DRAFT

Interim Staff Guidance Augmenting

NUREG-1537, Part 2,

“Guidelines for Preparing and

Reviewing Applications for the

Licensing of Non-Power Reactors:

Standard Review Plan and Acceptance Criteria,”

for the Production of Radioisotopes

[ENTER DATE HERE]

Contents

ABBREVIATIONS

ADAMS	Agencywide Documents Access and Management System
AHR	aqueous homogeneous reactor
ALARA	as low as reasonably achievable
ANSI/ANS	American National Standards Institute/American Nuclear Society
APE	area of potential effect
ASTM	American Society for Testing and Materials
BGEPA	Bald and Golden Eagle Protection Act
BLM	U.S. Bureau of Land Management
BR-2	Belgian Reactor-2
CAAS	criticality accident alarm system
CEQ	Council on Environmental Quality
CFR	*Code of Federal Regulations*
DBA	design-basis accident
DBC	design-basis criteria
DOE	U.S. Department of Energy
DP	decommissioning plan
DWMEP	Division of Waste Management and Environmental Protection
EA	environmental assessment
ECCS	emergency core cooling system
EFH	essential fish habitat
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ER	environmental report
ESF	engineered safety feature
FP	fission product
FM	Factory Mutual
FNMC	fundamental nuclear material control
FONSI	finding of no significant impact
FSME	Office of Federal and State Materials and Environmental Management Programs
HEU	highly enriched uranium
HFR	High-Flux Reactor
HVAC	heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
IROFS	item(s) relied on for safety
ISA	integrated safety analysis
ISG	interim staff guidance
kW	kilowatt(s)
LCO	limiting condition for operation
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LSSS	limited safety system setting
MBTA	Migratory Bird Treaty Act

MHA	maximum hypothetical accident
MIPS	medical isotope production system
MMPA	Marine Mammal Protection Act
Mo	molybdenum
MPC	maximum permissible concentration
NCS	nuclear criticality safety
NEPA	National Environmental Policy Act of 1969
NFPA	National Fire Protection Association
NMFS	National Marine Fisheries Service
NNSA	National Nuclear Security Administration
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NRCS	Natural Resources Conservation Service, U.S. Department of Agriculture
NRHP	National Register of Historic Places
NRU	National Research Universal
PSAR	preliminary safety analysis report
OSHA	Occupational Safety and Health Administration
RAM	remote area monitors
RG	regulatory guide
SAR	safety analysis report
SGI	Safeguards Information
SHPO	State Historic Preservation Office
SL	safety limits
SNM	special nuclear material
SR	surveillance requirement
Tc	technetium
TEDE	total effective dose equivalent
THPO	Tribal Historic Preservation Office
TS	technical specification(s)
U-235	uranium-235

Introduction

Purpose

This interim staff guidance (ISG) augments the following:

- NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,” Part 1, February 1996

- NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” Part 2, February 1996

This ISG updates and expands the content of NUREG-1537 to provide guidance for applicants in preparing a license application and for the U.S. Nuclear Regulatory Commission (NRC) staff in evaluating the application and issuing a license for the following:

- A heterogeneous or an aqueous homogeneous (non-power) reactor (AHR) as a utilization facility pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities”

- A production facility for the separation of byproduct material from special nuclear material (SNM) pursuant to 10 CFR Part 50. The production facilities addressed in this ISG are facilities that will separate isotopes from the following sources:

  – targets irradiated in a non-power reactor

  – the core of an AHR

  – the content of a subcritical multiplier solution tank containing SNM and fission products resulting from incident accelerator-generated neutrons

Overview of Medical Isotope Production

For the past two decades, the United States has relied on imported medical radioisotopes to perform approximately 40,000 medical procedures daily. Simultaneously, U.S. policy has been to reduce the use of highly enriched uranium (HEU). The Energy Policy Act of 2005 called for the National Research Council to study ways to ensure a reliable supply of medical isotopes and, furthermore, to do so without the use of HEU. Global shortages of medical isotopes during 2009 and 2010 have underscored the need for prompt action to ensure a reliable domestic supply. The U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) subsequently entered into agreements with domestic commercial firms to encourage the expeditious construction of medical isotope production facilities, which will require NRC operating licenses to operate. Potential license applicants have filed letters of intent or otherwise expressed their intent to obtain NRC licenses to operate such facilities. While licensing regulations are in place that can be applied to all technologies proposed to date, the NRC has not developed and published guidance on application content and a standard review plan that addresses each of these technologies. The guidance presented in this document augments existing regulatory guidance to define a means to license medical isotope production facilities in a manner that ensures adequate protection of public health and safety, promotes the common defense and security, and is protective of the environment.

While numerous isotopes are commonly used as radiopharmaceuticals today, the isotope currently in highest demand is molybdenum-99 (Mo-99). Mo-99 decays with a 66-hour half-life to technetium-99m (Tc-99m), which, in turn, decays with a 6-hour half-life to Tc-99. Common industry practice is to produce bulk Mo-99 and ship it to a manufacturer of generators that are sent to hospitals, medical centers, or radio-pharmacies. The generator manufacturer loads the Mo-99 onto a chromatographic-separation or ion-exchange column where it decays to Tc-99m, which is periodically washed (i.e., eluted) from the column with isotonic saline solution, leaving the Mo-99 in place for subsequent decay and production of additional Tc-99m. This ISG applies only to the bulk production of isotopes and not to the manufacture of devices to dispense radiopharmaceuticals, such as generators.

Two techniques commonly used for the production of Mo-99 are neutron activation of natural molybdenum, which is 24-percent Mo-98, and the fissioning of uranium-235 (U-235), which has a fission yield of 6-percent Mo-99. Fission product Mo-99 has become the most common method of production, because it has very high specific activity. Mo-99 is produced using the fission process when neutrons fission U-235 in a target placed in a reactor, in the fuel solution of an AHR, or in a solution tank (or reaction vessel) containing U-235 used as a subcritical multiplier of neutrons produced by accelerator interactions. Other techniques of producing Mo-99 have been studied (e.g., the removal of a neutron from enriched stable Mo-100 accelerator targets).

A history and analysis of medical isotope research and development and descriptions of the development of an international isotope production industry and the U.S. role appear in the report by the Nuclear and Radiation Studies Board of the National Research Council, *Medical Isotope Production Without Highly-Enriched Uranium*, issued by the National Academies Press in 2009. Among its findings, the report characterizes Mo-99 production before 2009 as follows:

Reactor	Country	Date of Initial Criticality	Supply of US Demand	Supply of World Demand
National Research Universal	Canada	1957	60%	40%
High-Flux Reactor	Netherlands	1961	40%	25%
Belgian Reactor 2	Belgium	1961	0	20%
Others	na	na	0	15%

The following findings of the report and subsequent events characterize the environment in which potential applicants have expressed interest in NRC licenses to construct and operate domestic medical isotope production facilities:

- Serious shortages of medical isotopes occurred domestically and internationally during 2009 and 2010.

- The National Research Universal (NRU) reactor experienced an unscheduled 18-month (May 2009 to August 2010) outage to repair a coolant leak.

- The High-Flux Reactor (HFR) required a scheduled 3-month (February 2010 to September 2010) piping repair outage This event occurred simultaneously with the NRU reactor outage.

- The majority of the world's isotope supply comes from reactors 50 years old or older.

- Only a small fraction of medical isotopes are produced from low-enriched uranium (LEU) (Australia and South Africa).

- 100 percent of the U.S. isotope demand comes from two sources; 85 percent of the world isotope demand comes from three sources.

- The April 2010 volcano in Iceland disrupted air transport in Europe, interfering with medical isotope distribution.

<u>Characterization of Potential Applications and Licensing Requirements</u>

The NRC staff has researched various isotope production technologies and facilities that it may be asked to license. Technical information has come from letters of intent, verbal and written inquiries regarding the licensing process, cooperative agreements announced by NNSA, and technical presentations at professional society meetings. Five technologies that are under consideration are identified below, along with an outline of the licensing requirements for each:

(1) Mo-99 is produced by accelerator interaction with enriched Mo-100 targets.

  – This requires a byproduct materials license issued by an Agreement State or, if the facility is located in a Non-Agreement State, by the NRC under 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." No additional NRC staff guidance is needed in this situation.

(2) Mo-99 is produced by the activation of natural Mo in existing research and power reactors.

  – Most nonpower reactors are licensed to perform "experiments", which may include the activation of targets. This constitutes normal use of the reactor. If the proposed use cannot be authorized under 10 CFR 50.59, "Changes, Tests, and Experiments," or is outside the scope of the technical specifications (TS) for approved experiments, a routine license amendment will be required.

  – Power reactor licenses generally do not allow the intentional activation of targets and the insertion and removal of targets from the core. Therefore, a routine amendment will be required for a power reactor. No additional NRC staff guidance is needed to clarify the licensing path in this situation.

(3) Mo-99 is produced by fissioning special nuclear material (SNM) in low enriched uranium (LEU) targets in existing or newly-constructed nonpower reactors. Mo-99 is then separated from the irradiated targets. These irradiations are governed by the facility license and TS.

- Heterogeneous reactors are addressed by the existing standard review plan for non-power reactors (NUREG-1537) and fueled experiments can be licensed based on that document with minimal additional guidance, as discussed later in this document.

- The facility where the isotope separation process occurs may be considered a production facility subject to licensing under 10 CFR Part 50. The NRC staff guidance for licensing a production facility is discussed later in this document.

(4) An LEU-fueled AHR and a production facility may be constructed and operated to separate the fission product Mo-99 from the liquid core after a short period of operation.

- The existing standard review plan for nonpower reactors (NUREG-1537) does not specifically address homogeneous fuels. The NRC staff guidance for licensing an AHR is discussed later in this document.

- The facility where the isotope separation process occurs may be considered a production facility subject to licensing under 10 CFR Part 50. The NRC staff guidance for licensing a production facility is discussed later in this document.

(5) A reaction vessel containing a subcritical solution of LEU may be constructed for the multiplication of accelerator-generated neutrons by fission of the uranium, and a facility may be constructed to separate the fission product Mo-99 from the solution after a short period of operation.

- The facility where the radioisotope separation process occurs may be considered a production facility and, if so, is subject to licensing under 10 CFR Part 50. The NRC staff guidance for licensing a production facility is discussed later in this document.

- The subcritical multiplier reaction vessel containing SNM by definition is not a reactor, because it cannot sustain a chain reaction. It may be included in the 10 CFR Part 50 production facility license as an assembly containing SNM that is authorized for use in conjunction with the production facility. A safety analysis report (SAR) accompanies the application to evaluate the performance of the solution tank relative to many of the same phenomena identified as licensing concerns for an AHR.

<u>Licensing of 10 CFR Part 50 Utilization Facilities</u>

NUREG-1537 contains guidance for licensing non-power reactors. While AHRs had been licensed and operated in the United States before 1996, no AHRs were in operation and none were anticipated in the foreseeable future when NUREG-1537 was written. As a result, NUREG-1537, Part 1, Chapter 4, “Reactor Description,” Section 4.2.1, states: “Most non-power reactors contain heterogeneous fuel elements consisting of rods, plates, or pins, which are addressed in the following sections. Homogeneous fuels should be described and analyzed in a comparable way.” In anticipation of an AHR application for the production of medical isotopes, the NRC staff has prepared this ISG to supplement NUREG-1537 where appropriate.

This ISG provides alternative guidance for AHRs and radioisotope production facilities. The content of NUREG-1537 Chapter 4 “Reactor Description,” Chapter 5, “Reactor Coolant Systems,” Chapter 6 “Engineered Safety Features,” Chapter 7 “Reactor Instrumentation,” Chapter 12 “Conduct of Operations,” Chapter 13, “Accident Analyses,” and Chapter 14, “Technical Specifications,” has changed significantly. This ISG contains guidance for all other chapters indicating how the remainder of NUREG-1537 as published can be effectively applied to an AHR application for a 10 CFR Part 50 utilization and radioisotope production facility license.

This ISG also provides guidance on applications for a new heterogeneous nonpower reactor license. In this case, NUREG-1537 remains generally applicable, but changes in regulations (e.g., 10 CFR 50.33(k)(1) and 10 CFR 50.75 related to decommissioning requirements) and updated reference documents are addressed.

Licensing of 10 CFR Part 50 Production Facilities

Facilities separating radioisotopes from irradiated SNM will be licensed as production facilities under 10 CFR Part 50 unless an exemption is applied for and granted, or the facility meets one of the exceptions to the definition for *Production facility* found in 10 CFR 50.2.

A facility meeting any of these exceptions is by definition not a production facility and is therefore not subject to the 10 CFR Part 50 production facility requirements; rather, it would be considered an SNM fuel cycle facility subject to the requirements of 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.” NUREG-1520, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,” Revision 1, issued May 2010, presents the standard review plan for a 10 CFR Part 70 facility.

The NRC staff has not previously developed guidance in the form of a standard review plan for a 10 CFR Part 50 production facility, therefore this ISG will provide such guidance. The ISG follows the structure of that prepared for a 10 CFR Part 50 utilization facility in NUREG-1537. Certain topics, such as site characterization and conduct of operations, are relevant to both production and utilization facilities and are incorporated by reference. Other topics, such as facility description and accident analysis, are significantly different; for these topics, the NRC staff engaged personnel with expertise in fuel cycle facilities and drew extensively from their expertise and the standard review plan in NUREG-1520.

Production facilities that employ the reaction vessel subcritical neutron multiplier method for producing radioisotopes present a special licensing situation. The isotope separation facility must be licensed as a production facility (unless it falls under one of the exceptions listed in subpart (3) the definition of Production facility found in 10 CFR 50.2) . Meanwhile, the reaction vessel is not, by definition, a reactor because the fission process occurring within the vessel is not self-sustaining. The SNM in the solution tank may therefore be licensed as material possessed by the licensee used in conjunction with the operation of the production facility.

While the reaction vessel is not a reactor, its safety analysis must consider phenomena analogous to those of an AHR. The reaction vessel can achieve relatively high power levels from the fission process. The production of reasonable and practical quantities of radioisotopes on a commercial scale may require operating power levels on the order of 50 to 75 kilowatts (kW). While the assembly is maintained subcritical, it will have to be operated very much like an AHR, with controls for managing temperature and pressure of the fuel solution, maintaining radiolytic gases at safe levels, and containing fission products, some of which are volatile in the solution. It will need the same protective structures, systems, and components that are required for an AHR. Many of the hazards and concerns associated with AHRs that are addressed in this ISG will also apply to the reaction vessel subcritical neutron multiplier. Applicants for licensing this type of facility should therefore follow the guidance in this ISG, as appropriate, for developing a safety analysis for both the reaction vessel containing the fission process and the associated radioisotope separation and purification processes involved in the radioisotope production process.

