United States
Nuclear Regulatory Commission

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

Standard Review Plan and Acceptance Criteria

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Office of Nuclear Reactor Regulation
Division of Reactor Program Managment

4 REACTOR DESCRIPTION

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

4.1 Summary Description

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

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

4.2 Reactor Core

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

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

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

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

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

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

4.2.1 Reactor Fuel

Areas of Review

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

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

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

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

Acceptance Criteria

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

- The design bases for the fuel should be clearly presented, and the design considerations and functional description should ensure that fuel conforms with the bases. Maintaining fuel integrity should be the most important design objective.

- The chemical, physical, and metallurgical characteristics of the fuel constituents should be chosen for compatibility with each other and the anticipated environment.

- Fuel enrichment should be consistent with the requirements of 10 CFR 50.64.

- The fuel design should take into account characteristics that could limit fuel integrity, such as heat capacity and conductivity; melting, softening, and blistering temperatures; corrosion and erosion caused by coolant; physical stresses from mechanical or hydraulic forces (internal pressures and Bernoulli forces); fuel burnup; radiation damage to the fuel and the fuel cladding or containment; and retention of fission products.

- The fuel design should include the nuclear features of the reactor core, such as structural materials with small neutron absorption cross-sections and minimum impurities, neutron reflectors, and burnable poisons, if used.

- The discussion of the fuel should include a summary of the fuel development and qualification program.

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

Review Procedures

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

Evaluation Findings

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

- The applicant has described in detail the fuel elements to be used in the reactor. The discussion includes the design limits of the fuel elements and clearly gives the technological and safety-related bases for these limits.

- The applicant has discussed the constituents, materials, components, and fabrication specifications for the fuel elements. Compliance with these specifications for all fuel acquisitions will ensure uniform characteristics and compliance with design bases and safety-related requirements.

- The applicant has referred to the fuel development program under which all fuel characteristics and parameters that are important to the safe operation of the reactor were investigated. The design limits are clearly identified for use in design bases to support technical specifications.

- Information on the design and development program for this fuel offers reasonable assurance that the fabricated fuel can function safely in the reactor without adversely affecting the health and safety of the public.

4.2.2 Control Rods

Areas of Review

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

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

Acceptance Criteria

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

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

- The control rod design should ensure that positioning is reproducible and that a readout of positions is available for all reactor operating conditions.

- The drive and control systems for each control rod should be independent from other rods to prevent a malfunction in one from affecting insertion or withdrawal of any other.

- The drive speeds and scram times of the control rods should be consistent with reactor kinetics requirements considering mechanical friction, hydraulic resistance, and the electrical or magnetic system

- The control rods should allow replacement and inspection, as required by operational requirements and the technical specifications.

- Technical specifications should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects and proposes limiting conditions for operations and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

- The applicant has described the control and safety rod systems for the reactor and included a discussion of the design bases, which are derived from the planned operational characteristics of the reactor. All functional and safety-related design bases can be achieved by the control rod designs.

- The applicant has included information on the materials, components, and fabrication specifications of the control rod systems. These descriptions offer reasonable assurance that the control rods conform with the design bases and can control and shut down the reactor safely from any operating condition.

- The staff has evaluated the information on scram design for the control rods and compared it with designs at other non-power reactors having

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

- (*For pulsing reactors*) The design and functional description of the transient rod system offer reasonable assurance that pulses will be reproducible and can be limited to values that maintain fuel integrity as determined by the thermal-hydraulic analyses

- The control rod design includes reactivity worths that can control the excess reactivity planned for the reactor, including ensuring an acceptable shutdown reactivity and margin, as defined and specified in the technical specifications.

- Changes in reactivity caused by control rod dynamic characteristics are acceptable. The staff evaluations included maximum scram times and maximum rates of insertion of positive reactivity for normal and ramp insertions caused by system malfunctions.

- The applicant has justified appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods and included them in the technical specifications.

4.2.3 Neutron Moderator and Reflector

Areas of Review

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

Areas of review should include the following:

- geometry

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

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

Acceptance Criteria

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

- The non-nuclear design bases such as reflector encapsulations should be clearly presented, and the nuclear bases should be briefly summarized. Non-nuclear design considerations should ensure that the moderator and reflector can provide the necessary nuclear functions.

- The design should ensure that the moderator and reflector are compatible with their chemical, thermal, mechanical, and radiation environments. The design specifications should include cladding, if necessary, to avoid direct contact with water or to control the escape of gases. If cladding used to avoid direct contact with reactor coolant should fail, the applicant should show that the reactor can continue to be operated safely until the cladding is repaired or replaced or should shut the reactor down until the cladding is repaired or replaced.

- The design should allow for dimensional changes from radiation damage and thermal expansion to avoid malfunctions of the moderator or reflector.

- The design should include experimental facilities that are an integral part of the reflector. If the facilities malfunction, the reflector components should neither damage other reactor core components nor prevent safe reactor shutdown.

- The design should provide for removal and/or replacement of solid moderator or reflector components and systems, if required by operational considerations.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects, and proposes limiting conditions for operations and surveillance requirements. The proposed technical specifications should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

- The moderator and reflector are integral constituents of a reactor core; the staff's evaluation of the nuclear features appears in Section 4 5. The designs take into account interactions between the moderator or reflector and the reactor environment. Reasonable assurance exists that degradation rates of the moderator or reflector will not affect safe reactor operation, prevent safe reactor shutdown, or cause uncontrolled release of radioactive material to the unrestricted environment.

- Graphite moderators or reflectors are clad in aluminum (*or state cladding material*) if they are located in an environment where coolant infiltration could cause changes in neutron scattering and absorption, thereby changing core reactivity. Reasonable assurance exists that leakage will not occur. In the unlikely event coolant infiltration occurs, the applicant has shown that this infiltration will not interfere with safe reactor operation or prevent safe reactor shutdown.

