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Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for 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
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
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
LCO	limiting condition for operation
LEU	low-enriched uranium
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
NOx	nitrogen oxide
NRC	U.S. Nuclear Regulatory Commission
NRCS	Natural Resources Conservation Service (U.S. Dept. of Agriculture)
NRHP	National Register of Historic Places
NRU	National Research Universal
PSAR	preliminary safety analysis report
OSHA	U.S. Occupational Safety and Health Administration
RAM	remote area monitor
Rev	Revision
RG	regulatory guide
SAR	safety analysis report
SGI	safeguards information
SHPO	State Historic Preservation Office
SL	safety limit
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, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Revision 0*,* February 1996

- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," Revision 0, February 1996

This ISG updates and expands the content of both NUREG-1537 Part 1 and Part 2 , respectively, 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 any of the following:

- A heterogeneous or an aqueous homogeneous (AHR) nonpower reactor 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 nonpower reactor

  - the core of an AHR

  - the content of a subcritical multiplier solution tank or reaction vessel 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) has subsequently entered into agreements with domestic commercial firms to encourage the expeditious construction of medical isotope production facilities, which will require NRC operating licenses. 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 ISG presented in this document augments existing guidance to define a means to expeditiously 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 many 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 practice is to produce bulk Mo-99 and ship it to a manufacturer of generators, which are sent to hospitals, medical centers, or radiopharmacies. 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 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) but have not been used for its bulk production.

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 U.S. 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 were experienced domestically and internationally during 2009 and 2010.

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

- The High-Flux Reactor simultaneously 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 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.

Characterization of Potential Applications and Licensing Requirements

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 under consideration are identified below along with an outline of the licensing requirements for each:

(1) Production of Mo-99 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 to NUREG-1537 is needed in this situation.

(2) Production of Mo-99 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 utilization of the reactor. If the proposed utilization 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 to NUREG-1537 is needed to clarify the licensing path in this situation.

(3) Production of Mo-99 by fissioning SNM in 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 nonpower 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. NRC staff guidance for licensing a production facility is discussed later in this document.

(4) Construction and operation of an LEU-fueled AHR and a facility 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. 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. NRC staff guidance for licensing a production facility is discussed later in this document.

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

- 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 a 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) accompanying the application must evaluate the performance of the reaction vessel relative to many of the same phenomena identified as licensing concerns for an AHR.

- 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. NRC staff guidance for licensing a production facility is discussed later in this document.

Licensing of 10 CFR Part 50 Utilization Facilities

NUREG-1537 presents guidance for the licensing of nonpower reactors. While AHRs had been licensed and operated in the United States before 1996, no AHRs were in operation and none were anticipated to be built 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.

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," have 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. This ISG follows the structure of that prepared for a 10 CFR Part 50 utilization facility, which is contained in NUREG-1537. Certain topics such as site characterization and conduct of operations were found relevant to both production facilities and utilization facilities and are incorporated by reference. Other topics such as facility description and accident analysis were found to be 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. 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 have to have 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.

Presentation of Interim Staff Guidance

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, Part 1, ‘Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,’ for the Production of Radioisotopes” **[ENTER DATE HERE]**

- “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]**

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 SARs 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

The wording in this chapter specifically refers to a reactor facility. Wherever the term “reactor” appears, it should be understood to mean a “nonpower reactor”, a “radioisotope production facility” or both, as appropriate.

1.1 Introduction

The inherent or passive safety features and any unique design features of the radioisotope production facility should be described.

1.2 Summary and Conclusions on Principal Safety Considerations

The first bullet should read: “consequences from the operation and use of the nonpower reactor and radioisotope production facility, and the methods used to ensure the safety of the facility.”

The last line in the second bullet should read: “…the type of building housing the facility and any special factors.”

1.3 General Description of the Facility

The description of the reactor facility required by NUREG-1537, Part 1, should be expanded to include the radioisotope production facility.