<u>Presentation of Interim Staff Guidance</u>

Considering the preceding factors, the NRC is publishing the following documents as the ISG augmenting the 1996 version of NUREG-1537 to better inform the licensing of a heterogeneous reactor or an AHR as a utilization facility and the licensing of a radioisotope production facility for the separation of byproduct materials from the fission products of irradiated SNM pursuant to 10 CFR Part 50:

- “Interim Staff Guidance Augmenting NUREG-1537, ”Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,“ Part 1 for the Production of Radioisotopes **[ENTER DATE HERE]**

- “Interim Staff Guidance Augmenting NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria”, Part 2 for the production of radioisotopes **[ENTER DATE HERE]**

Abstract and Introduction

The “Abstract” and “Introduction” sections of the current NUREG-1537 present some background and general information that is applicable to all non-power reactors and that can also apply to a radioisotope production facility that is licensed under 10 CFR Part 50. Applicants preparing SAR’s for radioisotope production facilities can use the information in these sections of the NUREG with the understanding that where the term “reactor” appears it can be interpreted to mean “reactor and production facility,” as appropriate. When preparing a SAR, applicants for a production facility license should use the NUREG as it is augmented by this ISG.

1 The Facility

This ISG is intended to supplement NUREG-1537. Listed below are additional items that the reviewer must consider when reviewing the application. In general, the acceptance criteria described in NUREG-1537 are applicable in reviewing the non-power reactor (utilization facility) and the radioisotope production facility (production facility). The following sections should be revised as indicated to include information that must be provided for the radioisotope facility.

1.1 Introduction

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term “reactor” appears, it may be interpreted to mean “non-power reactor” and “radioisotope production facility,” as appropriate. The discussion should be expanded to include the radioisotope production facility, as applicable.

1.2 Summary and Conclusions on Principal Safety Considerations

Areas of Review

The areas of review should include the following information about the radioisotope production facility:

- Facility Description: a general description of the purpose of each feature and the interrelationships between features

- Process Overview: a general description of the different processes at the facility involving licensed material

Acceptance Criteria

The information listed below pertaining to the radioisotope production facility should be added to the acceptance criteria.

- The application presents information at a level of detail that is appropriate for general familiarization and understanding of the proposed radioisotope production facility. This information should be consistent with that presented in the ISA summaries and accident analyses in Chapter 13 of the SAR but may be less detailed.

- The overview describes the relationship of specific facility features to the major processes that will be ongoing at the facility.

- This description includes the building locations of major process components; drawings illustrating the layout of the buildings and structures within the controlled area boundary are used to support the description.

- The application has portions marked to identify any proprietary or sensitive information related to the facility, if applicable.

- The process overview is acceptable if it summarizes the major chemical or mechanical processes involving licensable quantities of radioactive material based, in part, on information presented in the ISA summary. This description should include the building locations of major process components and brief accounts of the process steps.

Review Procedures

The reviewer should confirm that the applicant submitted all information requested in the format and content guide. The information presented in this section is informational in nature and does not require technical analysis. Furthermore, the reviewer should use the information in this section only as background for the more detailed descriptions in later sections of the application.

Evaluation Findings

If the license application provides sufficient information and the regulatory acceptance criteria are appropriately satisfied, the staff will conclude that this evaluation is complete. The reviewer will write material suitable for inclusion in the safety evaluation report prepared for the entire application. The report will include a statement summarizing what was reviewed and why the reviewer finds the submittal acceptable. Specific topics that should be included in the reviewer's comments are given in the current headings 1 through 9 of this section in NUREG-1537.

1.3–1.4

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term "reactor" appears, it is understood to mean "non-power reactor" and "radioisotope production facility." The discussion should be expanded to include the radioisotope production facility, as applicable.

1.5 Comparison with Similar Facilities

Areas of Review

The current content of this section of NUREG-1537 is limited to citations of heterogeneous non-power reactors. In the case of applications for other types of reactors (i.e., AHR designs), other references should be included.

For AHR applications, the following bullets may be added for related facilities:

- HYPO reactor at LANL
- SUPO reactor at LANL
- TRACY reactor at JAERI
- HRE reactor at ORNL
-

Any information about similar radioisotope production facilities or operations should also be included here.

1.6–1.8

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term "reactor" appears, it is understood to mean "non-power reactor" and "radioisotope production facility." The discussion should be expanded to include the radioisotope production facility, as applicable.

2 Site Characteristics

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term “reactor” appears, it may be interpreted to mean “non-power reactor” (homogeneous as well as heterogeneous) and “radioisotope production facility,” as appropriate. The discussion should be expanded to include the radioisotope production facility, as applicable.

2.6 Bibliography

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.16-2008 “Emergency Planning for Research Reactors,” ANS, La Grange Park, IL.

U. S. Nuclear Regulatory Commission, Regulatory Guide (RG) 1.145, Rev. 1, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” February 1983.

3 Design of Structures, Systems, and Components

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term “reactor” appears, it may be interpreted to mean “non-power reactor” and “radioisotope production facility,” as appropriate. The discussion should be expanded to include the radioisotope production facility, as applicable. The technical reviewer should apply the areas of review, acceptance criteria, review procedures, and evaluation findings to both the reactor and the production facility.

3.1–3.4

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term “reactor” appears, it may be interpreted to mean “non-power reactor” and “radioisotope production facility,” as applicable. The discussion should be expanded to include the radioisotope production facility, as applicable.

3.5 Systems and Components

The standard review plan and acceptance criteria of these sections are applicable if, wherever the term “reactor” appears, it is understood to mean “non-power reactor” and “radioisotope production facility.” This section has been divided into two parts, one for the reactor and one for the production facility. The discussion should be expanded to include the radioisotope production facility, as applicable. The guidance in NUREG-1537 for *areas of review, acceptance criteria, review procedures,* and *evaluation findings* should be used for the review of these parts in both Section 3.5a, “Reactor Facility,” and Section 3.5b, “Radioisotope Production Facility.”

3.5a Reactor Facility

NUREG-1537, Part 1, Section 3.5, applies to the reactor facility.

3.5b Radioisotope Production Facility

The applicant should provide the same type of information prescribed in Section 3.5a on the design, construction, and operating characteristics of all safety-related systems and components in the radioisotope production facilities.

3.6 Bibliography

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.2-2009, “Quality Control for Plate-Type Uranium-Aluminum Fuel Elements,” ANS La Grange Park, IL.

American National Standards Institute/American Nuclear Society, ANSI/ANS 15.8-2005, “Quality Assurance Program Requirements for Research Reactors,” ANS, La Grange Park, IL.

4 Reactor Description

The NRC originally wrote NUREG-1537 for heterogeneous reactors, specifying the content of a chapter describing the reactor. To expand the use of NUREG-1537 to AHRs or a radioisotope production facility, additional chapters should be provided per this ISG, as applicable. The result should be one or two chapters with the following titles:

Chapter 4a1, "Heterogeneous Reactor Description"
Chapter 4a2, "Aqueous Homogeneous Reactor Description"
Chapter 4b, "Radioisotope Production Facility Description"

The ISG for each of these options follows.

4a1 Heterogeneous Reactor Description

NUREG-1537, Part 2, Chapter 4, should be used for guidance in preparing this chapter.

4a2 Aqueous Homogeneous Reactor Description

NUREG-1537, Part 2, Chapter 4, should be replaced in its entirety with the guidance below.

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

In following the instructions in this chapter for the AHR, it should be noted that the fuel solution performs the function of the fuel, moderator, and target. In the following sections, any direct reference to a moderator or target applies to designs that might use a solid moderator or target. It should also be noted that no fuel cladding is used in the AHR design, and consequently, the concept of fission product barrier performed by the cladding is no longer valid. The cladding's role is now performed by the reactor vessel and the boundaries of any penetrations (coolant coils, control rod channels, and fuel solution transfer pipes) in the reactor vessel. The primary fission product barrier in a production facility consists of vessels and associated piping that contains the irradiated SNM and fission products (in solid, liquid or gaseous form) during the separation process.

The glossary below contains terms often used when discussing an AHR.

Boiling: Vapor generation due to phase change that results when a fluid is brought to its saturation temperature.

Fission Product Barrier: That portion of the primary system boundary in contact with fission products only (principally, the gas management system boundary).

Fuel Barrier: That portion of the primary system boundary in contact with the fuel solution (principally the vessel, cooling coils, control rod thimbles, piping, and valves).

Neutron Moderator: In an AHR, moderators are materials in the core that consist of light elements (preferable with hydrogen atoms) . Moderators can be either liquid or solid form. Coolant in the cooling coils also contributes to the moderating capacity.

Primary Cooling Systems: Replaces the term “primary coolant system” for an AHR. The primary cooling systems for an AHR are those components and systems that remove heat from the core.

Primary System Boundary: Consists of all structures that prevent the release of fuel, fission gas, or other fission products. For an AHR, this includes the reactor vessel, waste handling tank, pumps, valves, and piping.

Radiolytic Gas Release: The chemical process that generates hydrogen, oxygen, and nitrogen oxides (NOx) from the fuel solution due to dissociation by irradiation.

Reactor Core: In an AHR, consists of that region of the vessel occupied by the solution containing the fission power producing fissile material. In an AHR, the core geometry may change with time due to changes in density and voiding of the solution. The core does not include that part of the fuel solution that may become entrained into the gas.

Reactor Fuel: In an AHR, refers to the dissolved fissionable material and fission products and the solvent they are dissolved in.

Recombiner: Device that recombines hydrogen and oxygen.

Vessel: For an AHR, the structure containing the core.

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 vessel, gas management system, 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 an AHR 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.

4a2.1 Summary Description

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

This section should contain a brief discussion of the way the facility design principles achieve the principal safety considerations. For the items requested, this section should include summaries of the format and content guide and descriptive text, summary tables, drawings, and schematic diagrams.

4a2.2 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 AHR that could be operated and shut down safely.

By reviewing this section, the reviewer gains an overview of the reactor core design and assurance that the SAR describes a complete operable AHR core. Subsequent sections should contain a description and analysis of the specifications, operating characteristics, and safety features of the reactor components. Although cooling systems should be discussed in Chapter 5, "Reactor Coolant Systems," of the SAR, relevant information should also be presented or referenced in this chapter. The information in the following sections should address these systems and components:

- reactor fuel, including the use of the reactor vessel as fuel and fission product barrier
- control rods
- solid neutron moderator (if any) and neutron reflector
- neutron startup source
- core support structures
- gas treatment system

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, as well as a reference to Chapter 7
- TS requirements and their bases, including testing and surveillance, or a reference to Chapter 14, "Technical Specifications"

4a2.2.1 Reactor Fuel

Areas of Review

The information in the SAR should include a reference to the fuel development program and the operational and limiting characteristics of the specific fuel used in the reactor.

The design basis for an AHR should be the maintenance of primary system boundary integrity under any conditions assumed in the safety analysis. Loss of integrity is defined as the escape of any fuel and fission products from the primary system boundary. Since the fuel in an AHR is an aqueous solution without cladding or encapsulation, the primary barrier is the interface surface between the fuel solution, including fission products, and any egress point. During operation, this interface includes the reactor vessel, the gas management system, the cooling coils, the control rod thimbles, and any pipes used for transferring fuel from and to the core. Therefore, the fuel solution must be shown to be compatible with the materials of construction for the fuel barrier (including fission products) for any normal or upset condition. The reviewer should be able to conclude that the applicant has included all information necessary to establish the limiting characteristics beyond which fuel barrier integrity could be lost.

Within the context of the factors listed in Section 4.2 of this review plan, the information on, and analyses of fuel should include the information requested in this section of the format and content guide. Sufficient information and analyses should support the limits for operational conditions. These limits should be selected to ensure the integrity of the fuel barrier. Analyses in this section of the SAR should address mechanical forces and stresses; corrosion and erosion of the fuel barrier, or collection of fission products, decay daughters, or fuel precipitates on the fuel barrier, whether caused by changes in solution chemistry (such as pH, density, pressure, and temperature) or from normal operation; hydraulic forces, including natural convection in the fuel solution; thermal changes and temperature gradients; and internal and external 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 TS.

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 to the bases. Maintaining fuel barrier integrity should be the most important design objective.

- The chemical and physical characteristics of the fuel constituents, including the solvent and any stabilizing additives, should be chosen for compatibility with each other and the anticipated environment, including interaction with the fuel barrier. Consideration should be given to fission product buildup in or precipitation from the homogeneous fuel solution.

- Fuel enrichment should be consistent with the requirements of 10 CFR 50.64, "Limitations on the Use of Highly Enriched Uranium (HEU) in Domestic Non-Power Reactors."

- The fuel operating parameters should take into account characteristics that could limit fuel barrier integrity, such as heat capacity and conductivity, melting, softening, and blistering temperatures of the vessel and cooling coil materials; corrosion and erosion caused by coolant or fuel solution; chemical compatibility of the fuel solution with the fuel barrier; physical stresses from mechanical or hydraulic forces (internal pressures, vibration, and Bernoulli forces); fuel burn-up; radiation damage to the fuel barrier; 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 various phenomena that result in changes to the initial fuel composition and properties should be considered. The submittal should include information on radiolytic gas formation, the transport and collapse and removal of gas, the return of condensate following recombination and condensation of gas or bubbles outside the core vessel, associated pH changes, potential fuel and fission product precipitation, and the addition of fuel and acid, along with the reactivity implications of these items.

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

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

Technical Rationale

The parameters included in the technical review have been identified as important, based on experience with previous operating AHRs, as discussed in References 2 and 3.

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 NRC staff's safety evaluation report:

- The applicant has described in detail the fuel solution to be used in the reactor. The discussion includes the design limits (chemical and physical) and clearly gives the technological and safety-related bases for these limits.

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

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

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

4a2.2.2 Control Rods

Areas of Review

The control rods in an AHR 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 trip 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. 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 reactor trip does not challenge the safety of the reactor or cause any undue strain on any systems or components associated with the reactor.

This section of the format and content guide discusses the areas of review.

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 tripping 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 TS.

- The action of the control rod (manual or automatic) should be such that it does not affect the stability of the core, which has been known to show significant variations in the power level but a return to a stable state following small perturbations (including physical ones from radiolytic gas formation and collapse), if the core is designed within an acceptable power density limit.

- TS should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects and proposes limiting conditions for operations (LCOs) and surveillance requirements, and they should be justified in this section 4a2.2.2 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 to the design bases and can control and shut down the reactor safely from any operating condition.

- Information on scram design for the control rods has been compared with designs at other non-power reactors having similar operating characteristics. Reasonable assurance exists that the reactor trip features designed for this facility will perform as necessary to ensure fuel barrier integrity and to protect the health and safety of the public.

- 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 TS.

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

- The applicant has justified appropriate design limits, LCOs, and surveillance requirements for the control rods and included them in the TS.

4a2.2.3 Solid Neutron Moderator and Neutron Reflector

Areas of Review

In this section of the SAR, the applicant should describe moderators and reflectors and their special features. The fuel solution of the AHR is self-moderating. The information pertinent to this section is, therefore, that for any *solid* moderator that might be added to the AHR design. The core of the aqueous homogeneous reactor is an aqueous fuel solution that self-moderates, surrounded by either a liquid or solid neutron reflector. The primary coolant is kept separate from the fuel material in cooling coils; these provide heterogeneous moderation within the homogeneous core solution. The solid reflectors are chosen primarily for favorable nuclear properties and physical characteristics. Section 4.2.1 of the SAR should contain a description of the relationship of all moderators to the core. Buildup of contaminating radioactive material in the moderator or coolant and reflector during reactor operation should be discussed in Chapter 1, "Radiation Protection Program and Waste Management," of the SAR.