- The moderator or reflector is composed of chemically inert materials incorporated into a sound structure that can retain size and shape and support all projected physical forces and weights. Therefore, no unplanned changes to the moderator or reflector would occur that would interfere with safe reactor operation or prevent safe reactor shutdown.

- The applicant has justified appropriate design limits, limiting conditions for operation, and surveillance requirements for the moderator and reflector and included them in the technical specifications.

4.2.4 Neutron Startup Source

Areas of Review

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

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

Areas of review should include the following:

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

Acceptance Criteria

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

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

- If the source is regenerated by reactor operation, the design and analyses should demonstrate its capability to function as a reliable neutron startup source in the reactor environment.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

- The design of the neutron startup source is of a type (i.e., neutron-emitting reaction) that has been used reliably in similar reactors licensed by NRC (*or the design has been fully described and analyzed*) The staff concludes this type of source is acceptable for this reactor.

- The source will not degrade in the radiation environment during reactor operation. Either the levels of external radiation are not significant or the source will be retracted while the reactor is at high power to limit the exposure.

- Because of the source holder design and fabrication, reactor neutron absorption is low and radiation damage is negligible in the environment of use. When radiation heating occurs, the holder temperature does not increase significantly above the ambient water temperature.

- The source strength produces an acceptable count rate on the reactor startup instrumentation and allows for a monitored startup of the reactor under all operating conditions.

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

- The source and holder design operate safely and reliably.

4.2.5 Core Support Structure

Areas of Review

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

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

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

Acceptance Criteria

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

- The design should show that the core support structure will conservatively hold the weight of all core-related components with and without the buoyant forces of the water in the tank or pool.

- The design should show that the core support structure will conservatively withstand all hydraulic forces from anticipated coolant flow with negligible deflection or motion.

- The methods by which core components (individual fuel elements, reflector pieces, control rods, experimental facilities, and coolant systems) are

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

- The effect of the local environment on the material of the core support structure should be considered in the design. The impact of radiation damage, mechanical stresses, chemical compatibility with the coolant and core components, and reactivity effects should not degrade the performance of the supports sufficiently to impede safe reactor operation for the design life of the reactor.

- The design should show that stresses or forces from reactor components other than the core could not cause malfunctions, interfere with safe reactor operation or shutdown, or cause other core-related components to malfunction.

- The design for a movable core should contain features that ensure safe and reliable operation. This includes position tolerances to ensure safe and reliable reactor operation within all design limits including reactivity and cooling capability. The description should include the interlocks that keep the reactor core from moving while the reactor is critical or while forced cooling is required, if applicable. The design should show how the reactor is shut down if unwanted motion occurs.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

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

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

- The core support structure contains grid plates that accurately position and align the fuel elements. This arrangement ensures a stable and reproducible reactivity. Hydraulic forces from coolant flow will not cause fuel elements to move or bow.

- The core support structure includes acceptable guides and supports for other essential core components, such as control rods, nuclear detectors, neutron reflectors, and incore experimental facilities.

- The core support structure provides sufficient coolant flow to conform with the design criteria and to prevent loss of fuel integrity from overheating.

- The core support structure is composed of materials shown to be resistant to radiation damage, coolant erosion and corrosion, thermal softening or yielding, and excessive neutron absorption.

- The core support structure is designed to ensure a stable and reproducible core configuration for all anticipated conditions (e.g., scrams, coolant flow change, and core motion) through the reactor life cycle.

- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the core support structure and included them in the technical specifications.

4.3 Reactor Tank or Pool

Areas of Review

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

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

Acceptance Criteria

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

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

Review Procedures

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

Evaluation Findings

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

- The tank system can withstand all anticipated mechanical and hydraulic forces and stresses to prevent loss of integrity which could lead to a loss of coolant or other malfunction that could interfere with safe reactor operation or shutdown.

- The penetrations and attachments to the tank are designed to ensure safe reactor operation. Safety and design considerations of any penetrations below the water level include analyses of potential malfunction and loss of coolant. The applicant discusses credible loss-of-coolant scenarios in Chapter 13, "Accident Analyses."

- The construction materials, treatment, and methods of attaching penetrations and components are designed to prevent chemical interactions among the tank, the coolant, and other components.

- The outer surfaces of the tank are designed and treated to avoid corrosion in locations that are inaccessible for the life of the tank. Tank surfaces will be inspected in accessible locations.

- The applicant has considered the possibility that primary coolant may leak into unrestricted areas, including ground water, and has included precautions to avoid the uncontrolled release of radioactive material.

- The design considerations include the shape and dimensions of the tank to ensure sufficient radiation shielding to protect personnel and components. Exposures have been analyzed, and acceptable shielding factors are included in the tank design.

- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the tank and included them in the technical specifications.

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

4.4 Biological Shield

Areas of Review

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

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

Acceptance Criteria

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

- The principal objective of the shield design should be to ensure that the projected radiation dose rates and accumulated doses in occupied areas do not exceed the limits of 10 CFR Part 20 and the guidelines of the facility ALARA (as low as is reasonably achievable) program discussed in Chapter 11 of the SAR.

- The shield design should address potential damage from radiation heating and induced radioactivity in reactor components and shields. The design should limit heating and induced radioactivity to levels that could not cause significant risk of failure.

- The tank or pool design, the coolant volume, and the solid shielding materials should be apportioned to ensure protection from all applicable radiation and all conditions of operation.

- Shielding materials should be based on demonstrated effectiveness at other non-power reactors with similar operating characteristics, and the

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

- The analyses should include specific investigation of the possibilities of radiation streaming or leaking from shield penetrations, inserts, and other places where materials of different density and atomic number meet. Any such streaming or leakage should not exceed the stated limits.

- The shielding at experimental facilities, such as out-of-service beam tubes, should be sufficient to match the shielding factors of the gross surrounding shield.

- Supports and structures should ensure shield integrity, and quality control methods should ensure that fabrication and construction of the shield exceed the requirements for similar industrial structures.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements. The applicant should justify the proposed technical specifications in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

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

- The design offers reasonable assurance that the shield can be successfully installed with no radiation streaming or other leakage that would exceed the limits of 10 CFR Part 20 and the guidelines of the facility ALARA program.