1.4–1.8

The standard format and content of these sections is applicable if, wherever the term “reactor” appears, it is understood to encompass both “nonpower reactor” and “radioisotope production facility.” The discussion should be expanded to include the radioisotope production facility as applicable.

2 SITE CHARACTERISTICS

The standard format and content guide from NUREG-1537, Part 1, Chapter 2, is applicable to a radioisotope production facility. The description of the site characteristics for both the reactor facility and radioisotope production facility must follow this guide. Wherever the term “reactor” appears, it should be understood to mean a “nonpower reactor,” a “radioisotope production facility” or both, as appropriate.

2.1.1.2 Boundary and Zone Area Maps

The fifth and sixth bullets call for the inclusion of rural and urban zones in the area maps. Only test reactor SARs need to include this information.

In this section, the controlled area prescribed by 10 CFR 20.1003, “Definitions,” and further specified by 10 CFR 70.61(f), is comparable to or is satisfied by one or more of the areas defined in this section. If it is not, the controlled area should be defined in this section.

2.6 <u>Bibliography</u>

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 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

NUREG-1537, Part 1, Chapter 3, is generally applicable to nonpower reactors and radioisotope production facilities. The standard format and content prescribed in Chapter 3, Sections 3.1 through 3.4, can remain unchanged provided that, where the term “reactor” appears, it should be understood to apply to both the “non-power reactor facility,” the “radioisotope production facility” or both, as appropriate. Section 3.5 will be divided into two subsections:

(1) Section 3.5a, “Reactor Facility”
(2) Section 3.5b, “Radioisotope Production Facility”

The standard format and content of Section 3.5 remains unchanged for the reactor facility. Section 3.5b should contain the same type of information prescribed for the reactor, but it should be limited to those systems and components relevant to the radioisotope production facility.

The following paragraphs should be added immediately following the first paragraph to the introduction in this chapter:

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

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

(1) preference for the selection of engineered controls over administrative controls to increase overall system reliability

(2) features that enhance safety by reducing challenges to items relied on for safety (IROFS)

3.1–3.4

The standard format and content of these sections are applicable if, wherever the term “reactor” appears, it should be understood to mean a “nonpower reactor” and a “radioisotope production facility.” The discussion should be expanded to include the radioisotope production facility as applicable.

3.5 <u>Systems and Components</u>

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.

The baseline design criteria for facilities that process SNM, and hazardous chemicals that are coincident with or result from operations with SNM, are prescribed in 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities." These criteria are similar to those enumerated either in this chapter regarding the general considerations in designing, constructing, and operating nonpower reactor facilities, or in other chapters addressing safety systems in greater detail. In lieu of reiterating these criteria in this ISG, the applicant should be aware of them and apply them appropriately to the SAR, particularly Chapters 3 through 9.

3.6 <u>Bibliography</u>

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

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

4 REACTOR AND ISOTOPE PRODUCTION FACILITY DESCRIPTION

NUREG-1537 was originally written for heterogeneous nonpower reactors. This ISG augments NUREG-1537, Part 1, to make it also applicable to aqueous homogeneous nonpower reactors and radioisotope production facilities. Additional sections have been added to this chapter as follows:

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

4a1 Heterogeneous Reactor Description

NUREG-1537, Part 1, Chapter 4, should be used for guidance in preparing this chapter.

4a2 Aqueous Homogeneous Reactor Description

NUREG-1537, Part 1, Chapter 4, should be replaced in its entirety with the following guidance:

In this chapter of the SAR, the applicant should describe and discuss 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.

While considering the guidance in this chapter for the AHR, it should be noted that the fuel solution performs the function of the fuel and moderator. If the reactor is intended to be used for the production of radioisotopes, the core can also be the radioisotope production target. However, in the following subsections, any direct reference to a moderator or a target is also intended to apply 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 primary fission-product barrier performed by the cladding does not apply to an AHR and a radioisotope production facility. The primary fission-product barrier in an AHR is 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.

