator usage, training, fit testing, selection, storage, maintenance, repair, and quality assurance through the written procedures.

After evaluating the acceptance criteria, the reviewer will perform a safety evaluation. The reviewer will prepare a safety evaluation report (SER) on the licensing action for the licensing project manager.

Evaluation Findings

The reviewer will draft an SER addressing the topic reviewed explaining why the NRC staff has reasonable assurance that the respiratory protection program is acceptable and that the health and safety of the workers is adequately protected. The NRC staff may propose license conditions to impose requirements in those areas in which the application is deficient. The NRC staff's SER should include the following kind of statement and conclusion:

The applicant has committed to an acceptable radiation protection program that includes a program to control airborne concentrations of radioactive material with engineering controls and respiratory protection.

11.4 REFERENCES

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities," ANS, LaGrange Park, Illinois, 2016.

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. (withdrawn)

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

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

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

US Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents."

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

US Nuclear Regulatory Commission, Regulatory Guide 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

US Nuclear Regulatory Commission, Regulatory Guide 8.10, Revision 2, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."

US Nuclear Regulatory Commission, Regulatory Guide 8.13, Revision 3, "Instructions Concerning Prenatal Radiation Exposure."

US Nuclear Regulatory Commission, Regulatory Guide 8.29, Revision 1, "Instruction Concerning Risks from Occupational Radiation Exposure."

US Nuclear Regulatory Commission, Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities."

12 CONDUCT OF OPERATIONS

Chapter 12 of this guide is applicable to reviewing conduct of operations for a non-power MSR. In this chapter of the SAR, the reviewer should verify that the applicant describes and discusses the conduct of operations at any MSR captured in the scope of this guidance. The conduct of operations involves administrative aspects of facility operations, 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 producers, 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 across various reactor technologies. 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 utility, research group or person (e.g., university provost or dean) with legal responsibility for the reactor license (the licensee) at the top of the organizational and 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 organizational 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 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 Institute/American Nuclear Society

(ANSI/ANS) 15.4, “Selection and Training of Personnel for Research Reactors,” contains additional guidance on 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 the training to meet 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 organization 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, “The Development of Technical Specifications for Research Reactors, and ANSI/ANS 15.4.
- 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 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 no 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 and 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, with the addition of plans such as 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 guidance in ANSI/ANS 15.1 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 MSR organization. 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 review 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 the 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 of making substantive, minor, and temporary changes to procedures

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 MSR facilities includes the following:

- The applicant should propose a minimum list of procedural topics as given in ANSI/ANS 15.1. 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 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 procedures 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 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 and Chapter 14 of the format and content guide. The review 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 the applicant will ensure proper management control and proper review 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, specifications 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.

Acceptance Criteria

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

- The applicant should discuss the events that are defined as reportable. The applicant should consider the guidance in ANSI/ANS 15.1 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 and Chapter 14 of the format and content guide.

Review Procedures

The review should compare the information the guidance in ANSI/ANS 15.1 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 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 of 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 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 limit 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 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 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

Area of Review

The reviewer should examine facility records and review the records to be retained and 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 reviewers should compare the information with the guidance in ANSI/ANS 15.1 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 Division of Preparedness and Response, Reactor Licensing Branch in the Office of Nuclear Security and Incident Response (NSIR) for review. For the review and evaluation of emergency plans at non-power reactors, the emergency plan reviewer should use NUREG-0849 and ANSI/ANS 15.16, "Emergency Planning for Research Reactors," 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 Non-Power Production and Utilization Facility Oversight Branch in NRR for the review and evaluation of security plans at non-power reactors. Security planning information is considered either proprietary or safeguards information (including Safeguards Information designated as Safeguards Information-Modified Handling) and must receive special handling. The technical reviewer should ensure that this information is handled properly. The security plan reviewer should contact the responsible branch in the Office of Nuclear Security and Incident Response 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 remind the quality assurance reviewer that this information is for a non-power reactor and that Regulator Guide 2.5 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 REACTOR OPERATOR TRAINING AND REQUALIFICATION