- Reactor components are sufficiently shielded to avoid significant radiation-related degradation or malfunction.

- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the shield and included them in the technical specifications.

4.5 Nuclear Design

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

4.5.1 Normal Operating Conditions

Areas of Review

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

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

Acceptance Criteria

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

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

- Anticipated rearrangements of core components should account for uranium burnup, plutonium buildup, and poisons, both fission product and those added by design, for the life of the reactor. All operating core configurations should be compact, allowing no space within the core large enough to accept the addition of a fuel element or the addition of reactivity beyond that analyzed and found acceptable in Chapter 13 of the SAR.

- The analyses should show initial and changing reactivity conditions, control rod reactivity worths, and reactivity worths of fuel elements, reflector units, and such incore components as experimental facilities for all anticipated configurations. There should be a discussion of administrative and physical constraints that would prevent inadvertent movement that could suddenly introduce more than one dollar of positive reactivity or an analyzed safe amount, whichever was larger These analyses should address movement, flooding, and voiding of core components.

- The reactor kinetic parameters and behavior should be shown, along with the dynamic reactivity parameters of the instrumentation and control systems. Analyses should prove that the control systems will prevent nuclear transients from causing loss of fuel integrity or uncontrolled addition of reactivity.

- The analyses should show that the control systems would prevent reactor damage if incore experimental facilities were to flood or void. This could be shown by reference to the analysis in Chapter 13 of the SAR.

- The information should include calculated core reactivities for the possible and planned configurations of the reactor core and control rods. If only one core configuration will be used over the life of the reactor, the applicant should clearly indicate this. For reactors in which various core configurations could be operated over time, the analyses should show the most limiting configuration (the most compact core and highest neutron flux densities). This information should be used for the analyses in Section 4.6 of the SAR.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

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

4.5.2 Reactor Core Physics Parameters

Areas of Review

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

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

Acceptance Criteria

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

- The calculational assumptions and methods should be justified and traceable to their development and validation, and the results should be compared with calculations of other similar facilities and previous experimental measurements. The ranges of validity and accuracy should be stated and justified.

- Uncertainties in the analyses should be provided and justified

- Methods used to analyze neutron lifetime, effective delayed neutron fraction, and reactor periods should be presented, and the results should be justified. Comparisons should be made with similar reactor facilities. The results should agree within the estimates of accuracy for the methods.

- Coefficients of reactivity (temperature, void, and power) should all be negative over the significant portion of the operating ranges of the reactor. The results should include estimates of accuracy. If any parameter is not negative within the error limits over the credible range of reactor operation, the combination of the reactivity coefficients should be analyzed and shown to be sufficient to prevent reactor damage and risk to the public from reactor transients as discussed in Chapter 13 of the SAR.

- Changes in feedback coefficients with core configurations, power level, and fuel burnup should not change the conclusions about reactor protection and safety, nor should they void the validity of the analyses of normal reactor operations, including pulsing, when applicable.

- The methods and assumptions for calculating the various neutron flux densities should be validated by comparisons with results for similar reactors. Uncertainties and ranges of accuracy should be given for other analyses requiring neutron flux densities, such as fuel burnup, thermal power densities, control rod reactivity worths, and reactivity coefficients.

- Technical specifications, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes limiting conditions for operation and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

- The analyses of neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity have been completed, using methods validated at similar reactors and experimental measurements.

- The effects of fuel burnup and reactor operating characteristics for the life of the reactor are considered in the analyses of the reactor core physics parameters.

- The numerical values for the reactor core physics parameters depend on features of the reactor design, and the information given is acceptable for use in the analyses of reactor operation.

- The applicant has justified appropriate limiting conditions for operation and surveillance requirements for the reactor core physics parameters and

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

4.5.3 Operating Limits

Areas of Review

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

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

Acceptance Criteria

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

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

- A transient analysis assuming that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region should be performed. This analysis could also be based on a credible failure of a movable experiment. The analysis should show that the reactor would not be damaged and fuel integrity would not be lost. Reactivity additions under accident conditions should be analyzed in Chapter 13 of the SAR.

- An analysis should be performed that examines reactivity assuming that the reactor is operating at its maximum licensed conditions, normal electrical power is lost, and the control rod of maximum reactivity worth and any non-scrammable control rods remain fully withdrawn. The analysis should show how much negative reactivity must be available in the remaining scrammable control rods so that, without operator intervention, the reactor can be shut down safely and remain subcritical without risk of fuel damage even after temperature equilibrium is attained, all transient poisons such as xenon are reduced, and movable experiments are in their most reactive position.

- On the basis of analysis, the applicant should justify a minimum negative reactivity (shutdown margin) that will ensure the safe shutdown of the reactor. This discussion should address the methods and the accuracy with which this negative reactivity can be determined to ensure its availability.

- The core configuration with the highest power density possible for the planned fuel should be analyzed as a basis for safety limits and limiting safety system settings in the thermal-hydraulic analyses. The core configuration should be compared with other configurations to ensure that a limiting configuration is established for steady power and pulsed operation, if applicable.

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

Review Procedures

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

Evaluation Findings

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

- The applicant has discussed and justified all excess reactivity factors needed to ensure a readily operable reactor. The applicant has also considered the design features of the control systems that ensure that this amount of excess reactivity is fully controlled under normal operating conditions.

- The discussion of limits on excess reactivity shows that a credible rapid withdrawal of the most reactive control rod or other credible failure that would add reactivity to the reactor would not lead to loss of fuel integrity. Therefore, the information demonstrates that the proposed amount of reactivity is available for normal operations, but would not cause unacceptable risk to the public from a transient.