A glossary of terms often used when discussing an AHR is found below:

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: 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.

Primary System Boundary: The 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, and pumps, valves, and piping.

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

Reactor Core: The 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 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 gas.

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

Recombiner: Device that recombines hydrogen and oxygen.

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

4a2.1 Summary Description

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

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

- thermal power level
- fuel type and enrichment
- use of gas-tight vessel
- forced and/or natural convection cooling
- type of coolant, solid moderator (if any), and reflector materials
- principal features for experimental programs or commercial isotope production programs or both as applicable
- pulsing or steady power
- novel concepts requiring substantial new development

- gas management system

4a2.2 Reactor Core

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

4a2.2.1 Reactor Fuel

In this section, the applicant should describe the reactor fuel system. Design features selected to ensure that the fuel (including fission products) barrier can withstand all credible environmental and irradiation conditions during its life cycle at the reactor site should be included. The discussions should address the in-core fuel operating conditions. Chapter 9, “Auxiliary Systems,” of the SAR should discuss handling, transport, and storage of fuel. Drawings and tables of design specifications and operating characteristics of the fuel should be presented.

In AHRs, radiolysis and fission-product gases build up within the reactor vessel above the liquid fuel. Hydrogen and oxygen from the radiolysis of water could lead to the development of an explosive gas mixture. In addition, where applicable, nitrate-based aqueous fuel will generate acidic nitrogen gases (NOx) by radiolysis. Gases generated during fission will also collect in the cover gas space. Therefore, information relevant to the headspace and the gas handling systems should be provided. (See References 1 and 2 at the end of this chapter.)

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

- Chemical composition, enrichment, uranium loading, and chemistry of the fuel. Information should be provided for fresh and reload fuel composition; solvent type and molarity; uranium loading and enrichment; expected fissile density in solution at operating pressure, temperature, and pH; uranium solubility; buildup of fission products and related decay daughters in solution, precipitates, and sweep gas system.
- Information on radiolytic gas formation and impact on reactor core chemistry, homogeneity, and reactivity. Implications of void formation and condensate return to the core for reactor performance should be discussed.
- Short-term changes in the chemistry of the fuel. Changes in the pH, temperature fluctuations, fission gas release, and changes in concentration because of radiolytic water and acid destruction would be expected during an operation cycle. The range of these fluctuations and their effects on reactor operation and control should be described.
- Long-term changes in the chemistry of the fuel. In particular, the buildup of fission products, activation products, and corrosion products would be of interest. Any plans and approaches for stabilizing or adjusting the fuel characteristics or composition should be included. Any plans regarding periodic reconstitution or purification of the fuel should also be included. Any scheduled periodic analysis plans for the fuel should be described. Finally, a description of the fuel at its end of life should be given.

- Description of the volume occupied by the fuel solution, including the height and diameter and the portion of the volume occupied by solids. Separate descriptions should be given for the critical and subcritical fuel (i.e., the gas evolved during irradiation should be included in the description of the critical fuel). Special features, such as reflectors, external geometrical designs to enhance cooling capability, and inherent safety or feedback provisions, should be discussed.
- Physical properties significant to safety that are important for the thermal-hydraulic analyses, such as solution density, power density and distribution, temperature, pH, pressure, heat capacity, gas evolution or diffusion (including fission-product gas), void formation or collapse, precipitation of fuel or fission-product complexes, and sweep gas.
- Material and structural information for the core vessel and coolant coils that relate to the integrity of the primary barrier, such as dimensions, fabrication methods, compatibility of materials, and specifications with tolerances.
- All types of fuel solution chemical constituents used should be described, as well as the fuel preparation method and location.
- Information on material parameters that could affect the integrity of the core vessel, the coolant components, control rod channels, and fuel transport piping, such as melting, softening, or blistering temperatures; corrosion; erosion; and mechanical factors, such as swelling, bending, twisting, compression, and shearing.
- A brief history of the fuel type, with references to the fuel development program, including summaries of performance tests, qualification, and operating history.
- Hydraulic forces, thermal changes and temperature gradients, internal pressures including that from fission products and gas evolution (including removal to gas treatment), pH control, pressure, precipitation, frothing, malfunctions of the gas treatment system, and radiation effects on the solution chemistry. Extended and more detailed discussion of these characteristics and effects may be included in Section 4a2.7, “Gas Management System” (addressed below).
- Adequate mixing of the fuel solution based on convection and gas evolution.