Areas of review should include the following:

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

Section 4a2.5 of the SAR should discuss nuclear characteristics of the moderator.

Acceptance Criteria

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

- The nonnuclear design bases, such as reflector encapsulations, should be clearly presented, and the nuclear bases should be briefly summarized. Nonnuclear design considerations should ensure that the moderator 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 cooling coil and core vessel material and construction methods to ensure primary barrier integrity. If the barrier should fail, the applicant should either show that the reactor can continue to be operated safely until the barrier is repaired or replaced or propose that the reactor be shut down until the barrier 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 provide for removal or replacement of solid moderator or reflector components and systems, if required by operational considerations.

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which describes important design aspects and proposes LCOs and surveillance requirements. The proposed TS should be justified in this section of the SAR.

Review Procedures

The reviewer should confirm that the information on the neutron moderator 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 4a2.5. 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 (state cladding material) if they are located in an environment where coolant or fuel solution 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 or fuel solution 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 materials incorporated into a sound structure that can retain size and shape and support all projected physical forces and weights. Therefore, no unplanned changes to the moderator or reflector would occur that would interfere with safe reactor operation or prevent safe reactor shutdown.

- The applicant has justified appropriate design limits, LCOs, and surveillance requirements for the moderator and reflector and included them in the TS.

4a2.2.4 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
- TS

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 radiation from the reactor during normal operation should not cause heating, fissioning, or radiation damage to the source materials or the holder.
- If the source is regenerated by reactor operation, the design and analyses should demonstrate its capability to function as a reliable neutron startup source in the reactor environment.
- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

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

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

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

- The source and holder operate safely and reliably.

4a2.2.5 Reactor Internals Support Structures

Areas of Review

An AHR fuel core is composed of the homogeneous fuel solution and off gas inside a reactor vessel; the core does not require a support structure beyond the reactor vessel. However, all other 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 barrier integrity depends on stable and reproducible control rod action and coolant flow patterns. 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 support structure for the core components and reactor vessel, 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.

This section of the format and content guide discusses additional areas of review.

Acceptance Criteria

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

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

- The design should consider the methods by which core components (reflector pieces, control rods, and coolant systems, and the fuel transport pipe) are attached to the core support structure. The information should include tolerances for motion and reproducible positioning. These tolerances should ensure that variations will not cause reactivity design bases, coolant design bases, safety limits, or LCOs in the TS to be exceeded.

- The design should consider the effect of the local environment on the material of the support structure. 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 prevent safe reactor operation for the design life of the reactor.

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

- The core of an AHR used for medical isotope production could vary in dimension, based on the purpose of the facility. Fuel can be transferred to and from the core during planned operations; consequently, there are devices to ensure that such operations do not occur inadvertently. The design for a changing core configuration should contain such features as position tolerances, to ensure safe and reliable reactor operation within all design limits, including reactivity and cooling capability. The description should include the interlocks that keep the reactor core configuration from changing while the reactor is critical or while forced cooling is required, if applicable. The design should show how the reactor is shut down if unwanted action occurs.

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation findings

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

- The applicant has described the support system for the 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 support structure includes acceptable guides and supports for other essential core components, such as control rods, nuclear detectors, and neutron reflectors.

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

- The support structure is composed of materials shown to be resistant to radiation damage, coolant or fuel solution 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., reactor trips, coolant flow change, and core motion) through the reactor life cycle.

- The applicant has justified appropriate LCOs and surveillance requirements for the core support structure and included them in the TS.

4a2.3 Reactor Vessel

Areas of Review

The vessel of the AHR is an essential part of the primary fuel system and is the primary fuel barrier (including fission products). The vessel may also provide some support for components and systems mounted to the core supports.

The areas of review are the design bases of the vessel and the design details needed to achieve those bases. This section of the format and content guide discusses the information that the applicant should submit for review.

Acceptance Criteria

The acceptance criteria for information on the reactor vessel should include the following:

- The vessel dimensions should include thickness and structural supports, and fabrication methods should be discussed. The vessel 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 vessel treatment should resist chemical interaction with the fuel solution and be chemically compatible with other reactor components in the primary system. The compatibility between the vessel material and fuel solution should be addressed to prevent fuel solution leakage.

- The dimensions of the vessel and the materials used to fabricate it should ensure that radiation damage to the vessel is minimized, so that the vessel will remain intact for its projected lifetime.

- The construction materials and vessel treatment should be appropriate for preventing fuel solution from corroding the vessel interior and pool water from corroding the exterior.

- A plan should be in place to assess irradiation of and chemical damage to the vessel materials. Remedies for damage or a replacement plan should be discussed.

- All penetrations and attachments to the vessel below the fuel solution level should be designed to avoid malfunction and loss of fuel solution.

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Technical Rationale

Fuel chemistry has been shown to affect corrosion and result in possible loss of vessel integrity, based on the experience from the operation of previous reactors, as described in References 2 and 3.

Review Procedures

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

Evaluation Findings

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

- Information has been provided on gas composition (hydrogen, oxygen, nitrogen (NOx), and fission gases) from radiolytic decomposition of fuel solution, as well as gas handling and condensate return.

- The vessel system can withstand all anticipated mechanical and hydraulic forces and stresses to prevent loss of integrity, which could lead to a loss of fuel solution or other malfunctions that could interfere with safe reactor operation or shutdown.

- The penetrations and attachments to the vessel are designed to ensure safe reactor operation. Safety and design considerations of any penetrations below the fuel solution level include analyses of potential malfunction and loss of fuel solution. The applicant discusses credible fuel spill and leak scenarios in Chapter 13, “Accident Analyses,” Section 13.1.4.

- The construction materials, treatment, and methods of attaching penetrations and components are designed to prevent chemical interactions among the vessel and the fuel solution, pool water, and other components.

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

- The applicant has considered the possibility that fuel solution 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 vessel to ensure sufficient radiation shielding to protect personnel and components. Exposures have been analyzed, and acceptable shielding factors are included in the vessel design.

- The applicant has justified appropriate LCOs and surveillance requirements for the vessel and included them in the TS.

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

4a2.4 Biological Shield

Areas of Review

The radiation shields around non-power reactors are called biological shields and are designed to protect personnel and reduce radiation exposures to reactor components and other equipment. The principal design and safety objective is to protect the employees and the public. The second design objective is to make the shield as thin as possible, consistent with acceptable protection factors. The radioisotope production AHR uses the neutron flux for fissioning and direct production of Mo-99. Access to this radioactive Mo-99 within a few days to a week is necessary because of the relatively short half-life of the material. This necessitates the transfer of the fuel solution to the separations facility at the plant site, and this should be addressed in the shield design. Traditional methods of improving protection factors without increasing shield thickness are to use materials with higher density, higher atomic numbers for gamma rays, and higher hydrogen concentration for neutrons. The optimum shield design should consider all of these.

This section of the format and content guide discusses areas of review.

Acceptance Criteria

The acceptance criteria for 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, "Standards for Protection Against Radiation," and the guidelines of the facility's ALARA (as low as reasonably achievable) program discussed in Chapter 11 of the SAR.

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

- The pool design 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.

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

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements. The applicant should justify the proposed TS in this section of the SAR.

Review Procedures

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

Evaluation Findings

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

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

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

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

- The applicant has justified appropriate LCOs and surveillance requirements for the shield and included them in the TS.

4a2.5 Nuclear Design

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

4a2.5.1 Normal Operating Conditions

Areas of Review

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

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

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

- The information should show a complete, operable 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. Reactivity analyses should include individual and total control rod effects.

- The information should describe anticipated power oscillations and their effects on safety-related equipment and systems. These oscillations should be shown to be self-damping and controllable.

- Anticipated core evolution should account for uranium burn-up; actinide and fission product buildup; changes in fuel solution chemical stability caused by radiolysis, including changes in pH, temperature, pressure, density, and specific heat capacity; and poisons, both from fission products and those added by design, for the life of the reactor. The information should also include an analysis of the total fuel solution volume as a function of total burn-up.

- The analyses should show initial and changing reactivity conditions, control rod reactivity worths, and reactivity worths of reflector units, as well as in-core components 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, including fission gas generation and failure of the gas recombiners.

- 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 the loss of fuel barrier integrity or an uncontrolled addition of reactivity.

- The information should include calculated core reactivities for the possible and planned configurations of the control rods. The reactivity impacts of radiolytic gas and void formation, fission product gas removal, fuel solution and acid addition, and condensate return to the core should be provided. If only one core configuration will be used over the life of the reactor, the applicant should clearly indicate this. The limiting core configuration during reactor life should be indicated. This information should be used for the analyses in Section 4.6 of the SAR. The information should also include reactivities for fuel solution storage and handling outside the reactor, fuel transport to and from the core, and the effects of core recycling after isotope removal processing.

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Technical Rationale

Power oscillations in AHRs are expected and usually are self-limiting because of the large negative reactivity feedback coefficients. It is necessary to ensure that oscillations are bounded for proper operation of the reactor, based on the operation of previous AHRs found in References 2 and 3.

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 burn-up; plutonium buildup; the buildup and removal of fission products, both in solution and in the gas management system; fuel solution condensate return to the core; fuel solution and acid addition; and the use of poisons, as applicable.

- The reactivity analyses include the reactivity values for the in-core components, such as control rods or cooling coils, and the ex-core components, such as the reflector and pool. The assumptions and methods have been justified.

- The analyses address the steady power operation and kinetic behavior of the reactor and show that the dynamic response of the control rods and instrumentation is designed to prevent uncontrolled reactor transients.

- The analyses show that any in-core components that could be flooded or voided could not cause reactor transients beyond the capabilities of the instrumentation and control systems to prevent fuel damage or other reactor damage. This also should include failure of radiolytic gas recombiners and subsequent pressure pulses resulting from deflagration or explosions of radiolytic gas.

- 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 LCOs and surveillance requirements for minimal operating conditions and included them in the TS. The applicant has also justified the proposed TS.

4a2.5.2 Reactor Core Physics Parameters

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 an AHR is to produce isotopes for use, while not posing an unacceptable risk to the health and safety of the public. By proper design (sufficiently low power density), the reactor will operate at steady power; however, power oscillation in AHRs is expected, and the reactor systems will be able to terminate or mitigate transients without reactor damage.

Areas of Review

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.

This section of the format and content guide further discusses the areas of review.

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 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 burn-up should not change the conclusions about reactor protection and safety, nor should they void the validity of the analyses of normal reactor operations.

- The methods and assumptions for calculating the various neutron flux densities should be validated by comparisons with results for similar reactors. Uncertainties and ranges of accuracy should be given for other analyses requiring neutron flux densities, such as fuel burn-up, thermal power densities, radiolytic gas production, control rod reactivity worths, and reactivity coefficients. This should include a description of the method of calculating and verifying the burn-up and the fuel composition after isotope removal. It also should include methods to analyze gas evolution and the generation of void spaces and predict their reactivity effects.

- TS, if required, should be proposed according to the guidance in Chapter 14 of the format and content guide, which proposes LCOs and surveillance requirements, and should be justified in this section of the SAR.

Review Procedures

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

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions, which 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 burn-up 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 LCOs and surveillance requirements for the reactor core physics parameters and included them in the TS. The applicant has also justified the TS.

4a2.5.3 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.

This section of the format and content guide discusses the areas of review.

Acceptance Criteria

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

- All operational requirements for excess reactivity should be stated, analyzed, and discussed. These could pertain to at least the following:

  - temperature coefficients of reactivity
  - fuel burn-up between reloads or shutdowns
  - void coefficients
  - xenon and samarium override
  - overall power coefficient of reactivity if not accounted for in the items listed above
  - fuel processing, handling, and recycling, and implications for reactor safety
  - effects of experiments

- Credible inadvertent insertion of excess reactivity should not damage the reactor or fuel barrier; 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 should be performed that assumes that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region. The analysis should show that the reactor would not be damaged and fuel barrier integrity would not be lost. Chapter 13 of the SAR should analyze reactivity additions under accident conditions.

- An analysis should be performed that examines reactivity, assuming that the reactor is operating under its maximum licensed conditions, normal electrical power is lost, and the control 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 and all transient poisons, such as xenon, are reduced, with consideration for the most reactive core loading.

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

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

- The effects of surface frothing as an intermittent reflector or moderator should be considered.

- The applicant should propose and justify TS for safety limits, LSSSs, LCOs, and surveillance requirements, as discussed in Chapter 14 of the format and content guide.

Review Procedures

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

Evaluation Findings

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

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

- The discussion of limits on excess reactivity shows that a credible rapid 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 barrier integrity. Therefore, the information demonstrates that the proposed amount of reactivity is available for normal operations but that it would not cause unacceptable risk to the public from a transient.

- The definition of the shutdown margin is negative reactivity obtainable by control 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 (the reviewer should insert the margin specified in the SAR) in the TS. 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 LSSSs, which are acceptable.

4a2.6 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 barrier integrity will not be lost under any reactor conditions, including accidents. In the case of a low-power AHR, there is no concern about damaging fuel; however, there is concern about damaging the fuel barrier (and fission product barriers).

Since the fuel solution is free to move in an aqueous form, the temperature within the fuel can more readily equalize; however, the power shape may still cause some hot spots, which may lead to instability and ultimately fuel and fission product precipitation. 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 AHR design may contain a flow loop that circulates radiolysis gas, fission gases, water vapor, and a cover gas. The reviewer needs to determine the constituents in the bubbly mixture and cover gas. The capacity of recombiners and condensers in the system may limit achievable stable and safe operation. The reviewer needs to determine if the makeup and flow rate of the circulating mixture is within the design limits of any recombiners for radiolysis gases or condensers of water vapor. The reviewer should also ensure that any sources and sinks of energy in the flow loop are within the design capacities of any heat exchangers in the loop.

This section of the format and content guide discusses the areas of review.

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 barrier integrity under all analyzed conditions. The discussion should include the consequences of these conditions and justification for the alternatives selected. It should also include the limiting power density to offset the onset of instability following perturbation to the system (including from radiolytic gas generation). These criteria could include the following:

  – There should be no coolant flow instability in any cooling coil that could lead to a significant decrease in fuel cooling. This can be ascertained using a suitable onset-of-flow-instability correlation.

  – The departure-from-nucleate-boiling ratio should be no less than 2.0 along any coolant coil.

- 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 TS.

- 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 barrier 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 jeopardizing safe reactor shutdown. Loss of normal electrical power should not change this criterion. These limits should be based on the thermal-hydraulic analyses and appear in the TS.

- For AHRs, undercooling may change the pH of the system, resulting in fuel or fission product plate-out or precipitation; this should be considered in the thermal-hydraulic design.

- The gas treatment system, including recombiners, will contain fission product gas and hazardous chemicals. Since this forms part of the fuel barrier, this section should consider the associated cooling systems and show their ability to maintain their functions and fuel barrier integrity under normal and abnormal operations.

- The pool water surrounding the reactor vessel is expected to provide some heat removal during steady-state operation. An analysis of the effects of loss of pool cooling should show that it would not affect fuel barrier (vessel) integrity under normal and abnormal operations.