- The definition of the shutdown margin is negative reactivity obtainable by control rods to ensure reactor shutdown from any reactor condition, including a loss of normal electrical power. With the assumption that the most reactive control rod is inadvertently stuck in its fully withdrawn position, and non-scrammable control rods are in the position of maximum reactivity addition, the analysis derives the minimum negative reactivity necessary to ensure safe reactor shutdown. The applicant conservatively proposes a shutdown margin of xx in the technical specifications. The applicant has justified this value; it is readily measurable and is acceptable.

- The SAR contains calculations of the peak thermal power density achievable with any core configuration. This value is used in the calculations in the thermal-hydraulic section of the SAR to derive reactor safety limits and limiting safety system settings, which are acceptable.

4.6 Thermal-Hydraulic Design

Areas of Review

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

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

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

Acceptance Criteria

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

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

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

Review Procedures

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

Evaluation Findings

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

- The information in the SAR includes the thermal-hydraulic analyses for the reactor. The applicant has justified the assumptions and methods and validated their results.

- All necessary information on the primary coolant hydraulics and thermal conditions of the fuel are specific for this reactor The analyses give the limiting conditions of these features that ensure fuel integrity.

- Safety limits and limiting safety system settings are derived from the thermal-hydraulic analyses. The values have been justified and appear in the technical specifications. The thermal-hydraulic analyses on which these parameters are based ensure that overheating during any operation or credible event will not cause loss of fuel integrity and unacceptable radiological risk to the health and safety of the public.

13 ACCIDENT ANALYSES

Other chapters of the SAR should contain discussions and analyses of the reactor facility as designed for normal operation. The discussions should include the considerations necessary to ensure safe operation and shutdown of the reactor to avoid undue risk to the health and safety of the public, the workers, and the environment. The analyses should include limits for operating ranges and reactor parameters within which safety could be ensured. The bases for the technical specifications should be developed in those chapters.

In this chapter the applicant should present a methodology for reviewing the systems and operating characteristics of the reactor facility that could affect its safe operation or shutdown. The methodology should be used to identify limiting accidents, analyze the evolution of the scenarios, and evaluate the consequences. The analyses should start with the assumed initiating event. The effects on designed barriers, protective systems, operator responses, and mitigating features should be examined. The endpoint should be a stable reactor. The potential radiological consequences to the public, the facility staff, and the environment should be analyzed. The information and analyses should show that facility system designs, safety limits, limiting safety system settings, and limiting conditions for operation were selected to ensure that the consequences of analyzed accidents do not exceed acceptable limits.

The applicant should also discuss and analyze a postulated accident scenario whose potential consequences are shown to exceed and bound all credible accidents. For non-power reactors, this accident is called the maximum hypothetical accident (MHA). Because the accident of greatest consequence at a non-power reactor would probably include the release of fission products, the MHA, in most cases, would be expected to contain such a scenario involving fuel or a fueled experiment and need not be entirely credible. The review and evaluation should concentrate on the evolution of the scenario and analyses of the consequences, rather than on the details of the assumed initiating event.

Because the consequences of the postulated MHA should exceed those of any credible accident at the facility, the accident is not likely to occur during the life of the facility. The MHA is used to demonstrate that the maximum consequences of operating the reactor at a specific site are within acceptable limits. The applicant may choose to perform sensitivity analysis of the assumptions of the MHA. For example, reactor operating time before accident initiation may be examined to determine the change in MHA outcome if a more realistic assumption is made. Assumptions made in the accident analysis may form the basis for technical specification limits on the operation of the facility. For example, if the accident analysis assumes that the reactor operates for 5 hours a day, 5 days a week, this may become a limiting condition for operation.

The information in this chapter should achieve the objectives stated in this chapter of the format and content guide by demonstrating that all potential accidents at the reactor facility have been considered and their consequences adequately evaluated. Each postulated accident should be assigned to one of the following categories, or grouped consistently according to the type and characteristics of the particular reactor. The information for a particular reactor may show that some of the following categories are not applicable:

- MHA
- insertion of excess reactivity (ramp, step, startup, etc.)
- loss of coolant
- loss of coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

The applicant should systematically analyze and evaluate events in each group to identify the limiting event selected for detailed quantitative analysis. The limiting event in each category should have consequences that exceed all others in that group. The discussions may address the likelihood of occurrence, but quantitative analysis of probability is not expected or required. As noted above, the MHA analyzed should bound all credible potential accidents at the facility.

The applicant should demonstrate knowledge of the literature available for non-power reactor accident analyses. The Bibliography section at the end of this chapter lists documents categorized as follows: non-power reactors (in general), radiological consequences, and fuel types.

Area of Review

Area of review should include the following: systematic analysis and discussion of credible accidents for determining the limiting event in each category. The applicant may have to analyze several events in a particular accident category to determine the limiting event. This limiting event should be analyzed quantitatively. The steps suggested for the applicant to follow once the limiting event is determined for a category of accidents are given in this chapter of the format and content guide.

Acceptance Criteria

For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 criteria (10 CFR 20.1 through 20.602 and

appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and appendices for research reactors licensed on or after January 1, 1994). For research reactors licensed before January 1, 1994, the doses that the staff has generally found acceptable for accident analysis results are less than 5 rem whole body and 30 rem thyroid for occupationally exposed persons and less than 0.5 rem whole body and 3 rem thyroid for members of the public. For research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above.

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

For MHAs for research reactors, acceptable consequences may exceed 10 CFR Part 20 limits. The reviewer will evaluate this on a case-by-case basis. The applicant should discuss why the MHA is not likely to occur during the operating life of the facility.

Review Procedures

Information in the SAR should allow the reviewer to follow the sequence of events in the accident scenario from initiation to a stabilized condition. The reviewer should confirm the following:

- The credible accidents were categorized, and the most limiting accident in each group was chosen for detailed analyses.
- The reactor was assumed to be operating normally under applicable technical specifications before the initiating event. However, the reactor may be in the most limiting technical specification condition at the initiation of the event.
- Instruments, controls, and automatic protective systems were assumed to be operating normally or to be operable before the initiating event. Maximum acceptable nonconservative instrument error may be assumed to exist at accident initiation.