Information and analyses should support the limits on operating conditions for the fuel. These limits are specified to ensure that the integrity of the fuel barrier will not be impaired by solution pH, radiolytic gas evolution, solution boiling, power oscillations, precipitation from solution, temperature and pressure extremes or distributions, and materials compatibility. They should form the design bases for this and other chapters of the SAR, for the reactor safety limits, and for other fuel-related TS.

4a2.2.2 Control Rods

In this section, the applicant should give information on the control rods, including all rods or control elements that are designed to change reactivity during reactor operation. The physical, kinetic, and electromechanical features demonstrating that the rods can fulfill their control and safety functions should be described. Results of computing control-rod reactivity worths may be presented in this section, but details of the calculation of reactivity effects should appear in Section 4.5, “Nuclear Design,” of the SAR. The information in this section should include the following:

- The number and types of rods (e.g., shim, safety, regulating, transient), their designed locations in the core, and their designed reactivity worth. Provide the considerations and bases for redundancy and diversity and discuss the limits on core configuration.
- The structural and geometric description, including the shape, size, materials, cladding, fabrication methods, and specifications with tolerances for the rods. This should include the type and concentration of neutron absorber, or emitter, if applicable. Also, provide calculations of changes in reactivity worth due to burnup and assessment of radiation damage, heating effects, and chemical compatibility with the coolant and other core components. If the control rods have followers, the design, composition, and reactivity effects of the follower should be discussed.
- The structural and mechanical design relative to the core vessel penetrations provided for them. Are the penetrations closed or open-ended thimbles or tubes? Will the rods require cooling during operation at power? How does the thermal-hydraulic design prevent boiling of the coolant and/or formation of radiolytic gas bubbles that may cause reactivity transients? Cooling calculations may be included in Chapter 5, “Cooling Systems.”
- The design of mechanical supports for the active component, the method of indicating and ensuring reproducible positioning in the core, and the drive mechanism of each type of rod. This information should include the source of motive power, usually electrical, and the systems ensuring scram capability.
- The kinetic behavior of the rods, showing either the positive or negative rate of reactivity changes, in the normal drive and scram modes of operation. This information should be supplied for all rods, including transient rods in a reactor designed for pulsing.
- The applicant should show that the control rod design conforms to the shutdown margin requirements.
- The scram logic and circuitry, interlocks and inhibits on rod withdrawal, trip release and insertion times, and trip or scram initiation systems should be summarized here and described in detail in Chapter 7, “Instrumentation and Control Systems.”
- Special features of the control rods, their core locations, power sources, drive or release mechanisms designed to ensure operability and capability to provide safe reactor operation and shutdown under all conditions during which operation is required in the safety analysis if there is a single failure or malfunction in the control system itself. Such features may include mechanisms to limit the speed of rod movement.
- TS requirements for the control rods and their justification. These are the limiting conditions for operation (LCO’s), surveillance requirements (SR’s), and design features as discussed in Chapter 14, “Technical Specifications,” of this standard format and content guide.