Technical Rationale

Previous experience with AHRs has indicated the importance of the interrelationship of temperature of the fuel solution, chemical pH, and radiolytic gas recombination rates, as described in References 2 and 3.

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, pH, and peaking). The basic approach is an audit of the SAR analyses, but the reviewer may also perform independent calculations to confirm SAR results or methods.

Evaluation Findings

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

- The information in the SAR includes the thermal-hydraulic analyses for the reactor. This includes radiolytic gas generation, void formation and collapse, and fuel solution mixing, which might minimize precipitation with the fuel volume or frothing on the fuel solution surface, and a subsequent core transient. 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 solution is specified for this reactor. The analysis has considered the various approaches and systems for heat removal, such as the cooling coils, the pool, and the gas management system. The analyses give the limiting conditions of the features that ensure fuel barrier integrity.

- Safety limits and LSSSs are derived from the thermal-hydraulic analyses. The values have been justified and appear in the TS. The thermal-hydraulic analyses on which these parameters are based ensure that overheating or overcooling during any operation or credible event will cause neither a loss of fuel barrier integrity and unacceptable radiological risk to the health and safety of the public nor fuel or fission product plate-out or precipitation that could lead to a loss of fission product integrity. The analysis includes methods for calculating the induced natural convection within the homogeneous fuel solution.

4a2.7 Gas Management System

Areas of Review

This section of the SAR should contain the design information on all components of the gas management system. The design information should be presented in drawings, diagrams, text, and analysis in sufficient detail for the staff to understand the flow of evolved gases and fission products from their generation in the reactor core to their ultimate release. Using this information, the staff should determine whether there is reasonable assurance that the gas management system can prevent a hydrogen deflagration or detonation hazard; contain hazardous chemicals and volatile fission products until they can be released safely, in accordance with environmental release criteria; and withstand any pressure transients within the reactor core.

In evaluating the analysis demonstrating these capabilities, the staff should ensure that these criteria can be met for the maximum power density that is considered credible during power oscillations. The applicant should justify the maximum fission product and radiolytic gas generation rates during power oscillations.

This section of the format and content guide discusses the areas of review.

Technical Rationale

Areas of review, acceptance criteria, and evaluation findings are all dictated by five hazards: an inadvertent criticality outside the reactor core, a radiolytic gas deflagration or detonation, an NOx release, a release of gaseous fission products, and an increase in the pressure in the headspace over the core. Although the reactor will operate in a steady-state mode, power oscillations may be possible. Therefore, the design must be sufficiently robust to sustain fission product and NOx generation, heat generation, and pressures that will occur at peak power. The dynamics of criticality accidents show that a sudden spike in power of several orders of magnitude can occur in solution systems. This can occur when there is a rapid reactivity insertion that causes the solution to go prompt critical. The spike is generally terminated by the negative reactivity effect of void formation caused by radiolytic gas generation. The actual first spike yield and total fission yield during accidents and planned critical excursions can vary widely, so fairly conservative assumptions should be made concerning the assumed dynamics during a prompt critical excursion.

Acceptance Criteria

The design of the gas management system should be found acceptable if it meets the following acceptance criteria:

- The geometry of all equipment and piping should be favorable (e.g., subcritical when filled with optimally moderated reactor core solution).

- If any portions of the equipment or piping are not favorable geometry, the applicant's analysis should demonstrate that no single failure can result in a criticality outside the core.

- Monitoring should be provided periodically for the long-term accumulation of fissionable material entrained in the system.

- The radiolytic gas recombiner must be capable of preventing a hydrogen deflagration or detonation anywhere within the gas confinement boundary, especially in the reactor vessel.

- The cooling system for the recombiner must be sufficient to dissipate the reaction heat.

- The construction materials must be compatible with the chemical environment (e.g., NOx gases), such that corrosion cannot lead to a loss of confinement.

- The maximum pressure resulting from heat and radiolytic gas generation must not exceed the design pressure for the system, unless redundant pressure relief features are described.

- The maximum release of fission gases must not exceed applicable regulatory criteria.

- The maximum release of hazardous chemicals must not exceed applicable regulatory criteria. (This should include any potential effect on workers in the production facility).

- Monitoring should be provided for concentrations of hazardous chemicals and fission products to detect buildup and leaks.

Chapter 5 contains acceptance criteria for any credited cooling function of the gas management system.

Technical Rationale

Most of these are events that can result in release pathways through the loss of confinement (e.g., by deflagration or detonation, corrosion, or overpressurization). The exception to this is criticality, which will result in the generation of more fission products (although they will be small compared to those generated during normal reactor operations). Criticality should not be allowed outside the reactor vessel, because there are no means to control it or adequately protect personnel outside such an environment. Ideally, all equipment that is connected to the reactor vessel should have favorable geometry (i.e.,the contained SNM will always have a sub-critical multiplication factor), although at some point a connection might need to be made to non-favorable geometry. Maintaining solution and aerosolized fuel within the reactor core (ideally) or the favorable geometry part of the gas management system (as an anticipated upset) is crucial. For chemical releases, both the effects of NOx on personnel and on equipment must be considered.

Review Procedures

The reviewer should confirm that the design of the gas management system and the associated analysis are sufficient to provide reasonable assurance of safe operation of the reactor and compliance with all applicable chemical and radiological release criteria.

Evaluation Findings

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

- The applicant has described the system in sufficient detail to prevent criticality outside the reactor vessel, caused by the entrainment of uranium in the gas, slow accumulation over time, or backflow of solution from the reactor vessel.

- The applicant has described the system in sufficient detail to prevent the occurrence of a radiolytic hydrogen deflagration or detonation that could breach confinement and result in exceeding the applicable regulatory limits on hazardous chemical or fission product releases.

- The applicant has designed the system to withstand the maximum pressure that could occur during credible power oscillations, so as to avoid breaching confinement and exceeding applicable regulatory limits.

- The applicant has designed the system to allow for control of the reactor during possible explosions or increases in pressure.

- The applicant has designed the system to be compatible with the chemical environment to which it will be exposed, avoiding corrosion that could result in a release of hazardous chemicals or fission products exceeding applicable regulatory limits.

- The applicant has designed sufficient surge capacity to contain hazardous chemicals and allow for the decay of fission products until they can be released in accordance with applicable regulatory limits.

Technical Rationale

These conclusions are driven by the consideration of hazards discussed previously.

4a2.8 References

1. "Homogeneous Aqueous Solution Reactors for the Production of Mo-99 and Other Short Lived Radioisotopes," IAEA—TECDOC-1601, International Atomic Energy Agency, September 2008.

2. Fluid Fueled Reactors "Part 1 Aqueous Homogeneous Reactors," James A. Lane, editor, Addison Wesley, 1958, http://moltensalt.org/references/static/downloads/pdf/.

3. C. Cappiello, T. Grove, and R. Malenfant, "Lessons Learned from 65 Years of Experience with Aqueous Homogenous Reactors," LA-UR-10-02947, Los Alamos National Laboratory, May 2010.

4b Radioisotope Production Facility Description

The structures, systems, and components that should be discussed in this section shall include processes containing SNM, particularly when material is separate from the reactor.

4b.1 Facility and Process Description

This section of the SAR expands the reactor summary description to include an isotope production facility. It should include the principal safety considerations that were factored into the design, construction, and operation. The design bases and functions of the systems and components should be presented in sufficient detail to allow a clear understanding and to ensure that the facility can be operated for its intended purpose and within regulatory limits for ensuring the health and safety of the operating staff and the public. Drawings and diagrams should be provided as necessary to allow a clear and general understanding of the physical facility features and of the processes involved.

Areas of Review

The summary should include the name, amount, and specifications (including chemical and physical forms) of the SNM that will be in process. The license application should include a list of byproduct materials (identity and amounts) in the process solutions, finished products, and wastes from the process.

It should also include a detailed description of the design and construction of the equipment that will be used while processing SNM outside the reactor. It should include enough detail to identify materials that may have moderating, reflecting, or other nuclear-reactive properties.

Acceptance Criteria

The summary should be found acceptable if it includes the following:

- The summary describes the chemical and physical forms of SNM in process, including the maximum amounts of SNM in process in various building locations.

- The application presents a summary description of the raw materials, byproducts, wastes, and finished products of the facility. This information should include data on expected levels of trace impurities or contaminants (particularly fission products or transuranic elements) characterized by identity and concentration.

- The application contains a general description of the design basis and implementation of any criticality safety features of the production facility per the requirements in 10 CFR Part 70 for establishing and maintaining a nuclear criticality safety program.

- The application contains a description of the design basis and implementation of any hazardous chemical safety features of the production facility per the requirements of 10 CFR Part 70 for establishing and maintaining a hazardous chemical safety program.

Review Procedures

The information submitted by the applicant in this section is informative in nature and requires no technical analysis. In addition, the reviewers use the information in this section only as background for the more detailed descriptions in later sections of the application. Therefore, the primary reviewer ascertains whether the descriptive information is consistent with the information presented in the accident analysis and emergency management plan.

Evaluation Findings

If the license application includes sufficient information to provide a general understanding of the production facility, the radioisotope production process and assurance that the regulatory acceptance criteria can be achieved, the staff should conclude that this evaluation is complete.

4b.2 Processing Facility Biological Shield

Biological shields are designed to protect personnel and minimize radiation exposures. The principal design objective is to protect the workers and the public. This section should present the design bases and a detailed description of the biological shield.

Areas of Review

The guidance provided in NUREG-1537, Part 2, Section 4.4, is applicable, provided that any reference to a reactor facility is understood to mean a radioisotope production facility, as appropriate..

Acceptance Criteria

In addition to the acceptance criteria provided in NUREG-1537, Part 2, Section 4.4, applicable to a radioisotope production facility, the following criteria should be considered:

- The shield design should include a detailed description of the design and construction of the biological shield in which radiochemical processes will be conducted. The shielding design basis, including any calculations that were used to prescribe the required form and substance of the shield, should be provided. It should also describe the functional design of the biological shield, showing entry and exit facilities for products, wastes, process equipment, and operating staff.

- The objective of the shield design should be to ensure that the projected radiation dose rates and accumulated doses do not exceed the limits of 10 CFR Part 20 and the guidelines of the facility's ALARA program.

- The application should include a detailed description of the ventilation system for the biological shield structure, including (1) the design basis and function, (2) the design and location of vent ducting, filters, and fans, (3) details on vent system operating limits under both normal and emergency operating conditions, and (4) the design basis and function of all filtering and sequestration systems provided to control release of particulate and gaseous airborne radioactive contaminants to the environment under normal and emergency conditions of operation.

All of the essential physical and operational features of the biological shield that are required to prevent the release of radioactive material and to maintain radiation levels below applicable radiation exposure limits prescribed in 10 CFR Part 20 for the protection of the staff and the public should be identified and included in the proposed technical specifications in Chapter 14.

Review Procedures

The guidance in the "*Review Procedures*" part of NUREG-1537, Part 2, Section 4.4, is applicable.

Evaluation Findings

The guidance in the "*Review Procedures*" part of NUREG-1537, Part 2, Section 4.4, is applicable.

4b.3 Radioisotope Extraction System

This section of the SAR should provide the design and detailed description of the radioisotope extraction process. The specific information required by Part 1, “Standard Format and Content,” of this ISG should be the subject of this review.

Areas of Review

The information should provide a complete description, including diagrams and drawings, as necessary, in sufficient detail to give a clear understanding of the extraction process and how the process can be performed safely within regulatory limits.

Acceptance Criteria

The application should provide processing details such as the following:

- description of the SNM in terms of physical and chemical form, volume in process, and radioactive inventory in process.

- description of the sequence of radioisotope extraction and the time increments involved.

- description of the processing apparatus, including tanks, piping, separation columns, reagent vessels, materials of construction, and process monitoring or control equipment.

- description of any required criticality control measures that are designed into the process systems and components. A detailed description of the Criticality Safety Program (CSP) is given in Chapter 6b.3, Part 2 of this ISG.

- description of any hazardous chemicals that are used or that may evolve during the process along with a description of provisions to protect the staff and the public from exposure.

- all of the essential physical and operational features of the radioisotope extraction system that are required to prevent the release of radioactive material and to maintain radiation levels below applicable radiation exposure limits prescribed in 10 CFR Part 20 and the chemical exposure limits prescribed in 10 CFR 70.61 for the protection of the staff and the public should be identified and included in the proposed technical specifications in Chapter 14.

Review Procedures

The primary reviewer should ascertain whether the descriptive information presented is sufficient to satisfy the objective of providing a clear understanding of the processes and whether it is consistent with the information presented in the accident analysis, engineered safety features (ESFs), and TS that are included in other chapters of the application.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions for the safety evaluation report:

- The process description(s) in the SAR provide a detailed account of the SNM in process along with any included fission-product radioactivity. The description of the post-irradiation processing after the fuel is removed from the reactor gives a clear understanding that these operations can be conducted safely in this facility.

- The processing facilities and apparatus have been described in sufficient detail to provide confidence that the SNM and byproduct material can be controlled throughout the process so that the health and safety of the public will be protected.

- The criticality control measures provided throughout the process are in accordance with double-contingency principle and the processing facility provides suitable defense-in-depth for the contained processes.

- Sufficient technical specifications and ESFs have been developed that provide safe margins for all safety-related process variables.

4b.4 SNM Processing and Storage

The contents of this section describe the processing components and procedures involved in handling, processing, and storing SNM.

4b4.1 Processing of Irradiated Special Nuclear Material

Areas of Review

This section of the SAR should contain information about processes with irradiated SNM as follows:

- The summary should specify the maximum amounts of SNM in storage or in process in various facility locations. It should describe the chemical and physical forms of SNM in process. The application presents a summary description of the process(es). This information should include data on expected levels of radioactivity, broken down by radionuclide (particularly volatile and long-lived fission products and transuranic elements). The radionuclide inventory should be projected with decay time and tabulated at various times throughout the process. The description should indentify points in the process where major separations are performed and describe the pathway of the separated radionuclides or other constituents.

- The application should provide a clear description of the process systems and components to allow a good understanding that the facility can be operated safely within regulatory limits. In particular, this summary should identify the proposed possession at the facility of any moderator or reflector with special characteristics, such as beryllium or graphite. The processing materials should be compatible with the process material contained to withstand the effects of corrosion and radiation. The processing system should be designed to manage any fission-product or radiolysis gases that evolve in the process.

- The application should include a detailed description of any required criticality control measures that are designed into the process systems and components

- The application should include a description of any hazardous chemicals that are used or that may evolve during the process along with a description of provisions to protect the staff and the public from exposure.

Acceptance Criteria

The application should provide processing details such as:

a. Description of the SNM in terms of physical and chemical form, volume in process, and radioactive inventory in process.

b. Description of the sequence of process steps and the time increments involved.

c. Description of the processing apparatus including any piping, separation columns, reagent vessels, materials of construction, process monitoring or control equipment.

d. Description of auxiliary equipment or apparatus that is required to remove or control heat and volatile gases that could evolve from the process.

e. The application includes a detailed description of any required criticality control measures that are designed into the process systems and components. A detailed description of the Criticality Safety Program (CSP) is given in Chapter 6b.3, Part 2 of this ISG.

f. The application includes a description of any hazardous chemicals that are used or that may evolve during the process along with a description of provisions to protect the staff and the public from exposure.