- The single malfunction that initiates the event was identified.

- Credit was taken during the scenario for normally operating reactor systems and protective actions and the initiation of engineered safety features.

- The sequence of events and the components and systems damaged during the accident scenario were clearly discussed.

- The mathematical models and analytical methods employed, including assumptions, approximations, validation, and uncertainties, were clearly stated.

- The radiation source terms were presented or referenced.

- The potential radiation consequences to the facility staff and the public were presented and compared with acceptable limits.

The reviewer should confirm that the facility design prevents loss of fuel integrity in the event of a credible loss-of-coolant accident (LOCA) or loss-of-flow accident. Emergency core cooling may be required for some non-power reactors to satisfy this condition.

Reactivity limits and the functional designs of control and safety-related systems should prevent loss of fuel integrity during credible accidents involving insertion of some fraction of excess reactivity. At a minimum, the amount of reactivity allowed for moveable or unsecured experiments should be analyzed. Applicable reactivity feedback coefficients and automatic protective actions, if applicable, should be included in the analyses.

Loss of fuel integrity should be prevented if normal electrical power is lost. Safe reactor shutdown should not be compromised or prevented by loss of normal electrical power.

Evaluation Findings

It is essential that all credible accidents at a non-power reactor be considered and evaluated during the design stage. Experience has shown that such facilities can be designed and operated so that the environment and the health and safety of the staff and the public can be protected. Because non-power reactors are designed to operate with primary coolant temperatures and pressures close to ambient, the margins for safety are usually large, and few, if any, credible accidents can be sufficiently damaging to release radioactive materials to the unrestricted area. For potential accidents and the MHA that could cause a release, the acceptance criteria

and review procedures discussed above are sufficiently comprehensive and will not be repeated for each postulated accident. However, the potential consequences, detailed analyses, evaluations, and conclusions are facility specific and accident specific. The findings for the nine major accident categories are presented below. These findings are examples only. The actual wording should be modified for the situation under review.

This section of the SAR should contain sufficient information to support the types of conclusions given below. Those conclusions will be included in the staff's safety evaluation report. The appropriate number for the reactor under evaluation should replace the notation "xx". The reviewer should modify these conclusions to conform to the reactor design under consideration.

Maximum Hypothetical Accident

- The applicant has considered the consequences to the public of all credible accidents at the reactor facility. A maximum hypothetical accident (MHA), an accident that would release fission products from a fuel element or from the failure of a fueled experiment and would have consequences greater than any credible accident, has been analyzed. The MHA, however, is not credible for a non-power reactor. (*The MHA is specific to the reactor design and power. The reviewer may have to evaluate an MHA that differs from the grouping of MHAs that follows. The reviewer should select from items a–e, if appropriate.*)

    (a) (*For TRIGA, PULSTAR, or SPERT fuel*), xx (*an agreed-upon number, normally one for TRIGA or SPERT fuel; although three has been accepted for PULSTAR, the number is determined on a case-by-case basis*) fuel assemblies lose cladding integrity while suspended in air (*or in the reactor pool*) in the reactor confinement (*or containment*). All fission products in the gap are released rapidly. The fuel assembly has just been removed from the maximum neutron flux position in the core after long, continuous operation at full licensed power (*or full fuel cycle*).

    (b) [*For low-powered (less than 2 MW) MTR fuel*] An assembly is stripped of all cladding on one face of one fuel plate while suspended in air (*or in the reactor pool*) in the reactor confinement (*or containment*). All fission products escape rapidly by physically sound processes (e.g., conservative analysis, experimental data, or the combination of the two verify the release process). The fuel assembly has just been removed from the maximum neutron flux position in the core after long, continuous operation at full licensed power (*or full fuel cycle*).

(c) (*For high-powered reactors*) Fuel cooling is compromised or reactivity is added to the reactor so that a certain amount of fuel melts causing cladding failure. Fission products are released into the reactor coolant and then into the facility air on the basis of conservative analysis, empirical information, or the combination of analysis and data.

(d) (*For reactors in which a fueled-experiment failure has greater consequences than fuel failure*) It is assumed that a fueled experiment fails in air (*or water*) in a reactor irradiation facility. (*Because failure could include melting, all available fission products, or that portion that is demonstrated by analysis, data, or a combination of the two*) Fission products are assumed to escape to the reactor confinement (*or containment*). The inventory of fissile material is the maximum allowed by technical specifications for a fueled experiment and is consistent with Regulatory Guide 2.2. The failure occurs after long, continuous operation at full licensed power.

(e) (*For AGN-201 fuel*) It is assumed that fissionable material is inserted into an irradiation facility in the reactor. The added reactivity causes a power excursion. Fuel failure does not occur and the radiological consequence is limited to whole-body dose of xx mrem to the reactor staff.

The reviewer should modify the following paragraphs, as appropriate:

- The air handling and filtering systems (i.e., confinement or containment) are assumed to function as designed, and radioactive material is held up temporarily in the reactor room and then released from the building. Realistic methods are used to compute external radiation doses and dose commitments resulting from inhalation by the facility staff. Realistic but conservative methods are used to compute potential doses and dose commitments to the public in the unrestricted area. Methods of calculating doses from inhalation or ingestion (or both) and direct shine of gamma rays from dispersing plumes of airborne radioactive material are applicable and no less conservative than those developed in Chapter 11 of the SAR. The exposure time for the reactor staff is xx and for the public it is xx.

- The calculated maximum effective doses for the MHA scenario are the following:

  - external—(xx mrem) staff; (xx mrem) public
  - internal—(xx mrem) staff; (xx mrem) public

- These doses and dose commitments are within the acceptable limits (*state the limits*). Because the assumptions of the scenario are conservative, the postulated accident would not be likely to occur during the life of the facility. The applicant has examined more realistic assumptions about operating time and release fractions that decreased the source term by xx percent of the one calculated, lowering the maximum doses by that factor (*if applicable*). Thus, even for the MHA, whose consequences bound all credible accidents possible at the facility, the health and safety of the facility staff and the public are protected.