Areas of Review

The applicant should discuss the initial reactor operator or senior reactor operator qualification program. In addition, 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 retaining 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 and senior reactor operator qualification plans include the following:

- The written examination for an operator will contain a representative selection of questions on the knowledge, skills, and abilities needed to perform licensed operator duties as required by 10 CFR 55.41 and 10 CFR 55.43.
- 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.
- 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 application before testing begins.
- Administrative requirements are specified such that each reactor operator and senior reactor operator maintain the conditions on their license as required by 10 CFR 55.53 and ANSI/ANS 15.4.

The acceptance criteria for the information on reactor operator and senior 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.
- Planned lectures should be given on a regular and containing 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 knowns 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 were 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 application 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 application 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 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 requirements should be given in the plan or in the technical specifications.

Review Procedures

The reviewer should confirm that the information in the reactor operator and senior reactor operator qualification and 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 judgement and knowledge of non-power reactors to determine the degrees of conformity with the regulations.

Evaluation Findings

The reactor operator and senior reactor operator qualification and requalification plans should contain sufficient information to support the following type of conclusion, which will be included in the staff's safety evaluation report:

- The application has submitted a reactor operator and senior reactor operator qualification and requalification plans that contains the program for reactor operator and senior operator initial qualification, 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 ANSI/ANS 15.4 and are acceptable. The plan and procedures given reasonable assurance that the reactor facility will be operated by competent operators.

12.11 STARTUP PLAN

Areas of Review

The application 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 content 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:

- Startup operations involving fuel should be described as processes with homogenous fuel in liquid or solid form, as appropriate. 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 applicant should have plans to confirm that fuel salt chemistry is within the expected and analyzed limits.
- The critical mass should be approximately known and should be exactly determined by a systematic approach to a critical experiment. A systematic approach to melting solid fuel or loading liquid fuel into the vessel should be defined.
- 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 element worths, power level monitors, scram and interlock functions, fuel salt 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 element 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 measure 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 MATERIAL CONTROL AND ACCOUNTING PLAN

The reviewer should ensure that a complete plan is filed in accordance with the requirements of 10 CFR Part 74, “Material Control and Accounting of Special Nuclear Material.” This section should include the following subsections:

Areas of Review

The reviewer should verify that the application includes information regarding compliance with Subparts A through E of 10 CFR Part 74, as appropriate, for the class of facility involved. All general and class-specific information should be included as applicable to the following:

- General requirement for all facilities
- Facilities with quantities of low strategic significance
- Facilities with quantities of moderate strategic significance
- Facilities with formula quantities of strategic SNM

Acceptance Criteria

The review shall verify that the material control and accounting plan must contains all of the information prescribed in 10 CFR Part 74 for the specific class of facility contained in the application.

Review Procedures

The reviewer should ascertain that the plan contains a clear, accurate, and thorough account of all inventory, measurement, recordkeeping, and reporting requirements prescribed by the regulations.

Evaluation Findings

The reviewer should be able to conclude the following from the submitted plan:

- The applicant has provided a complete plan that will ensure that all SNM in the facility will be properly accounted for and that will enable the licensee to achieve the specific objectives prescribed in the regulations for the relevant class of material that will be possessed at the facility.

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, 2007 (R2013).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.4, "Selection and Training of Personnel for Research Reactors," ANS, La Grange Park, Illinois, 2016.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8, "Quality Assurance Program Requirements for Research Reactors," ANS, LaGrange Park, Illinois, 1995 (R2013).

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16, "Emergency Planning for Research Reactors," ANS, LaGrange Park, Illinois, 2015.

U.S. Nuclear Regulatory Commission, Regulator Guide 2.5, "Quality Assurance Program Requirements for Research Reactors and Test Reactors," Revision 1, 2010.