All of the essential physical and operational features of the irradiated SNM processing system that are required to prevent the release of radioactive material and to maintain radiation levels below applicable radiation exposure limits prescribed in 10 CFR Part 20 for the protection of the staff and the public should be identified and included in the proposed technical specifications in Chapter 14.

Review Procedures

The primary reviewer should ascertain whether the descriptive information presented is sufficient to satisfy the objective of providing a clear understanding of the processes and that it is consistent with the information presented in the accident analysis, engineered safety features and technical specifications that are included in other chapters of the application.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions for the safety evaluation report:

- The process description(s) in the SAR provides a detailed account of the SNM in process along with any included fission-product radioactivity. The process descriptions for reconditioning fuel for continued use, for disposal as waste or for some other appropriate purpose are sufficient to provide a clear understanding that these operations can be conducted safely in this facility.

- The processing facilities and apparatus have been described in sufficient detail to provide confidence that the SNM and byproduct material can be controlled throughout the process so that the health and safety of the public will be protected.

- The criticality control measures provided are in accordance with double-contingency principle and the processing facility provides suitable defense-in-depth for the contained processes.

- Sufficient technical specifications and ESFs have been developed that provide safe margins for all safety-related process variables.

4b.4.2 Processing of Unirradiated Special Nuclear Material

Operations with unirradiated SNM in the form of reactor fuel are generally addressed in NUREG-1537, Chapter 9, “Auxiliary Systems.” This discussion may be located in Chapter 9 or in this section of Chapter 4. This ISG presents it in Section 4b.4.2 in the interest of maintaining the continuity of discussion of all operations with SNM in the radioisotope production facility.

Regarding new fuel entering the facility, the application should provide a narrative describing all operations involving receipt, qualification, movement, storage, and preparation for use in the reactor. The application should explain the technical basis for the design and implementation of each operation.

Areas of Review

Areas of Review are prescribed in this section of Part 1, format and content of this ISG

Acceptance Criteria

The reviewer should ascertain that the application includes the information prescribed in Part 1 of this section of the ISG as follows:

- A description of all operations involving SNM before it is used as fuel in the reactor (e.g., receipt, storage, transfer and preparation for use in the reactor).

- A description of the detailed procedures used in each operation including a description of the quantity, physical form and chemical form of the SNM involved in each operation and enough detail to enable development and analysis of potential accident sequences in Chapter 13.

- The location of each operation with SNM.

- A description of the equipment employed in each operation.

- A description of any criticality safety features and management measures per the requirements of Chapter 6b, Parts 1 & 2 of this ISG.

- A description of any preventive or mitigative features and management measures to control the use of hazardous chemicals that are used with or evolve from operations with SNM. (Refer to ISG section 12.1.6, "Production Facility Safety Program").

All of the essential physical and operational features of the unirradiated SNM processing system that are required to prevent the release of radioactive material and to maintain radiation levels below applicable radiation exposure limits prescribed in 10 CFR Part 20 for the protection of the staff and the public should be identified and included in the proposed technical specifications in Chapter 14.

Review Procedures

The primary reviewer should ascertain whether the descriptive information presented is sufficient to satisfy the objective of providing a clear understanding of the processes and that it is consistent with the information presented in the accident analysis, engineered safety features and technical specifications that are included in other chapters of the application.

Evaluation Findings

This section of the SAR should contain sufficient information to support the following types of conclusions for the safety evaluation report

- The process description(s) in the SAR provide a detailed account of the SNM in process. Each operation with SNM in receipt, transport, storage or preparation for use is described in sufficient detail to show that there is reasonable assurance that these operations can be conducted safely.

- The storage, transport and processing facilities and apparatus have been described in sufficient detail to provide confidence that the SNM can be controlled throughout the process so that the health and safety of the public will be protected.

- The criticality control measures provided are in accordance with double-contingency principle and the processing facility provides suitable defense-in-depth for the contained processes.

- Sufficient technical specifications and ESFs have been developed that provide safe margins for all safety-related process variables.

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5 Reactor Coolant Systems

This chapter of NUREG-1537 was written for heterogeneous reactors, specifying the content of a chapter describing the reactor. To expand the use of NUREG-1537 to AHRs or a radioisotope production facility, additional chapters should be provided according to this ISG, as applicable. The result should be one or two chapters with the following titles:

Chapter 5a1, "Heterogeneous Reactor Coolant System "
Chapter 5a2, "Aqueous Homogeneous Reactor Cooling System"
Chapter 5b, "Radioisotope Production Facility Cooling System"

This document specifies the ISG for each of these options below.

5a1 Heterogeneous Reactor Coolant Systems

NUREG-1537, Part 2, Chapter 5 should be used for guidance in preparing this chapter.

5a2 Aqueous Homogeneous Reactor Cooling System

Replace Chapter 5 of NUREG-1537, Part 2, in its entirety with the following guidance: This chapter contains guidance for evaluating the design bases, descriptions, and functional analyses of the AHR cooling systems. The principal purpose of the cooling system is to safely remove the fission and decay heat from the fuel and dissipate it to the environment. In an AHR, the primary cooling systems are those components and systems that remove heat from the core, where the core consists of that region of the vessel occupied by the solution containing the fission power producing fissile material. In an AHR, the core geometry may change with time, because of changes in density and voiding of the solution. The core does not include that part of the fuel solution that may become entrained into the cover gas or the vapor above the core.

For an AHR, the applicant should describe and discuss, in this chapter, all systems that remove and dispose of the heat from the reactor. The design bases of the core cooling systems for the full range of normal operation should be derived from Chapter 4 of the SAR.

For an AHR, the primary cooling system removes heat from the core by being in direct contact with the fuel solution through a structural barrier or by the heat removal accompanying the fluid mass transport out of and into the fuel solution (i.e., evaporation and condensation); it is the cooling system that removes the largest fraction of the core heat.

The AHR may include other cooling systems that remove a significant fraction of the total heat produced by the core and fission products, however. Cooling systems may include core heat removal by radiolysis gas management systems or passive heat removal through the reactor vessel to a surrounding pool. Reactors operating at very low power levels may be cooled solely by passive heat removal through the vessel wall.

In addition, the "Secondary Coolant Systems" for an AHR are defined as those systems and components that transfer heat from the primary cooling systems to the environment or intermediate heat sink(s). Secondary cooling systems may consist of additional heat exchangers and pumps to circulate the coolant. The secondary coolant system is that which removes the heat from the primary cooling system to the environment.

In this chapter, the applicant should identify and discuss reactor cooling systems, including auxiliary and reactor core subsystems, that remove heat from the reactor core, as well as major components. The description should include, for example, information on core cooling coils that might be the primary cooling system, and the partition of heat removal by additional reactor cooling systems that remove core heat directly, such as the radiolysis gas management system, or passively, through the reactor vessel. These additional reactor cooling systems should be summarized in Section 5a2.1 and discussed in detail in Chapter 4, "Reactor Description," if reactor core systems such as gas management systems, are involved. Chapter 9, "Auxiliary Systems," should discuss details of auxiliary systems using coolant, other than the primary cooling system, such as passive core cooling by the pool surrounding the vessel.

This chapter should also describe and discuss all auxiliary systems and subsystems that use and contribute to the heat load of either the primary or secondary cooling system. Chapter 9, "Auxiliary Systems," should discuss any auxiliary systems using coolant from other sources, such as building service water. The design bases of any features of the core cooling system designed to respond to potential accidents or to mitigate the consequences of potential accidents should be derived from the analyses in Chapter 13, "Accident Analyses." These features should be summarized in this chapter and discussed in detail in Chapter 6, "Engineered Safety Features," of the SAR. In this chapter, the applicant should discuss and reference the TS needed to ensure operability consistent with the assumptions in the SAR analyses.

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

5a2.1 Summary Description

In this section, the applicant should give a brief description of reactor cooling systems, including the supplementary core heat removal pathways, summarizing the principal features. Information should include the following:

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

The applicant should summarize the principal features of the reactor cooling systems unique to the AHR. In addition to the primary cooling system, other means of heat transport from the core should be described, including the corresponding amount of heat transported from the core and the fraction of total core heat removed. These are the supplementary core heat removal pathways.

5a2.2 Primary Cooling System

Areas of Review

For an AHR, the term “primary cooling system” replaces the term “primary coolant system.” The primary cooling systems for an AHR are those components and systems that remove heat from the core.

The primary cooling system is a key component in the overall design and should have the capability to do the following:

- remove the fission and decay heat from the core during normal reactor operation and decay heat during reactor shutdown

- transfer the heat to a secondary cooling system for controlled dissipation to the environment

- maintain high water quality to limit corrosion of cooling coils, control and safety rods, reactor vessel or pool, and other essential components

- prevent uncontrolled leakage or discharge of contaminated coolant to the unrestricted environment

The basic requirements for these functions are generally derived and analyzed in other chapters of the SAR. In this chapter, the applicant should describe how the cooling system provides these functions. Section 5a2.2 of the format and content guide discusses specific areas of review for this section.

The liquid fuel solution in an AHR is expected to be highly corrosive and contain mobile radioactive fission product species. In addition, no fuel cladding barrier exists for the fuel solution, as is characteristic of solid fuel elements in conventional non-power reactors. Therefore, it may be appropriate to consider solid material barriers that isolate the primary coolant from the fuel (such as cooling tube walls) as analogous to fuel cladding. Because this could affect the design of AHR cooling systems, consideration should be given to the following:

- construction materials of components and fabrication specifications of safety-related components as they relate to corrosion resistance to the fuel solution.

- coolant quality requirements for operation and shutdown conditions, given the presence of liquid fuel solution on the core side of components. (Due to the fluid and potentially volatile nature of the fuel and its constituents, prevention of corrosion on either side of the cooling system components is of major concern).

- locations, designs, and functions of essential components, such as cooling coils located in the reactor vessel, as these components ensure that the primary cooling system is operable and that uncontrolled loss or discharge of fuel solution from the fuel core tank into the primary cooling system does not occur

Section 5a2.2 of the format and content guide discusses specific areas of review for this section. For an AHR, the applicant should provide information in this section on the reactor cooling systems unique to these principal features of AHRs.

Acceptance Criteria

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

- Chapter 4 of the SAR should contain analyses of the reactor core, including coolant parameters necessary to ensure removal of heat from the core. Safety limits (SLs) and LSSSs and limiting conditions for operation (LCOs) should be derived from those analyses and be included in the TS. Examples of cooling system variables on which LSSSs and LCOs may be established are maximum thermal power level for operation, minimum and maximum coolant temperatures, minimum and maximum coolant flow rates, and coolant pressure range. The analyses in this section should show that the components and the functional design of the primary cooling system will ensure that no LSSS will be exceeded through the normal range of reactor operation. The analyses should address forced flow or natural-convection flow in the primary cooling system, as applicable.

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

- The descriptions and discussions should show that sufficient instrumentation, coolant parameter sensors, and control systems are provided to monitor and ensure stable coolant flow, respond to changes in reactor power levels, and provide for a rapid reactor shutdown in the event of loss of cooling. There should also be instrumentation for monitoring the radiation of the primary coolant, because elevated radiation levels could indicate a loss of primary coolant barrier integrity. There should be routine sampling for gross radioactivity in the coolant and less frequent radioactive spectrum analysis to identify the isotopes and concentrations found in the coolant. This spectrum analysis may also detect primary cooling system integrity failure at its earliest stages.

- The primary coolant should provide a chemical environment that limits corrosion of the primary coolant barrier, control and safety rod surfaces, reactor vessels or pools, and other essential components. Other requirements for water purity should be analyzed in the SAR, and proposed values of conductivity and pH should be justified. Experience in non-power reactors has shown that the primary water conditions, electrical conductivity ≤5 μmho/cm (micro-mho is a measure of electrical conductivity in water, and the reciprocal of micro-ohm — both terms are a measure of water purity) and pH between 5.5 and 7.5 can usually be attained with good housekeeping and a good filter and demineralizer system. Chemical conditions should be maintained, as discussed in Section 5a2.4 of this standard review plan.

- Radioactive species, including nitrogen-16 and argon-41, may be produced in the primary coolant. Additional radioactivity may occur as a result of neutron activation of coolant contaminants and fission product leakage from the fuel. Provisions for limiting radiological hazards for personnel should maintain potential exposures from coolant radioactivity below the requirements of 10 CFR Part 20 and should be consistent with the facility's ALARA program. To ensure that facilities or components for controlling, shielding, or isolating nitrogen-16 are acceptable, potential exposures should not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility's ALARA program. Section 5a2.6 of this standard review plan discusses the nitrogen-16 control system.

- Argon-41 is another radionuclide that can be produced in the primary cooling system. Because it may be an important radionuclide released to the environment during a reduction-in-cooling event, special analyses and discussion of its production and consequences should be provided in Chapter 11 of the SAR. If any special design or operational features of the primary cooling system modify or limit exposures from argon-41, they should be discussed in this section of the SAR. This discussion should demonstrate that any facilities or components added to the primary cooling system to modify argon-41 releases can limit potential personnel exposures to the values found acceptable in Chapter 11.

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

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

- If contaminated coolant were lost from the primary cooling system, the design and analyses should ensure that potential personnel exposures and uncontrolled releases to the unrestricted environment do not exceed acceptable radiological dose consequence limits derived from the accident analyses. The radiological consequences from the contaminated coolant should be discussed in Chapter 11 and summarized in this section of the SAR. Necessary surveillance provisions should be included in the TS.

- The primary coolant should provide a chemical environment that limits corrosion of primary coolant barrier material and components of the primary cooling system, given the presence of liquid fuel solution on the core side of the primary coolant barrier material. The barrier should prevent the fuel solution from contaminating the primary coolant, and the primary coolant from diluting the fuel solution, with accompanying reactivity and chemistry effects.

- The design of the primary cooling system components ensures that the system is operable and that uncontrolled loss or discharge of fuel solution from the fuel core tank into the primary system does not occur. The system design should avoid uncontrolled release of fuel solution to the primary coolant. If contaminated coolant were lost from the primary cooling system, the design and analyses should ensure that potential personnel exposures and uncontrolled releases to the unrestricted environment do not exceed acceptable radiological dose consequence limits derived from the accident analyses. The designs and locations of such essential components as fuel tank core cooling coils should provide this reasonable assurance. The application should demonstrate that the design of the primary cooling system can limit potential personnel exposures to the values found acceptable in Chapter 11.

- Acceptance criteria for fuel integrity and fuel cladding integrity limits should be limits unique to the AHR liquid fuel solution core, as defined in Chapter 4. These criteria provide the acceptable margin to the breach of the fuel solution boundary integrity.

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

Review Procedures

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

Evaluation Findings

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

- The primary cooling system is designed in accordance with the design bases derived from all relevant analyses in the SAR.

- Design features of the primary cooling system and components give reasonable assurance of primary system boundary integrity under all possible reactor conditions. The system should be designed to remove sufficient fission heat from the core to allow all licensed operations without exceeding the established LSSSs that are included in the TS.