Insertion of Excess Reactivity

The reviewer should select one of the two findings that follow:

(1) The applicant has discussed possible methods by which excess reactivity could be inserted accidentally into the reactor to cause an excursion. Rapid insertions were initiated by (*state the initiators analyzed, some examples follow*):

- dropping of a fuel assembly or a fueled experiment into a core vacancy
- removal or ejection of a control, safety, or transient rod
- sudden malfunction, movement, or failure of an experiment or experimental facility
- insertion of a surge of cold primary coolant
- malfunction of reflector components

Slow insertions were initiated by (*state the initiators analyzed, some examples follow*):

- insertion of a fuel assembly or fueled experiment into a core vacancy
- malfunction of a control or safety rod system
- operator error, especially at reactor startup (inadvertent criticality)
- malfunction of power level indicator, especially at reactor startup

- protracted malfunction, movement, or leakage of an experiment or experimental facility

- malfunction of reflector components

The applicant has discussed the scenario for the above events, presented a qualitative evaluation, and compared the likely consequences.

The SAR shows that physical limitations and technical specifications provide acceptable assurance that inadvertent removal or ejection of a control rod, a safety rod, or both, is prevented unless sufficient fuel has been removed, which would ensure subcriticality. Similar controls offer acceptable assurance that fuel or fueled-experiment handling above the core is prevented unless the control rods are in position to ensure subcriticality. Even with such controls, fuel or a fueled experiment could be handled while the reactor is in a critical state and while the core has a fuel vacancy at the core periphery. It is postulated that a fuel element or fueled experiment is inadvertently dropped into the vacancy, rapidly inserting reactivity equal to its worth at that position The reactor enters a supercritical state by xx% $\Delta k/k$, which induces a stable positive reactor period of xx msec. Reactor power increases so fast that safety rods are assumed not to move significantly during the transient, even though both the period scram and power level scram are tripped. The power level and fuel temperature are analyzed by validated and acceptable methods. The analyses show that the steam void formed in the core reduces reactivity sufficiently to terminate the excursion, or the prompt negative temperature coefficient of the fuel reduces reactivity sufficiently to terminate the excursion. The safety rods continue to insert within their required drop time, which stabilizes the subcritical reactor. During the transient, xx MW-sec/g of energy was deposited in the hottest point of a fuel element, raising its maximum temperature to xx °C. Because this temperature is lower than the safety limit temperature of the fuel cladding xx °C, fuel integrity would not be lost. Therefore, no fission products would be released from the primary barrier by this accident. (*This approach could also be used for experiment malfunction and other rapid additions of reactivity.*)

Because of the peak power level during the transient, the operator inserting the fuel or fueled experiment was exposed to a brief pulse of radiation. The integrated dose was computed not to exceed xx mrem, which is below acceptable limits for occupational exposures.

or

(2) The SAR shows that physical and technical specification limitations give reasonable assurance that a rapid insertion of reactivity is not credible. However, malfunction of the control rod drive mechanism or operator error during reactor startup could cause an inadvertent withdrawal of the control rod and an unplanned increase in reactor power. The accident scenario assumes that the reactor has a maximum load of fuel (consistent with the technical specifications), the reactor is operating at full licensed power, and the control system malfunction withdraws the control rod of maximum reactivity worth at its maximum drive speed. Both the power level scram and reactor period scram are assumed to be operable. (*In some analysis it is assumed that the first scram that would terminate the reactivity addition fails and that the second scram terminates the event. In some cases, both scrams are assumed to have failed. If this is the case, the evaluation should be modified appropriately.*) The continuous removal of the rod causes a continuous decrease of reactor period and a continuous increase in reactor power. The analyses, including trip level uncertainties and rod-drop delays, show that the period scram terminates the power increase before the thermal reactor power reaches xx MW. The thermal-hydraulic analysis shows that the energy deposited and instantaneous power level would not raise the peak temperature in the hottest fuel element above xx °C. Because this temperature is lower than the safety limit temperature for fuel cladding (xx °C), fuel integrity would not be lost. (*This approach could be used for other slow additions of reactivity.*)

Loss of Coolant

- The applicant has discussed possible methods by which sufficient primary coolant would be lost rapidly to pose a risk to adequate removal of heat from the fuel. The credible accident with the worst potential consequences is initiated by the catastrophic failure of (*state the component that fails, usually a beam tube or primary coolant pipe*), which would allow a coolant loss at xx liter/min initially. The scenario assumes that the reactor is operating at full licensed power and has been operating long enough for the fuel to contain fission products at equilibrium concentrations. Therefore, the maximum possible decay heat is available at the start of the event. The pool level scram shuts down the reactor when the coolant reaches the technical specification level. Coolant reaches the top of the core in xx min, and the bottom of the core in xx min. At this time, decay heat raises fuel temperatures. For the SAR analyses, the applicant used validated and acceptable methods to calculate fuel temperature changes.

The reviewer should select one of the following situations:

- With natural-convection air cooling, the analyses show that the peak fuel temperature will not exceed xx °C in xx hr, which is below the temperature necessary for fuel cladding to maintain fuel integrity.

- With the emergency core cooling system functioning as designed, the analyses show that the peak fuel temperature reaches no more than xx °C, which is below the temperature necessary for fuel cladding to maintain fuel integrity.

- As the primary coolant escapes and the reactor core becomes uncovered, the decay fission products constitute an unshielded gamma-ray source near the bottom of the pool. This source could expose personnel above the pool to direct gamma radiations and personnel on the floor of the reactor room to scattered gamma radiations. The applicant has analyzed both locations, including the potential doses to facility staff and the public in unrestricted areas. The delay time while the water is escaping from the reactor pool allows the facility staff to take cover and avoid doses larger than xx mrem. The maximum potential dose rates in the unrestricted area would not exceed xx mrem/hr, which provides sufficient time for protective action, if required, so that no doses would exceed acceptable limits.