U.S. Nuclear Regulatory Commission, Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," Revision 2, 2017.

U.S. Nuclear Regulatory Commission, Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," Revision 1, 1983.

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, NUREG-1065, "Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Facilities," Revision 2, 1995.

U. S. Nuclear Regulatory Commission - NUREG/BR-0006, “Instructions for Completing Nuclear Material Transfer Reports (DOE/NRC Forms 741, 741A, and 740M),” Revision 8, May 2018 or Draft Rev. 9, July 2019.

U. S. Nuclear Regulatory Commission - NUREG/BR-0007, “Instructions for the Preparation and Distribution of Material Status Reports (DOE/NRC Forms 742, and 742C),” Revision 7, May 2018 or Draft Rev. 8, July 2019.

13 ACCIDENT ANALYSES

Chapter 13 of this guide is applicable to reviewing a description of the accident analyses for a non-power MSR. Other chapters of the SAR should contain discussions and analyses of the MSR 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 MSR applicant should identify all the sources, locations, quantities, and potential release paths for radioactive material. The MSR applicant should also identify and evaluate those chemical hazards that are under NRC's regulatory jurisdiction with the potential for significant consequences to workers or the public (see the Memorandum of Understanding between the U.S. Nuclear Regulatory Commission and the Occupational Safety and Health Administration (ADAMS Accession No. ML11354A432)). 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 (including MSRs), 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 fission products, or both, outside the vessel and does not need to be entirely credible. The MHA need not include an active reactor core scenario because other plant systems may provide the bounding accident conditions (e.g., the gas management system or the fuel handling system). 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.

The MHA is used to demonstrate that the maximum consequences of operating the reactor at a specific site are within acceptable limits. Therefore, an MHA is postulated that results in consequences bounding those of any credible accident likely to occur over the life of the facility. 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 non-power MSR format and content guide (Part 1) by demonstrating that the applicant has considered all credible accidents at the reactor facility and adequately evaluated their consequences. 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. For non-power MSRs, the following categories are applicable, but not limited to:

- MHA
- Reduction in fuel salt inventory from a barrier failure (includes rupture of the vessel, waste-handling tanks, the gas management system, the polishing system, pumps, valves, heat exchangers, and piping)
- Increase in fuel salt inventory
- Reduction in cooling
- Reactivity and power distribution anomalies
- Mishandling or malfunction of fuel (includes fuel salt composition changes)
- Experiment malfunction (if experiment capabilities are included)
- External events (includes natural hazards and manmade events)
- Mishandling or malfunction of equipment (i.e., stuck-open relief valve, dropping a heavy object, fires)
- Loss of normal electrical power

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 MSR accident analysis.

Areas of Review

Areas of review should include 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. The limiting event should be analyzed quantitatively. Chapter 13 in Part 1 of this guidance, the format and content guide, suggests the steps for the applicant to follow once the limiting event is determined for a category of accidents.

Acceptance Criteria and Dose Limits

The safety analysis must meet the requirements set forth in 10 CFR 50.34, "Contents of application; technical information." In particular, a construction permit application must include a safety analysis report as described in 10 CFR 50.34(a), "Preliminary safety analysis report;" an operating license application must include a safety analysis report as described in 10 CFR 50.34(b), "Final safety analysis report." For a research reactor, the results of the accident analysis are generally compared with 10-CFR Part 20 criteria (10 CFR 20.1001 through 20.2402 and appendices). 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. 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.

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.

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 we categorized, and the most limiting accident in each group was chosen for detailed analysis.
- The reactor was assumed to be operating normally under applicable technical specification before the initiating event. However, the reactor is 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 conservative uncertainty values should be applied to instrument readings 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 ESFs required to be operable by technical specifications.
- 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 radiological consequences to the facility staff and the public were presented and compared with acceptable limits.