- The design and location of primary cooling system components have been specifically selected to avoid coolant loss that could lead to primary system boundary failure or to an uncontrolled release of excessive radioactivity.

- The primary cooling system is designed to convert, in a passive or fail-safe method, to natural-convection flow sufficient to avoid loss of fuel integrity. *(This feature is evaluated in conjunction with the reviews of the reactor description and accidents. It is applicable to licensing AHRs to operate with forced-convection coolant flow).*

- The chemical quality of the primary coolant will limit corrosion of the primary cooling coils, the control and safety rod cladding or thimbles, the outside of the reactor vessel and other essential components of the primary cooling system for the duration of the license and for the projected utilization time of the fuel.

- Systems are present that will prevent hydrogen concentrations from reaching combustible limits.

- Primary cooling system instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.

- The TS, including testing and surveillance, provide reasonable assurance of necessary primary cooling system operability for reactor operations, as analyzed in the SAR.

- The design bases of the primary cooling system provide reasonable assurance that the environment and the health and safety of the public will be protected.

5a2.3 Secondary Cooling System

Areas of Review

The secondary cooling system of an AHR should be designed to transfer reactor heat from the primary and possibly other cooling systems to the environment. The secondary cooling system should be designed for continuous operation at the licensed power level. Therefore, the secondary cooling system in these reactors must be designed to dissipate heat continuously. In this section of the SAR, the applicant should justify how any necessary heat dissipation is accomplished. Section 5a2.3 of the format and content guide discusses specific areas of review for this section.

Acceptance Criteria

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

- The analyses and discussions in Section 5a2.3 should demonstrate that the secondary cooling system is designed to allow the primary cooling system to transfer heat, as necessary, to ensure fuel integrity. The analyses should address primary cooling systems operating with forced flow, natural-convection flow, or both, for reactors licensed for both modes. The design should show that the secondary cooling system is capable of dissipating all necessary fission and decay heat for all potential reactor conditions, as analyzed in the SAR.

- The primary coolant will usually contain radioactive contamination. The design of the total cooling system should ensure that release of such radioactivity through the secondary cooling system to the unrestricted environment would not lead to potential exposures of the public in excess of the requirements of 10 CFR Part 20 and the ALARA program guidelines. Designs should ensure that the primary cooling system pressure is lower than the secondary cooling system pressure across the heat exchanger under all anticipated conditions, that the secondary cooling system is closed, or that radiation monitoring and an effective remedial capability are provided. The secondary cooling system should prevent or acceptably mitigate an uncontrolled release of radioactivity to the unrestricted environment. Periodic samples of secondary coolant should be analyzed for radiation. Action levels and required actions should be discussed.

- The secondary cooling system should accommodate any heat load required of it in the event of a potential ESF operation or accident conditions, as analyzed in Chapters 6 and 13 of the SAR.

- The secondary cooling system design should provide for any necessary chemical control to limit corrosion or other degradation of the heat exchanger and prevent chemical contamination of the environment.

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

Review Procedures

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

Evaluation Findings

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

- Design features of the secondary cooling system and components will allow the transfer of the necessary reactor heat from the primary cooling system under all possible reactor conditions.

- Locations and design specifications for secondary cooling system components ensure that malfunctions in the system will not lead to reactor damage, fuel failure, or an uncontrolled release of radioactivity to the environment.

- Secondary cooling system instrumentation and controls are designed to provide all necessary functions and to transmit information on the operating status to the control room.

- The secondary cooling system is designed to respond, as necessary, to such postulated events as a reduction in cooling caused by either a loss of primary coolant or primary coolant flow.

- The TS, including testing and surveillance, provide reasonable assurance of necessary secondary cooling system operability for normal reactor operations.

5a2.4 Primary Coolant Cleanup System

Areas of Review

Experience has shown that potable water supplies are usually not sufficiently pure for use as a reactor primary coolant without additional cleanup. The AHR concepts considered here consist of a configuration with a primary cooling system immersed in fuel solution. The primary coolant is separated from the fuel solution by a material barrier that isolates the mobile fission products from the cooling system components. The primary cooling coil material is an example of a primary coolant barrier. The purity of the primary coolant should be maintained as high as reasonably possible for the following reasons:

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

- to limit the concentrations of particulate and dissolved contaminants that might become radioactive by neutron irradiation

Section 5a2.4 of the format and content guide discusses specific areas of review for this section.

Acceptance Criteria

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

- The primary coolant quality should be maintained in the ranges established as acceptable in Chapters 4 and 11 of the SAR. Experience has shown that water quality conditions and electrical conductivity ≤5 μmho/cm and pH between 5.5 and 7.5, can usually be achieved by good housekeeping and a cleanup loop with particulate filters and demineralizers. These water quality conditions would be acceptable unless the SAR analyses establish other purity conditions as acceptable.

- Radioactively contaminated resins and filters should be disposed of or regenerated in accordance with radiological waste management plans discussed in Chapter 11, and potential exposures and releases to the unrestricted environment shall not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility's ALARA program.

- The location, shielding, and radiation monitoring of the water cleanup system for routine operations and potential accidental events should be such that the operating staff and the public are protected from radiation exposures exceeding the requirements of 10 CFR Part 20 and acceptable radiological consequence dose limits for accidents.

- The location and functional design of the components of the water cleanup system should ensure the following:

  - Malfunctions or leaks in the system do not cause uncontrolled loss or release of primary coolant.

  - Personnel exposure and release of radioactivity do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility's ALARA program.

  - Safe reactor shutdown is not prevented.

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

Review Procedures

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

Evaluation Findings

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

- The design bases and functional descriptions of the primary water cleanup system give reasonable assurance that the required water quality can be achieved. The design ensures that corrosion and oxide buildup of cooling coils and other essential components in the primary cooling system will not exceed acceptable limits.

- The primary coolant cleanup system and its components have been designed and selected so that malfunctions are unlikely. Any malfunctions or leaks will not lead to radiation exposure to personnel or releases to the environment that exceed the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines.

- The plans for controlling and disposing of radioactivity accumulated in components of the primary water cleanup system, which results from normal operations and potential accident scenarios, conform to applicable regulations, including 10 CFR Part 20 and acceptable radiological consequence dose limits for accidents.

- The TS, including testing and surveillance, provide reasonable assurance of necessary primary water cleanup system operability for normal reactor operations.

5a2.5 Primary Coolant Makeup Water System

Areas of Review

During operations, it may be necessary to replace or replenish the primary coolant. Coolant may be lost through radiolysis, leaks from the system, and other operational activities. It might also be plausible that primary coolant is bled off to storage or holding tanks where evaporation would reduce makeup inventory. Although each reactor should have a makeup water system or procedure to meet projected operational needs, the system need not be designed to provide a rapid, total replacement of the primary coolant inventory. Section 5a2.5 of the format and content guide discusses specific areas of review for this section.

Acceptance Criteria

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

- The projected loss of primary coolant inventory for anticipated reactor operations should be discussed. The design or plan for supplying makeup water should ensure that those operational requirements are satisfied.
- If storage of treated makeup water is required by the design bases of the primary cooling system, the makeup water system or plan should ensure that such water is provided.
- Not all AHRs must provide makeup water through hardware systems directly connecting the reactor to the facility's potable water supply. However, for those that do, the makeup water system or plan should include components or administrative controls that prevent potentially contaminated primary coolant from entering the potable water system.
- The makeup water system or plan should include features to prevent loss or release of coolant from the primary cooling system.
- The makeup water system or plan should include provisions for recording the use of makeup water to detect changes that indicate leakage or other malfunctions of the primary cooling system.
- The applicant should identify operational limits, design parameters, and surveillances to be included in the TS.

Review Procedures

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

Evaluation Findings

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

- The design bases, functional descriptions, and procedures for the primary coolant makeup water system give reasonable assurance that the quantity and quality of water required will be provided.
- The system design or procedures will prevent overfilling the primary cooling system or a malfunction of the makeup water system, as well as the loss or release of contaminated primary coolant that would exceed the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines.

- The system design or procedures will prevent contaminated primary coolant from entering the potable water system through the makeup water system.

- The TS, including testing and surveillance, provide reasonable assurance of necessary makeup water system operability for normal reactor operations.

5a2.6 Nitrogen-16 Control System

Areas of Review

Nitrogen-16, a high-energy beta and gamma ray emitter with a half-life of approximately 7 seconds, is a potential source of high radiation exposure at water-cooled reactors. It tends to remain dissolved in the coolant water as it leaves the core. The quantity and concentration of nitrogen-16 should be considered and provisions made to control personnel exposure. Because of the relatively short half-life, potential doses can be decreased by delaying the coolant within shielded regions. If the reactor makes use of natural-convection cooling to a large open pool, stirring or diffusing the convection flow to the surface can produce a delay. For forced-flow cooling, passing the coolant through a large shielded and baffled tank can produce the delay. Section 5a2.6 of the format and content guide discusses specific areas of review for this section.

Acceptance Criteria

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

- The reduction in personnel exposure to nitrogen-16 should be consistent with the nitrogen-16 analyses in Chapter 11 of the SAR. The total dose should not exceed the requirements of 10 CFR Part 20 and should be consistent with the facility's ALARA program.

- The system design should not decrease cooling efficiency so that any LSSS would be exceeded, or lead to an uncontrolled release or loss of coolant if a malfunction were to occur, or prevent safe reactor shutdown and removal of decay heat sufficient to avoid damage to core components and other components of the primary fission product barrier.

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

Review Procedures

The reviewer should evaluate the design bases and functional requirements of the system that controls personnel exposures to nitrogen-16 by (1) confirming the amount of nitrogen-16 predicted by the SAR analysis at the proposed power level and the potential personnel exposure rates, including exposures from direct radiation and airborne nitrogen-16; (2) reviewing the type of system control and the anticipated decrease in exposure rates; (3) reviewing the effect of the proposed system on the full range of normal reactor operations; and (4) reviewing the possible effects of malfunctions of the nitrogen-16 control system on reactor safety, safe reactor shutdown, and release of contaminated primary coolant.

Evaluation Findings

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

- Design bases and design features give reasonable assurance that the nitrogen-16 control system can function as proposed and can reduce potential doses to personnel, so that they do not exceed the requirements of 10 CFR Part 20 and are consistent with the facility's ALARA program.

- Design and functional operation of the nitrogen-16 control system give reasonable assurance that the system will not interfere with reactor cooling under anticipated reactor operating conditions and will not reduce cooling below the acceptable thermal-hydraulic performance discussed in Chapter 4 of the SAR.

- Design features give reasonable assurance that malfunction of the nitrogen-16 control system will not cause uncontrolled loss or release of primary coolant and will not prevent safe reactor shutdown.

- The TS, including testing and surveillance, provide reasonable assurance of necessary nitrogen-16 control system operability for normal reactor operations.

5a2.7 Auxiliary Systems Using Primary Coolant

Areas of Review

The primary coolant may serve functions other than cooling the reactor core. Some of these auxiliary functions could involve cooling other heated components, which may affect the heat load of the primary cooling system. Auxiliary uses of the primary coolant could affect its availability as a fuel coolant, which is its principal use. Although the principal discussions of these auxiliary systems should be located in other sections of the SAR, their effects on the coolant systems should be summarized in this section.

Acceptance Criteria

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

- The system should remove sufficient projected heat to avoid damage to the cooled device.

- The system should not interfere with the required operation of the primary core cooling system.

- Any postulated malfunction of an auxiliary system should not cause the uncontrolled loss of primary coolant or prevent a safe reactor shutdown.

- The shielding system using primary coolant should provide sufficient protection factors to prevent personnel exposures that exceed the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines.

- The system should not cause radiation exposures or release of radioactivity to the environment that exceed the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines.

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

Review Procedures

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

Evaluation Findings

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

- The applicant has described and analyzed auxiliary systems that use primary coolant for functions other than core cooling, has derived the design bases from other chapters of the SAR, has analyzed any reactor components located in high-radiation areas near the core for potential heating that could cause damage to the reactor core or failure of the component, and has planned acceptable methods to remove sufficient heat to ensure the integrity of the components. The coolant for these systems is obtained from the purified primary cooling system without decreasing the capability of the system below its acceptable performance criteria for core cooling.

- The applicant has analyzed any reactor components or auxiliary systems for which primary coolant helps shield personnel from excessive radiation exposures. The use of the coolant for these purposes is acceptable, and the estimated protection factors limit the exposures to the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines. There is reasonable assurance that credible and postulated malfunctions of the auxiliary cooling systems will not lead to an uncontrolled loss of primary coolant, radiation exposures, or the release of radioactivity to the unrestricted environment that exceed the requirements of 10 CFR Part 20 and the facility's ALARA program guidelines.

- The TS, including testing and surveillance, provide reasonable assurance of necessary auxiliary cooling system operability for normal reactor operations.

5b Radioisotope Production Facility Cooling Systems

Add the following guidance to NUREG-1537, Part 2, Chapter 5:

The reviewer should ascertain that the application has either provided an adequate analysis to ensure that there is no need for auxiliary cooling during the course of any part of the radioisotope production process or has provided a complete description of the design and operation of any required cooling system.

Areas of Review

The reviewer should ascertain that the application includes a complete analysis of the thermal characteristics of the material in process during all phases of the radioisotope production process. If required, the design, construction and operation of the auxiliary cooling system should be clearly and completely explained in text, drawings, or diagrams, as necessary.

Acceptance Criteria

The acceptance criteria specified in NUREG-1537, Part 2, Section 5a2.7, may be used as they would apply to any required radioisotope processing cooling system.

In the event that cooling of the SNM solution is required during the radioisotope extraction process adequate precautionary measures are in place to prevent detrimental changes to the physical or chemical characteristics of the SNM solution. As an example, precautions against exceeding the soluble limits of the SNM in solution due to overcooling should be in place.

Review Procedures

The review procedures specified in NUREG-1537, Part 2, Section 5a2.7, may be used as they would apply to the radioisotope processing cooling system.

Evaluation Findings

The evaluation and findings specified in NUREG-1537, Part 2, Section 5a2.7, may be used as they would apply to the radioisotope processing system.

6 Engineered Safety Features

This NUREG chapter, as written, is limited to ESFs pertaining to heterogeneous reactors. The NUREG can be expanded to apply to AHRs and to radioisotope production facilities, as well. Chapters can be added, where appropriate, as follows:

Chapter 6a1, “Heterogeneous Reactor Engineered Safety Features”
Chapter 6a2, “Homogeneous Reactor Engineered Safety Features”
Chapter 6b, “Radioisotope Production Facility Engineered Safety Features”

This ISG augments NUREG-1537 to make it applicable to a heterogeneous reactor, an AHR, and a radioisotope production facility through the guidance provided in the following chapters, as applicable.

6a1 Heterogeneous Reactor Engineered Safety Features

NUREG-1537, Part 2, as written, applies to a heterogeneous reactor. This chapter of the SAR needs no additional guidance.

6a2 Homogeneous Reactor Engineered Safety Features

This ISG augments NUREG-1537, Part 2, to include application to an AHR.