- To determine the maximum potential consequences for fuel integrity and personnel, the applicant has analyzed a loss-of-coolant scenario in which all primary coolant is lost instantaneously (*if applicable*). The other assumptions are the same as for the slower loss evaluated above. Although the assumptions for this scenario exceed those discussed above, fuel integrity should be ensured and personnel doses would not exceed acceptable limits.

Loss of Coolant Flow

The reviewer should select one of the two findings that follow:

(1) The applicant has discussed possible methods by which coolant flow through one or more fuel channels could be interrupted while the reactor is operating. The postulated initiating events range from total loss of forced flow as a result of pump or normal electrical power failure to blockage of xx fuel channel(s) by a foreign object. The scenario assumes that the reactor has been operating at full power and fission product decay rates have reached equilibrium.

When the pump stops, a conservative assumption is that forced flow stops instantly (*pump coastdown can be used in the calculations if appropriate*). The coolant-flow scram shuts down the reactor within the technical

specification time limits for circuit delays and rod-drop times. The reactor is designed to change passively to natural-convection flow when forced flow ceases. However, during the changeover, there is a transient period before natural-convection flow can remove decay heat. The analyses account for this transient, showing that the peak fuel temperature does not reach an unacceptable value. Therefore, the maximum credible loss-of-flow accident would not cause loss of fuel integrity.

For blocked fuel cooling channels, the applicant has analyzed heat transfer around the area of the blockage. Appropriate assumptions have been made concerning the amount of time that passes without detection of the blockage. If the blockage is indicated by the reactor instrumentation, reactor operators take appropriate action. Thermal-hydraulic analysis shows that the peak fuel temperature in the area of the blockage will not reach an unacceptable value. Therefore, such blockage would not cause fuel integrity to be lost.

or

(2) The applicant has shown that fuel cooling channel blockage could lead to fuel melting and fuel cladding failure. The analysis shows that this event is bounded by the fuel failure discussed in the section on the MHA. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and the public are protected.

(If the MHA is not a fuel failure accident, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that a blocked fuel-cooling channels event is not the MHA.)

Mishandling or Malfunction of Fuel

- The applicant has discussed initiating events that could damage fuel or accidentally release fission products from irradiated fuel in the core, in storage, or in between the core and the storage area. The events that would cause the worst radiological consequences have been analyzed by the applicant. This event is *(provide description)*.

- The analysis shows that this event is bounded by the fuel failure discussed under the MHA. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and the public are protected.

(If the MHA is not a fuel failure accident, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that this mishandling or malfunction of fuel is not the MHA.)

Experiment Malfunction

- The applicant has discussed the types of experiments that could be performed at the reactor within its license and technical specifications. The discussions include events that could initiate accidents such as *(list events, some examples are given below):*

  - melting, leaking, detonation, or failure of the experimental material or its encapsulation, allowing radioactive material to escape into the reactor room or the air exhaust stream to the unrestricted environment

  - movement or misplacement of an experiment into a location of radiation intensity higher than that for which it was planned

  - movement, melting, or other failure of a neutron-absorbing experiment, which causes positive reactivity to be inserted inadvertently into the reactor

  - movement, failure, or leakage of an experimental facility, which causes positive reactivity to be inserted inadvertently into the reactor or radioactive material to be released by the malfunction

- The analysis shows that the technical specifications that limit experiment types and magnitudes of reactivities give reasonable assurance that the potential consequences of these initiating events would be less severe than those already evaluated in the section on the MHA or in fuel handling accident scenarios.

  (If the MHA is not a release of radioactive material, the reviewer should use wording similar to the conclusions for the MHA fuel failure presented above to state the conclusions for this type of accident. The wording should be modified to account for the fact that experiment malfunction is not the MHA.)

Loss of Normal Electrical Power

- The applicant has discussed the events that could result from the sudden loss of normal electrical power. The reactor is designed so that the force of gravity automatically inserts safety or control rods (*or describe the system used that does not require electrical power*) and shuts down the fission reactions when power is lost. Furthermore, reactors with natural-convection cooling are not affected (*reactors with forced-convection cooling passively change to natural convection to remove decay heat when power is lost*).

The reviewer should modify the following statement to apply to the reactor under discussion.

- Most licensed non-power reactors have a large reserve of coolant in the pool that can absorb decay heat for hours, if necessary, without transfer of heat to the secondary system. In a few non-power reactors, emergency electrical power is eventually required to transfer heat to the secondary system. In some non-power reactors, emergency electrical power must be available for specified instrument and control functions. Emergency power design is discussed in Chapter 8 of the SAR. On the basis of these considerations, loss of normal electrical power at a non-power reactor would not pose undue risk to the health and safety of the public.

External Events

- The design to withstand external events and the potential associated accidents is discussed in Chapters 2 and 3 of the SAR. The reactor facility is designed to accommodate these events by shutting down, which would not pose undue risk to the health and safety of the public. For events that cause facility damage (*seismic events that damage the reactor facility or pool*), the damage is within the bounds discussed for other accidents in this chapter. Therefore, exposure to the staff and the public is within acceptable limits and external events do not pose an unacceptable risk to the health and safety of the public. (*An external event could be the MHA if enough damage is done to the facility to damage fuel. The conclusion above for the MHA would apply.*)

Mishandling or Malfunction of Equipment

Initiating events under this heading would require a case-by-case, reactor-specific discussion. If the SAR discusses additional events that fall outside the eight categories, the potential consequences should be compared with similar events already analyzed or with the MHA, as applicable.

Bibliography

Non-Power Reactors

American Nuclear Society (ANS), 5.1, "Decay Heat Power in Light Water Reactors," LaGrange Park, Illinois, 1978.

Atomic Energy Commission, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844, March 23, 1962.

Baker, L., Jr., and Just, L. C., "Studies of Metal-Water Reactions at High Temperatures, III, Experimental and Theoretical Studies of the Zirconium Water Reaction," ANL 6548, Argonne National Laboratory, 1962.