The reviewer should confirm that the integrity of the fuel system boundary will be maintained under all credible accidents analyzed. The fuel system boundary consists of all structures and coatings that prevent the exposure of fuel, fission gas, or other fission products to the next credited barrier layer. For an MSR, the fuel system boundary includes the vessel, waste-handling tanks, drain tanks or fuel holding tanks, the gas management system, pumps, valves, heat exchangers, and piping. The applicant should provide information on the following fuel salt parameters for each credible accident identified:

- Temperature
- Pressure
- Flow
- Density
- Reactivity
- Radiological inventory
- Chemical/physical property changes

The reviewer should determine whether the applicant has categorized and analyzed all credible accidents, in terms of the limiting phenomena identified below, that could pose a challenge to the integrity of the fuel system boundary. Limiting phenomena refer to those physical phenomena that could occur during the course of a transient or accident that significantly affects the subsequent likelihood of failure of the fuel system boundary. The accident analyses should credit subsequent release barriers in different locations in the facility. The reviewer should also determine whether the applicant has identified the limiting sources and amounts of radionuclides that could be released within the facility or to the outside environment, thereby exposing the facility staff or the general public to radiation. Limiting phenomena for MSRs may include (but are not limited to):

- Precipitation of fission products

- Precipitation of fuel (uranium, thorium, etc.)
- Fuel salt chemistry including redox control and fission product polishing
- Reactor flow and the location of delayed neutron production
- Active reactor core voiding due to gaseous fission product formation
- Criticality accidents
- Tritium production (depends on salt selection)

The limiting phenomena are analogous to phenomena for heterogeneous non-power reactors that set operational and safety limits. For example, departure from nucleate boiling has been identified as a phenomenon that greatly increases the likelihood of cladding failure in light-water reactors and is useful in deriving quantitative operating and safety limits. Additionally, regulatory limits such as fuel cladding oxidation are tied to the phenomenon of loss of cladding ductility. While the specific safety limits will depend on the reactor design, the reviewer should confirm that the applicant has addressed these limiting phenomena in its definitions of operating and safety limits as well as accident analyses.

Reactivity limits and the functional designs of control and safety-related systems should limit and mitigate loss of barrier integrity during credible accidents involving insertion of some fraction of excess reactivity. The analysis should include applicable reactivity feedback coefficients and automatic protective actions. If applicable, the amount of reactivity allowed for moveable or unobscured experiments should be analyzed.

The reviewer should confirm that loss of normal electrical power and subsequent reduction in cooling will not lead to a challenge to the release barriers. A loss of normal electric power should not compromise safe reactor shutdown.

Evaluation Findings:

It is essential that all credible accidents at a non-power MSR be considered and evaluated during the design stage. Experience has indicated 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 MSRs are designed to operate with fuel salt pressure close to ambient, the margin for safety is 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 eleven major accident categories are presented below. These findings are examples only. The reviewer should modify the actual wording of the findings for each accident scenario to address the situation under review.

13.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

This section of the SAR should contain sufficient information to support the types of conclusions given below. These conclusions will be included in the staff’s safety evaluation report. The appropriate value for the reactor under evaluation should replace the notations “xx” and “yy.” The reviewer should modify these conclusions to conform to the reactor design under consideration.

13.1.1 Maximum Hypothetical Accident

The reviewer should determine whether the following finding is applicable:

The applicant has considered the consequences to the public of all credible accidents at the reactor facility. An MHA, which is an accident that would release fission products past an initial MSR system boundary and would have consequences greater than any credible accident, has been analyzed. The consequences of the MHA scenario bound the consequences of all credible events; however, the MHA need not be a credible accident. [*The MHA is specific to the reactor design and power. The reviewer may have to evaluate an MHA that differs from the suggested list of MHAs below*.]