Introduction

The third paragraph, fourth bullet, should read: “loss of reactor vessel integrity or fuel mishandling.”

Add a ninth bullet to read: “criticality accident”.

6a2.1 Summary Description

This section, as written, applies to an AHR facility.

6a2.2 Detailed Description

This section, as written, applies to an AHR facility.

6a2.2.1 Confinement

This section of the current NUREG applies to an AHR facility with the following modifications:

Replace the first, second, and third sentences of the second paragraph with: “During normal operations, the reactor may release small amounts of radioactive material. Specifically, relatively small amounts of fission-product gaseous and iodine radionuclides and some argon-41 could escape from the reactor primary fission product barrier. The applicant should describe how these releases to the environment will be controlled so that neither the public nor the facility’s operating staff will receive radiation doses greater than regulatory limits”.

The following sections, as written, apply to an AHR facility.

Areas of Review

Acceptance Criteria

Review Procedures

Evaluation and Findings

6a2.2.2 Containment

The current wording in this section of the NUREG generally applies to an AHR.

6a2.3 Emergency Core Cooling

The current version of this section of the NUREG applies to an AHR.

Areas of Review

Where reference is made to the fuel cladding, it is understood to mean the primary fission-product barrier in an AHR.

If the conclusion in other chapters of the SAR is that areas outside the core will require primary cooling (e.g., recombiner), this section must also evaluate a loss of coolant in these systems.

Acceptance Criteria

This section, as written, applies to a non-power reactor or radioisotope production facility.

Review Procedures

This section should read: "The reviewer should evaluate the accidents in Chapter 13 of the SAR to determine the scenario and consequences for a LOCA and to ascertain if the integrity of the reactor vessel or the FP gas management system can be compromised. The reviewer should verify that the proposed ECCS can prevent or mitigate the degradation of the reactor vessel and, if applicable, the FP gas management system. The reviewer should compare the design details of the ECCS with the design and functional requirements of the SAR LOCA and also the mitigated radiological consequences with 10 CFR Part 20 or 10 CFR Part 100 (for an AHR with a power level greater than 1 MW(t)) as applicable, to determine if the design is acceptable".

Evaluation Findings

The first bullet should read: "The applicant has identified a potential maximum hypothetical LOCA that could lead to unacceptable reactor vessel degradation or loss of FP gas management system integrity and unacceptable radiological consequences".

The second bullet should read: "The applicant's analysis of this accident in Chapter 13 includes a proposed ECCS whose design and function is to cool the fuel (and the FP gas management system, as appropriate) to prevent failure of the reactor vessel and associated containment".

6b Radioisotope Production Facility Engineered Safety Features

The radioisotope production process involves the separation of certain fission-product isotopes from irradiated SNM. Certain operations with SNM will be subject to the requirements of 10 CFR Part 70. Under 10 CFR Part 70 IROFS are identified from the accident analysis. Some IROFS may be comparable or equivalent to the ESFs required under 10 CFR Part 50. This ISG will include such IROFS in the ESFs and will refer to them as such.

Introduction

This section of NUREG-1537 generally applies to a radioisotope production facility, with the understanding that wherever the term “reactor” or “non-power reactor” appears, it is taken to mean “radioisotope production facility, as appropriate” Other changes that will make this section more appropriate for a production facility are as follows:

- The bullets in the third paragraph should be changed to the following:

    – loss of cooling (if it is required)
    – loss of primary fission-product barrier
    – failure of process control equipment
    – operator error
    – loss of electric power
    – criticality accident
    – hazardous chemical release
    – external events, such as fire, flood, earthquake

6b.1 Summary Description

The current version of this section of NUREG-1537 applies, provided the term “reactor” is understood to mean “radioisotope production facility”, as appropriate.

6b.2 Detailed Descriptions

The current version of this section of NUREG-1537 applies, provided the term “reactor” is understood to mean “radioisotope production facility”, as appropriate.

In addition to any radiological hazards associated with operations in a production facility, 10 CFR Part 70, Subpart H, “Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material,” specifies limits regarding exposure to hazardous chemicals. These limits should also be considered when reviewing this section of the SAR.

6b.2.1 Confinement

The wording in this section of NUREG-1537 generally applies to a radioisotope production system, provided the term “reactor facility” is understood to mean “radioisotope production facility”, as appropriate.

Areas of Review

The current wording in NUREG-1537 applies. Consider the consequences of hazardous chemical exposures, if they are possible.

Acceptance Criteria

The fourth bullet should read: "Where reference is made to radiological exposure limits, it is understood to mean radiological and chemical exposure limits".

Review Procedures

Wherever the term "radiological exposures" appears, it is understood to mean "radiological and chemical exposures."

Evaluation Findings

Wherever the terms "reactor facility" and "radiological exposure" appear, it is understood that they mean "non-power reactor" or "radioisotope production facility" and "radiological or chemical exposure," respectively.

The fourth bullet should read: "The consequences from accidents to the public, the environment, and the operating staff will be reduced to values or levels that do not exceed the applicable limits of 10 CFR Part 20 for radiological exposures and also the requirements of 10 CFR Part 70, Subpart H, with regard to chemical exposures".

6b.2.2 Containment

The wording in this section of NUREG-1537 generally applies to a radioisotope production facility, provided the term "reactor facility" is understood to mean "radioisotope production facility", as appropriate.

Areas of Review

With the above interpretation of terminology, this section, as written, applies to a production facility.

Acceptance Criteria

With the above interpretation of terminology, this section, as written, applies to a production facility.

Review Procedures

The last sentence should read: "The net projected radiological and chemical exposures should be compared with the limits of 10 CFR Part 20 for radiological exposuresand 10 CFR Part 70, Subpart Hwith regard to chemical exposures to determine if the design is acceptable".

Evaluation Findings

The first bullet should read: "The applicant has either identified a potential or maximum hypothetical accident that results in projected exposures to the staff or the public that, without containment, would be greater than acceptable limits, or the applicant has elected to provide containment for adherence to the ALARA principle".

The second bullet should read: "The design and functional features proposed for the containment reasonably ensure that exposures will be reduced below the limits of 10 CFR Part 20 for radiological exposures and 10 CFR Part 70, Subpart H, for chemical exposures, with an additional…"

6b.2. 3 Emergency Cooling System

This section (6.2.3) of NUREG-1537 is applicable to a radioisotope production facility with the following changes:

- reference to ECCS in this section 6b.2.3 is interpreted to mean emergency cooling for the radioisotope production and irradiated fuel processing systems.

- reference to reactor or non-power reactor in this section 6b.2.3 is interpreted to mean radioisotope production and irradiated fuel processing facilities.

- reference to fuel cladding in this section 6b.2.3 is interpreted to mean primary fission product barrier.

Areas of review:

Replace the first paragraph with:

In the event it is necessary to provide cooling to the radioisotope production and irradiated fuel processing systems to maintain the primary fission product barrier, emergency cooling must be provided.

Acceptance criteria:

The current wording of this section of NUREG-1537, Part 2 is applicable to a radioisotope production facility provided that the fourth bullet includes an additional reference to the performance criteria for limiting exposure to hazardous chemicals per 10 CFR 70.61.

Review Procedures:

The current wording of this section of NUREG-1537, Part 2 is applicable to a radioisotope production facility provided the above mentioned interpretations of ECCS and fuel cladding are applied.

Evaluation Findings:

The current wording of this section of NUREG-1537, Part 2 is applicable to a radioisotope production facility provided the above mentioned interpretations of ECCS and fuel cladding are applied.

6b.3 Nuclear Criticality Safety for the Processing Facility

The primary purpose of this section is to determine, with reasonable assurance, that the applicant has designed a facility that will provide adequate protection against criticality hazards related to the storage, handling, and processing of licensed materials. The facility design must adequately protect the health and safety of workers and the public during normal operations and credible accident conditions from the accidental criticality risks in the facility. It should also protect against facility conditions that could affect the safety of licensed materials and thus present an increased risk of criticality or radiation release.

Another purpose of this review is to determine, with reasonable assurance, whether the licensee's or applicant's nuclear criticality safety (NCS) program, as described in the license application, is adequate to ensure compliance with the regulatory requirements and will support safe possession and use of nuclear material at the facility. The review should examine the parts of the license application that describe the NCS program. The review should ensure that either the license application for a new facility or the license amendment to an existing facility meets the applicable regulatory requirements.

Areas of Review

- The general organization and administration methods used by the applicant that relate to NCS, including the experience, educational requirements, responsibilities, and authorities of NCS management and staff, should meet the requirements.

- Process descriptions are narrative descriptions of the site, facility, and processes with respect to criticality safety for normal operations. The criticality process description can include flow diagrams, major process steps, and major pieces of equipment, with emphasis on the criticality safety control.

- Criticality accident evaluations should include accident analyses involving licensed materials and an interpretation of the sequence of events. It is presumed that all criticality accident analyses would assume high consequences; therefore, the applicant should include every credible event that could result in an uncontrolled criticality event.

- Criticality accident analyses should be identified, including the assumption that all criticality accidents are high-consequence events and that the applicant's bases and methods are based on using preventive controls.

- Criticality process safety controls should be provided for criticality safety, and a description of their safety function should be described. The applicant should use enough safety controls to demonstrate that, under normal and abnormal credible conditions, all nuclear processes remain subcritical.

- Criticality management measures should ensure that the reliability and availability of the safety controls are adequate to maintain subcriticality.

Acceptance Criteria

The reviewer should find the applicant's criticality safety program information acceptable if it provides reasonable assurance that the acceptance criteria discussed below are adequately addressed and satisfied:

- The applicant describes a facility criticality accident alarm system (CAAS) that meets the requirements of 10 CFR 70.24, "Criticality Accident Requirements."

- The applicant commits to American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.3-1997, "Criticality Accident Alarm System," as modified by RG 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," issued October 2005 by the NRC. RG 3.71 lists the following exceptions to the standard:

  – At or above the mass limits, the applicant should require CAAS coverage in each area where SNM is handled, stored, or used.

  – A requirement that two detectors cover each area needing CAAS coverage.

  – A requirement that a CAAS be capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 0.2 Gy (20 rads) of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute.

- The applicant provides a description of a CAAS that is appropriate for the facility for the type of radiation detected, the intervening shielding, and the magnitude of the minimum accident of concern.

- The CAAS is designed to remain operational during credible events, such as a seismic shock equivalent to the site-specific, design-basis earthquake or the equivalent value specified by the Uniform Building Code.

- The CAAS is designed to remain operational during credible events, such as a fire, an explosion, a corrosive atmosphere, or other credible conditions.

- The criticality accident alarm is clearly audible in areas that must be evacuated or there are alternative notification methods that are documented to be effective in notifying personnel that evacuation is necessary.

- The applicant either, commits to the following national standards, as they relate to these requirements: ANSI/ANS-8.7-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials"; ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials"; ANSI/ANS-8.10-1983, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement"; ANSI/ANS-8.12-1987, "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors"; ANSI/ANS-8.15-1981, "Nuclear Criticality Control of Special Actinide Elements"; and ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors" or, the applicant specifies safety limits based on validated calculational methods.

- The applicant commits to rendering operations safe, by shutdown and quarantine, if necessary, in any area where CAAS coverage has been lost and not restored within a specified number of hours. The number of hours should be determined on a process-by-process basis, because shutting down certain processes, even to make them safe, may carry a larger risk than being without a CAAS for a short time. The applicant should commit to compensatory measures (e.g., limiting access, halting SNM movement) when the CAAS system is not functional. These provisions are included in the technical specifications governing the operational requirements for the CAAS.

- The applicant shall institute emergency procedures per 10 CFR 70.24(a)(3) to include the following management provisions:

    - The applicant has an emergency plan preferably according to the guidance in ANS/ANSI 8.23-1997, "Nuclear Criticality Accident Emergency Planning and Response"

    - The applicant commits to providing fixed and personnel accident dosimeters in areas that require a CAAS. These dosimeters should be readily available to personnel responding to an emergency, and there should be a method for prompt onsite dosimeter readouts.

    - The applicant commits to providing emergency power for the CAAS or providing justification for the use of continuous monitoring with portable instruments.

- The applicant describes a program that ensures compliance with the double-contingency principle, where practicable. Processes in which there are no credible accident sequences that lead to criticality meet the double-contingency principle by definition. This principle, as given in ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," states that at least two changes in process conditions must occur before criticality is possible. If there are no process changes leading to criticality, then the principle is satisfied. Each process that has accident sequences leading to criticality should have sufficient controls in place to ensure double-contingency protection. This may be provided by either (1) control of two independent process parameters, or (2) control of a single process parameter, such that at least two independent failures would have to occur before criticality is possible. The first method is preferable because of the inherent difficulty in preventing common-mode failure when controlling only one parameter.

- The applicant meets the acceptance criteria in Section 13b of the standard review plan, as they relate to the identification, consequences, and likelihood of NCS accident sequences, as well as descriptions of IROFS for NCS accident sequences.

- The applicant should consider the upsets listed in Appendix A to ANSI/ANS-8.1-1983 in identifying NCS accident sequences.

- The applicant describes how it performed the safety analyses for the new process and how the process satisfies the principles of the baseline design criteria (BDC)) (refer to 10 CFR 70.64(a) & (b)). The applicant also explains how it applies defense in depth to higher risk accident sequences. Acceptable defense-in-depth principles for the criticality safety design are those that support a hierarchy of controls: prevention, mitigation, and operator intervention, in order of preference.

- The applicant describes proposed facility-specific or process-specific relaxations or additions to BDC, along with justifications for such relaxations.

- The safety analysis describes how the applicant used criticality safety in BDC establishing the design principles, features, and control systems of the new process.

- The applicant describes and commits to implementing and maintaining an NCS program to meet the regulatory requirements of 10 CFR Part 50 and 10 CFR Part 70.

- The application states the NCS program objectives, which should include those listed in this chapter.

- The application outlines an NCS program structure that is consistent with current industry practices (e.g., ANSI/ANS-8.1-1998 and ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety") and that defines the responsibilities and authorities of key program personnel.

- The NCS program requires the applicant (licensee) to establish and maintain NCS safety limits and operating limits for the possession and use of fissile material and to maintain management measures to ensure the availability and reliability of the controls. Criticality control limits and management measures are included in the technical specifications as required by 10 CFR 50.36.

- The SAR specifies that modifications to the facility or safety program will be evaluated for their impact on criticality as required by 10 CFR 50.59.

- The reviewer should find the applicant's NCS organization and administration acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

  – The applicant meets the acceptance criteria as they relate to NCS, including organizational positions, functional responsibilities, experience, and qualifications of personnel responsible for NCS.

  – The applicant meets the intent of ANSI/ANS-8.1 and ANSI/ANS-8.19 (see RG 3.71), as they relate to organization and administration.

  – The NCS organization should be independent of operations to the extent practical.

  – The applicant commits to providing distinctive NCS postings in areas, operations, work stations, and storage locations relying on administrative controls for NCS.

  – The applicant commits to requiring its personnel to perform activities in accordance with written, approved procedures when the activity may affect NCS.

Unless a specific procedure deals with the situation, personnel shall take no action until the NCS staff has evaluated the situation and provided recovery procedures.