Baker, L., Jr., and Liimatakinen, R. C., "Chemical Reactions" in Volume 2 of The Technology of Nuclear Reactor Safety, Thompson and Beckerly (eds.), Cambridge, Massachusetts: The MIT Press, 1973, pp. 419-523.

Hunt, C. H., and DeBevee, C. J., "Effects of Pool Reactor Accidents," General Electric Technical Information Series, GEAP 3277, November 2, 1959.

Woodruff, W. L., "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," Nuclear Technology, 64, February 1984, pp. 196-206.

Radiological Consequences

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," Publication 30, Part 1, Chapter 8, Pergamon Press, 1978/1979.

Lahti, G. P., et al., "Assessment of Gamma-Ray Exposures Due to Finite Plumes," Health Physics, 41, 1981, p. 319.

U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

U.S. Nuclear Regulatory Commission, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, October 1977.

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

Behavior of Zirconium-Hydride Fueled Reactors (TRIGA)

Atomic Energy Commission, "In the Matter of Trustees of Columbia University in the City of New York," Docket No. 50-208, Atomic Energy Commission Reports, Issued May 18, 1972.

Baldwin, N. L.; Foushee, F. C.; and Greenwood, J. S., "Fission Product Release From TRIGA-LEU Reactor Fuels," in Seventh Biennial U.S. TRIGA Users Conference, San Diego, CA, 1980.

Coffer, C. O.; Shoptaugh, J. R.; Jr., and Whittemore, W. L., "Stability of the U-ZrH TRIGA Fuel Subjected to Large Reactivity Insertion," GA-6874, General Atomics Company, transmitted by letter dated July 25, 1967 (Docket No. 50-163), January 1966.

Foushee, F. C., and Peters, R. H., Summary of TRIGA Fuel Fission Product Release Experiments," Gulf-EES-A10801, September 1971.

General Atomics Company, "Technical Foundations of TRIGA," GA-0471, August 1958.

Kessler, W. E., et al., "Zirconium-Hydride Fuel Behavior in the SNAPTRAN Transient Tests," Transactions of the American Nuclear Society, 9, 1966, p. 155.

Lindgren, J. R., and Simnad, M. T., "Low-Enriched TRIGA Fuel Water-Quench Safety Tests," Transactions of the American Nuclear Society, 33, 1979, p. 276.

Shoptaugh, J. R., Jr., "Simulated Loss-of-Coolant Accident for TRIGA Reactors," GA-6596, Gulf General Atomic, August 18, 1965.

Simnad, M. T., "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," GA-4314, E-117-833, General Atomics Company, February 1980.

Simnad, M. T., and Dee, J. B., "Equilibrium Dissociation Pressures and Performance of Pulsed U-ZrH Fuels at Elevated Temperature," in Thermodynamics of Nuclear Materials, Proceedings of a Symposium, Vienna, September 4-8, 1967, International Atomic Energy Agency, 1968.

Simnad, M. T.; Foushee, F. C.; and West, G. B., "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, 28, 1976, pp. 31-56.

U.S. Nuclear Regulatory Commission, Generic Credible Accident Analysis for TRIGA Fueled Reactors," NUREG/CR-2387, Pacific Northwest Laboratory, 1982.

U.S. Nuclear Regulatory Commission, Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors, NUREG-1282, 1987.

West, G. B., et al., "Kinetic Behavior of TRIGA Reactors," GA-7882, General Atomics Company, March 1967.

Behavior of Plate Fueled Reactors (MTR)

Bullock, J. B., "Calculation of Maximum Fuel Cladding Temperatures for Two Megawatt Operation of the Ford Nuclear Reactor," Memorandum Report No. 1, Memorial Phoenix Project, Michigan, June 1962.

Forbes, S. G., et al., Instability in the SPERT I Reactor, Preliminary Report, IDO-16309, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, October 1956.

Knexevich, M., et al., "Loss of Water at the Livermore Pool Type Reactor," Health Physics II, 1965, pp. 481-487.

Miller, R. N., et al., Report of the SPERT I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor, IDO-16883, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, June 1964.

Nyer, W. E., et al., Experimental Investigations of Reactor Transients, IDO-16285, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, April 1956.

Shibata, T., et al., "Release of Fission Products from Irradiated Aluminide Fuel at High Temperatures," Nuclear Science and Engineering, 87, 1984, pp. 405-417.

Sims, T. M., and Tabor, W. H., Report on Fuel-Plate Melting at the Oak Ridge Research Reactor, July 1, 1963, ORNL-TM-627, Oak Ridge National Laboratory, October 1964.

Wett, J. F., Jr., Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air, ORNL-2892, Oak Ridge National Laboratory, April 16, 1960.

U.S. Nuclear Regulatory Commission, "Analysis of Credible Accidents for Argonaut Reactors," NUREG/CR-2079, Pacific Northwest Laboratories, April 1981.

U.S. Nuclear Regulatory Commission, Rubenstein, L. S., memo to Tedesco, R. L., "Design Basis Event for the University of Michigan Reactor," June 17, 1981.

U.S. Nuclear Regulatory Commission, "Fuel Temperatures in an Argonaut Reactor Core Following a Hypothetical Design Basis Accident (DBA)," NUREG/CR-2198, Los Alamos National Laboratory, February 1981.

U.S. Nuclear Regulatory Commission, Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors, NUREG-1313, 1988.

Hunt, C. H., and C. J. DeBevee, "Effects of Pool Reactor Accidents," General Electric Technical Information Series, GEAP 3277, Pleasanton, CA, November 2, 1959.

Woodruff, W. L., "A Kinetics and Thermal-Hydraulics Capability for the Analysis of Research Reactors," *Nuclear Technology*, 64, February 1984, pp. 196-206.

Radiological Consequences

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," Publication 30, Part 1, Chapter 8, Pergamon Press, Oxford, NY, 1978/1979.

Lahti, G. P., et al., "Assessment of Gamma-Ray Exposures due to Finite Plumes," *Health Physics*, 41, 1981, p. 319.

U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.