Possible MHAs for an MSR could be one or a combination of the following events:

- Complete loss of reactor fuel salt inventory (*e.g.*, vessel rupture or drain tank rupture)
- Complete loss of the gas management system inventory (*e.g.*, system rupture)
- Release of used fuel constituents to the environment (*e.g.*, waste holding tank rupture)
- Energetic dispersal of the contents from the fuel system boundary with bypass of any scrubbing or delay capacity (*e.g.*, a guard vessel, other credited release barriers).
- External hazards including seismic, flooding, high winds, missile impact, aircraft impact, and fires
- Excessive tritium production

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 by the various design barriers 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 duration of the accident considered for the calculation of doses for the facility staff is (*xx*) and for the public it is (*yy*)."

The calculated total effective dose equivalent (TEDE) for the MHA scenario are the following:

- Licensee staff – *xx* mrem
- Maximum exposed member of the public – *yy* mrem
- Nearest residence – *xx* mrem

These doses and dose commitments are within the acceptable limits [*state the limits*]. Because the assumptions of the scenario are conservative, the doses calculated will likely not be exceeded by any accident considered credible. 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.

13.1.2 Reduction in Fuel Salt Inventory

The reviewer should determine whether the following finding is applicable:

The applicant has evaluated the consequences of potential reduction in fuel salt inventory within the fuel system boundary. The applicant has discussed possible methods by which fuel salt could be leaked from the fuel system boundary. The following items have been evaluated: valve leakage; pipe leakage; and inter-system leakage. The applicant has also evaluated the means of detecting fuel salt leakage. The MHA bounds the consequences of the limiting loss of fuel salt inventory from the fuel system boundary. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13.1.3 Increase in Fuel Salt Inventory

The reviewer should determine whether the following finding is applicable:

The applicant has evaluated the consequences of potential increases in fuel salt inventory within the fuel system boundary. The applicant has discussed possible methods by which fuel salt inventory could be increased within the fuel system boundary. The following items have been evaluated: systematic overfill, primary heat exchanger leakage; fuel and salt makeup tank leakage; and other inter-system leakage. The applicant has also evaluated the means of detecting fuel salt inventory increases. The MHA bounds the consequences of the limiting increase of fuel salt inventory within the fuel system boundary. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13.1.4 Reduction in Cooling

The reviewer should determine if the following finding is applicable:

The applicant has considered postulated events that lead to reduction or loss of cooling. The following initiators have been analyzed:

- Loss of electrical power
- Loss of forced circulation
- Failure of active components in the primary cooling system or the normal heat dissipation system (loss of heat sink)
- Primary heat exchanger tube rupture
- Flow obstruction in primary heat exchangers
- Full or partial freezing in system piping
- Loss of cooling to the gas management system
- Loss of cooling to the drain tank (if applicable)

The consequences of reduction in cooling events have been analyzed and shown to be bounded by the MHA. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13.1.5 Reactivity and Power Distribution Anomalies

The reviewer should determine if the following finding is applicable:

The applicant has considered the following initiators that could cause reactivity or power distribution anomalies in the active reactor core:

- Pressurization of the fuel salt
- Excessive active reactor core voiding
- Excessive cooling
- Fuel or fuel salt injection
- Interfacing systems

- Active reactor core geometry changes
- Reflector or moderator changes
- Loss of forced circulation
- Loss of electrical power
- Spurious control element actuation
- Misaligned control elements, if applicable
- Reactivity or power anomaly effects of an experiment malfunction, if applicable

The consequences of reactivity and power distribution anomalies have been analyzed and shown to be bounded by the MHA. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13.1.6 Mishandling or Malfunction of Fuel

The reviewer should determine if the following finding is applicable:

The applicant has considered the consequences of fuel salt mishandling events, such as excessive leakage or spillage that could potentially initiate an unintended criticality event in an area or location where it could pose a threat to facility staff. This includes fuel salt in the vessel, in storage, or in between the vessel and the storage area and fissionable material that has not been mixed with salt. The MHA bounds the accident dose consequences of such postulated events. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.
The applicant considered the consequences of fuel salt malfunction or fuel salt composition change events for an MSR. These events include failure to control pH, temperature, or pressure of the fuel salt, which can impact the physical or chemical form of the fuel or salt resulting in adverse chemical effects, such as fuel precipitation or excessive corrosion. The MHA bounds the accidental dose consequences of such postulated events. Therefore, doses to the staff and the public will be within acceptable limits, and the health and safety of the staff and public will be adequately protected.