- The applicant commits to requiring its personnel to report defective NCS conditions to the NCS program management.

- The applicant describes organizational positions, experience of personnel, qualifications of personnel, and functional responsibilities.

- The applicant commits to designating an NCS program director who will be responsible for implementing the NCS program.

• The applicant's NCS surveillance requirements should be considered acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

  - Training and Procedures

    a. The applicant meets the intent of ANSI/ANS-8.19-1996 and ANSI/ANS-8.20, "Nuclear Criticality Safety Training," as they relate to training.

    b. The applicant commits to training all personnel to recognize the CAAS signal and to evacuate promptly to a safe area.

    c. The applicant commits to providing instruction and training regarding the policy in the SRP guidance for NCS organization procedures.

    d. The applicant commits to ANSI/ANS-8.18-1996 as it relates to procedures.

  - Audits and Assessments

    • The applicant commits to ANSI/ANS-8.19-1996, as it relates to audits and assessments.

      i. The applicant commits to conducting and documenting walkthroughs (i.e., observation of operations to ensure compliance with criticality limits) of all operating SNM process areas, so that all such areas will be reviewed at some specified frequency. The reviewer should consider the complexity of the process, the degree of process monitoring, and the degree of reliance on administrative controls in assessing the acceptability of the specified frequency. Identified weaknesses should be referred to those responsible for facility corrective actions and should be promptly and effectively resolved. A graded approach may be used to establish an NCS walkthrough schedule.

ii. The applicant commits to conducting and documenting periodic NCS audits (such that all NCS aspects of surveillance requirements will be audited at least every 2 years). A graded approach may be used to justify an alternative NCS audit schedule.

iii. Audit requirements will be included in the Administrative Controls section of the facility technical specifications

- The reviewer should consider the applicant's NCS technical practices acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

  – NCS evaluations will be performed using industry-accepted and peer-reviewed methods.

  – NCS limits on controlled parameters will be established to ensure that all nuclear processes are subcritical, including an adequate margin of subcriticality for safety.

  – Methods used to develop NCS limits will be validated to ensure that they are used within acceptable ranges and that the applicant used both appropriate assumptions and acceptable computer codes.

  – The applicant commits to demonstrating (1) the adequacy of the margin of subcriticality for safety by ensuring that the margin is large compared to the uncertainty in the calculated value of K effective (K_{eff}) (effective multiplication factor); (2) that the calculation of K_{eff} is based on a set of variables within the method's validated area of applicability; and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

- The reviewer must use judgment in assessing whether the margin of subcriticality for safety is sufficient to provide reasonable assurance of subcriticality. The reviewer should consider the following factors:

  – conservatism in the calculations, beyond that needed to accommodate uncertainties in the modeled parameters (e.g., geometric tolerances).

  – confidence in subcriticality generated by the applicant's validation process, including the following:

    a. similarity between the benchmark experiments and calculations to be performed.

    b. sufficiency of the benchmark data (both quality and quantity).

    c. rigor of the validation methodology (e.g., trending, statistical testing).

    d. conservatism in the statistical parameters.

- sensitivity of the system to changes in modeled parameters, and therefore sensitivity to errors.

- corroborating evidence of subcriticality from other sources (e.g., knowledge of neutron physics for well-characterized systems, such as finished fuel).

- risk considerations, including the likelihood of actually attaining an abnormal condition.

* The applicant includes a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following:

  - a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, determine the bias and uncertainty in the bias, and determine the upper subcritical limit.

  - a summary of the physical systems and area(s) of applicability covered by the validation report, noting that it is not necessary to include the full range of numerical parameters that defines the area of applicability.

  - a description of the methods used to justify applying the methodology outside the area or areas of applicability.

  - a summary of the plant-specific benchmark experiments used to validate the methodology.

  - a description of the margin of subcriticality for safety and its justification.

  - a description of the controlled software and hardware.

  - a description of the verification process, including verification upon changes to the calculational system and upon some specified period.

* The applicant's validation methodology, as described above, should be found acceptable if either (1) the applicant commits to following ANSI/ANS-8.24-2006, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," as endorsed by RG 3.71; or (2) the methodology follows current industry practices in terms of selecting the benchmark experiments, assessing their applicability, determining the area(s) of applicability, extending the area(s) of applicability beyond the range of benchmark data, and statistically analyzing the data. This requires that the NCS reviewer remain aware of current practices in the area of criticality code validation.

* The reviewer should consider the applicant's commitment to NCS technical practices acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant's use of a single NCS control to maintain the values of two or more controlled parameters constitutes only one component necessary to meet double-contingency protection.

- In general, the applicant should commit to the following order of preference for NCS controls: (1) passive engineered, (2) active engineered, (3) enhanced administrative, and (4) simple administrative. When using other than a passive engineered control, the applicant should justify the choice of the type and manner.

- When they are relevant, the applicant should consider heterogeneous effects. Heterogeneous effects are particularly relevant for LEU processes where, all other parameters being equal, heterogeneous systems are more reactive than homogeneous systems.

• The use of mass as a controlled parameter should be considered acceptable in the following circumstances:

  - When mass limits are derived for a material that is assumed to have a given weight percent of SNM, determinations of mass are based on either (1) weighing the material and assuming that the entire mass is SNM; or (2) conducting physical measurements to establish the actual weight percent of SNM in the material.

  - When fixed geometric devices are used to limit the mass of SNM a conservative process density is assumed in calculating the resulting mass.

  - Instrumentation used to measure mass is subject to facility surveillance requirements.

• The use of geometry as a controlled parameter should be considered acceptable if, before beginning operations, all dimensions and nuclear properties that use geometry control are verified.

• The use of density as a controlled parameter should be considered acceptable in the following circumstances:

  - When process variables can affect the density, the accident analysis shows them to be controlled by surveillance requirements.

  - Instrumentation used to measure density is subject to facility surveillance requirements.

• The use of enrichment as a controlled parameter should be considered acceptable if the following apply:

  - Either a method of segregating enrichments is used to ensure that differing enrichments will not be interchanged or the most limiting enrichment is applied to all material.

- Measurements of enrichment are obtained by using instrumentation subject to facility management measures.

- The use of reflection as a controlled parameter should be considered acceptable in the following circumstances:
  - In the evaluation of an individual unit, the wall thickness of the unit and all reflecting adjacent materials of the unit are considered. The materials adjacent to the unit should be farther than 30 centimeters (12 inches).
  - After all fixed reflectors are accounted for, the controls to prevent the presence of any transient reflectors (e.g., personnel) in the accident analysis are identified as ESFs or TS, or both.

- The use of moderation as a controlled parameter should be considered acceptable if the following apply:
  - When using moderation, the applicant commits to ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."
  - When process variables can affect the moderation, the accident analysis shows them to be controlled by ESFs or TS, or both.
  - Moderation is measured by using instrumentation subject to facility surveillance requirements.
  - The design of physical structures prevents the ingress of moderators.
  - When moderation needs to be sampled, dual independent sampling methods are used.
  - Firefighting procedures for use in a moderation-controlled area evaluate the use of moderator material.
  - After all credible sources of moderation are evaluated, measures are instituted to prevent or control the inadvertent introduction of moderating materials into a moderation-controlled area.

- The use of concentration as a controlled parameter should be considered acceptable in the following circumstances:
  - When process variables can affect the concentration, the accident analysis shows them to be controlled by ESFs or TS, or both.
  - Concentrations of SNM in a process are limited unless the process is determined to be safe at any credible concentration.
  - When using a tank containing a concentration-controlled solution, the tank is normally closed and locked to prevent unauthorized access.

- When concentration needs to be sampled, dual independent sampling methods are used.

- After identification of possible precipitating agents, precautions are taken to ensure that such agents will not be inadvertently introduced.

• The use of interaction as a controlled parameter should be considered acceptable if the following applies:

  - To maintain a physical separation between units, engineered controls are used to ensure a minimum spacing. If engineered controls are not feasible, augmented administrative controls are used.

  - The structural integrity of the spacers or racks should be sufficient for normal and credible abnormal conditions.

• The use of neutron absorption as a controlled parameter should be considered acceptable in the following circumstances:

  - When using borosilicate-glass raschig rings, the applicant commits to ANSI/ANS-8.5-1996, “Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material.”

  - When using fixed neutron absorbers, the applicant commits to ANSI/ANS-8.21-1995, “Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors.”

  - In the evaluation of absorber effectiveness, neutron spectra are considered (e.g., cadmium is an effective absorber for thermal neutrons but ineffective for fast neutrons).

• The use of volume as a controlled parameter should be considered acceptable if the following apply:

  - Fixed geometry is used to restrict the volume of SNM.

  - When the volume is measured, the instrumentation used is subject to facility surveillance requirements.

• The reviewer should consider the applicant’s description of additional commitments for the NCS program acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

  - The applicant commits to using the NCS program to promptly detect any NCS deficiencies by means of operational inspections, audits, or investigations and report those deficiencies in ESFs or TS, or both; NCS function; or surveillance requirements to those individuals who are responsible for the facility’s corrective actions, so as to prevent recurrence.

- The applicant commits to supporting the facility change mechanism process by performing NCS evaluations per the requirements of 10 CFR 50.59 to determine changes to processes, operating procedures, criticality controls, ESFs, TS, and surveillance requirements will require a license amendment.

- The applicant commits to retaining records of NCS deficiencies and documenting any corrective actions taken.

• The applicant's description of measures to implement the reporting requirements for criticality safety-related commitments should be considered acceptable if the commitments are consistent with the overall program commitments and the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

  - The applicant has a program for evaluating the criticality significance of NCS events and an apparatus in place for making the required notification to the NRC Operations Center. Qualified individuals should make the determination of significance for NCS events. The determination of loss or degradation of double-contingency protection should be made against the license.

  - The applicant incorporates the reporting criteria and the report content requirements into the facility emergency procedures.

  - The applicant commits to issuing the necessary report, based on whether the criticality controls credited were lost (i.e., they were unreliable or unavailable to perform their intended safety functions), irrespective of whether the safety limits of the associated parameters were actually exceeded.

• If the applicant intends to conduct activities to which an NRC-endorsed standard applies, the intent of the standard should be met by satisfying the following acceptance criteria:

  - The license application contains a commitment to follow the requirements (i.e., "shall" statements) of the standard, subject to any exceptions taken by the NRC. The application clearly specifies the version of the standard and the specific provisions to which the applicant is committing, and,

  - If there are requirements in a standard to which the applicant does not commit, it provides sufficient information for the staff to determine if the requirements are not relevant to the applicant's activities or the license application contains other commitments that are equivalent.

• If the licensee commits to a standard that the NRC has not endorsed, is not the most current version endorsed by the NRC, or is an unendorsed version of a previously endorsed standard, the license application should include justification for this commitment.

• The reviewer should find the applicant's criticality safety information acceptable if it provides reasonable assurance that the acceptance criteria presented below are adequately addressed and satisfied. The applicant may elect to incorporate some or all of the requested process information in the facility and process description rather than in this section. Either approach is acceptable, as long as the information is adequately cross-referenced:

- Process descriptions are sufficiently detailed to allow an understanding of the criticality to permit development of potential accident sequences.

- The use of analysis to demonstrate compliance is acceptable in the following circumstances:

  a. The applicant provides a general description of the criticality hazards.

  b. Each hazard identified by the applicant includes a criticality-hazard evaluation of potential interactions and key assumptions, vessels, process equipment, and facility personnel.

  c. The applicant provides reasonable assurance that measures to mitigate the consequences of accident sequences are consistent with actions described in the standard review plan.

  d. All the credible criticality accident sequences are assumed to have high consequences.

- The application should demonstrate the management measures proposed to determine that safety controls are available and reliable to ensure subcriticality by briefly describing the following:

  - procedures to ensure the reliable operation of engineered controls (e.g., inspection and testing procedures and frequencies, calibration programs, functional tests, corrective and preventive maintenance programs, criteria for acceptable test results).

  - procedures to ensure that administrative controls will be correctly implemented, when required (e.g., employee training and qualification in operating procedures, refresher training, safe work practices, development of standard operating procedures, training program evaluation).

  - the configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements used by the applicant.

  - management provisions for the following:

    a. training and qualifications of NCS management and staff

    b. auditing, assessing, and upgrading the NCS program

    c. maintaining current NCS safety-basis documentation

d. installing and maintaining a CAAS to detect and annunciate an inadvertent nuclear criticality

e. referring NCS deficiencies to the corrective action program

f. retaining records of the NCS program, including independent reviews, audits, and documentation of corrective actions taken

g. preparing production facility operating staff for NRC operator license examination

Review Procedures

After the application has been accepted, the primary reviewer should conduct a complete review of the application and determine if it meets the conditions for approval specified in this section. The primary reviewer should consult with the supporting reviewers, as appropriate, to identify and resolve any issues of concern related to the licensing review. The primary reviewer should coordinate with other primary reviewers of other standard review plans to confirm that the application meets all acceptance criteria pertinent to NCS. The reviewer should also coordinate with other primary reviewers in radiation protection, chemical safety, and fire protection, as well as other disciplines, as appropriate (e.g., seismic), to ensure consideration of any cross-cutting issues.

The primary reviewer should review the applicant's NCS information in the license application for completeness with respect to the requirements.

The reviewer should identify and note any items or issues that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the commitments made in the license application are implemented through procedures and training.

Nuclear Criticality Safety Program

The reviewer should review all aspects of the applicant's NCS program, including management, organization, and technical practices. The reviewer should identify and note any items or issues relating to the NCS program and commitments that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the commitments made in the license application are implemented through procedures and training.

Safety Analysis

The results of the criticality safety analysis support the overall safety basis for the criticality safety evaluation. The reviewer should assess the criticality safety risks identified and ensure that the level of safety is reflected in the design and the operational process and controls for the facility. The reviewer should establish that the applicant's facility design, operations, and criticality safety controls provide reasonable assurance that they will function as intended, be reliable and available to perform their safety function, and provide for the safe possession and use of licensed material at the facility.

Evaluation Findings

The SAR should contain sufficient information to support the following types of conclusions in the SER:

The NRC staff has reviewed the nuclear criticality safety (NCS) program and requirements for criticality safety for [name of facility] according to this standard review plan. The NRC staff has reasonable assurance of the following:

- The applicant will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

- The applicant's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

- The applicant will have the capability to perform adequate safety analyses of all production processes that will be conducted in the facility. Credible postulated criticality accident scenarios can be performed and adequate preventive and mitigative controls and measures will be included in the production facility technical specifications as required by 10 CFR 50.36.

- The applicant will develop, implement, and maintain a criticality accident alarm system in accordance with both the requirements in 10 CFR 70.24 and the facility emergency management program. The applicant will have in place an NCS program.

Based on this review, the NRC staff concludes that the applicant's NCS program provides reasonable assurance of the protection of public health and safety, including that of workers, and the environment.

6.4 References

American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998- "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors"

ANSI/ANS – 8.3-1997- "Criticality Accident Alarm Systems"

ANSI/ANS – 8.7 – 1975 "Guide for Nuclear Criticality Safety in the Storage of Nuclear Materials"

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