13.1.7 Experiment Malfunction

The reviewer should determine if the following finding is applicable:

The applicant has discussed the types of experiments (if applicable) that could be performed at the non-power MSR within its license and technical specifications. The discussions include events that could initiate accidents such as:

- loss of cooling or other malfunction in a fueled experiment resulting in liquification or volatilization of the fissile component
- loss of cooling capability in a strongly absorbing non-fueled experiment resulting in absorber failure and rapid increase in reactivity
- placement of an experiment component in an unplanned location, causing effects that were not evaluated
- failure of an experiment containing highly reactive contents
- failure of an experiment and release of corrosive materials in the reactor coolant
- detonation of an explosive experiment
- failure of an experiment that affects fuel salt flow, including localized fuel salt solidification due to overcooling by the failed experiment

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.

13.1.8 External Events

The reviewer should determine if the following finding is applicable:

Chapters 2 and 3 of the SAR discuss the design of the MSR facility and its ability to withstand external events and the potential associated accidents. The MSR 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, fire, flooding, or wind events that damage the reactor facility), the damage to equipment or release barriers 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.

13.1.9 Mishandling or Malfunction of Equipment

Initiating events under this heading would require a case-by-case, reactor-specific discussion. The applicant should consider the consequences of mishandling or malfunction of equipment that could result in the spillage or leakage of contaminated fluids, including inter-connected facilities and systems. If the SAR discusses additional events that fall outside the previous categories, the potential consequences should be compared with similar events already analyzed or with the MHA, as applicable.

13.1.10 Loss of Normal Electrical Power

The reviewer should determine if the following finding is applicable:

The applicant has discussed accident initiators that could result from onsite or offsite power interruptions. Emergency power supplies, if provided, are assumed to operate. Loss of normal electrical power effects on decay heat removal is discussed in Subsection 13.1.4

13.2 REFERENCES

13.2.1 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 Factor 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, pp. 419-523, 1973.

13.2.2 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, p. 319, 1981.

U.S. Nuclear Regulatory Commission, “Atmosphere 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, “Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities,” Regulatory Guide 2.6, September 2017.

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

13.2.3 Molten Salt Reactors

Beall, S. E., et al., “MSRE Design and Operations Report: Part 5, Reactor Safety Analysis Report,” ORNL-TM-732, August 1964.

Bettis, E. S., et al., “Design Studies of a Molten-Salt Reactor Demonstration Plant,” ORNL-TM-3832, June 1972.

Holcomb, D. E., et al., “Fast Spectrum Molten Salt Reactor Options,” ORNL/TM-2011/105, July 2011.

Muhlheim, M. D., “Identification of Initiating Events for aSMRs,” ORNL/TM-2013/513, June 2013.

Robertson, R. C., et al., “Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor,” ORNL-4541, June 1971.

14 TECHNICAL SPECIFICATIONS

Chapter 14 of this guide is applicable to reviewing the technical specifications for a non-power MSR. The format for the acceptance criteria follows the format of American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1. 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 nonpower MSRs (Part 1).
- 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, "The Development of Technical Specifications for Research Reactors," and Part 1 of the Non-Power MSR Guidance, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power MSRs: Format and Content."
- 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.

14.1 REFERENCE

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.1, "The Development of Technical Specifications for Research Reactors," ANS, LaGrange Park, Illinois, 2007 (R2013).

16 OTHER LICENSE CONSIDERATIONS

Chapter 16 of this guide is applicable to reviewing other license considerations for a non-power MSR. 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. Other topics should be determined by the applicant on a case-by-case basis and added as separate sections.

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 or fuel salt 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 salt, the fuel system boundary, 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.