4a2.2.3 Neutron Moderator and Reflector

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

4a2.2.4 Neutron Startup Source

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

4a2.2.5 Reactor Internals Support Structure

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

- Since the reactor core is an aqueous solution, the AHR core support structure is the reactor vessel. Therefore, this section should discuss the vessel and reflector vertical and lateral support structure, as well as the support for the reactor control and cooling components and any other components connected to the reactor vessel. The fuel positioning function of a heterogeneous reactor core support structure is not applicable to an AHR.
- The materials of construction, including considerations for radiation damage, corrosion, erosion, chemical compatibility with coolant and fuel solution and core components, potential effects on reactivity, induced radioactivity, and maintenance.
- Design features of the support structures that accommodate other systems and components such as radiation shields, reflectors, coolant coils and piping (including accommodation for buoyant and dynamic loads such as vibration), control rod drive thimbles, coolant plenums or deflectors, gas treatment systems, and nuclear detectors. Piping for fuel transfer to and from the core should be specifically addressed.
- TS that control important design features, LCO’s, and SR’s, as discussed in Chapter 14 of this format and content guide. The applicant should justify these TS in this section of the SAR.

4a2.3 Reactor Vessel and Pool

The core of the AHR is an aqueous solution within a gas-tight vessel. This vessel may rest in a pool, which acts as a shield and removes some small amount of heat from the reactor. In this section, the applicant should present all information about the vessel and pool necessary to demonstrate their integrity. The information should include the following:

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

4a2.4 Biological Shield

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

- The design bases for the radiation shields (e.g., water, concrete, or lead), including the projected reactor power levels and related source terms and the criteria for determining the required protection factors for all applicable nuclear radiation activity. Chapter 11 should present information about conformance with the regulations for radiation exposure and the facility ALARA (as low as reasonably achievable) program. The design bases should include the designed reactor power levels, the associated radiation source terms, and other radiation sources within the pool or tank that require shielding.
- The design details and the methods used to achieve the design bases. The applicant should discuss the protection of personnel and equipment functions. The information

should specify the general size and shape of the shields and the methods used to ensure structural strength, rigidity, and functional integrity. The applicant should discuss the distribution of shielding factors between liquid (e.g., water) and solid (e.g., concrete, lead) materials. If loss of shield integrity could cause a reduction in cooling, the features to prevent the loss of integrity should be described.

- The materials used and their shielding coefficients and factors, including a detailed list of constituents and their nuclear and shielding properties. The applicant should discuss radiation damage and heating or material dissociation during the projected lifetime of the reactor; induced radioactivity in structural components; potential radiation leakage or streaming at penetrations, interfaces, and other voids; shielding at experimental facilities; and shielding for facilities that store fuel and other radioactive materials within the reactor pool or tank.
- The assumptions and methods used to calculate the shielding factors, including references to and justification of the methods. Detailed results of the shielding calculations should give both neutron and gamma-ray dose rates at all locations that could be occupied. The applicant should calculate shield penetrations and voids, such as beam ports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.
- Methods used to prevent neutron irradiation and activation of ground water or soils surrounding the reactor shield that could enter the unrestricted environment. The applicant should estimate the maximum activity should such activation occur and describe remedial actions.
- TS that control important design features, LCO’s, and SR’s as discussed in Chapter 14 of this standard format and content guide. The applicant should justify these TS in this section of the SAR.

Regulatory Guide 2.1, “Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors,” has been replaced by Regulatory Guide 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” issued May 2009.

4a2.5 Nuclear Design

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

4a2.5.1 Normal Operating Conditions

In this section, the applicant should present information on the core geometry and configurations. The limiting core configuration for a reactor is the core that would yield the highest power density using the fuel specified for the reactor. All other core configurations should be demonstrated to be encompassed by the safety analysis of the limiting core configuration. Sections 4a2.5.3 and 4a2.6 should give further information on power density limitations. The information in the SAR should include the following:

- Discussion and analyses to reflect the impact of the gas management system that is contained within the fission-product barrier on the core physics. The discussion should include holdup times and subsequent release of fission-product gases and NOx. Sweep gas operation and discharge limits, recombiner operations, and reactivity impacts associated with these operations will need to be included in this section and in Section 4a2.7.
- Discussions and analyses of the reactor operating characteristics. The applicant should describe in detail the effects of changes in configuration and fuel chemistry, including effects related to pH, pressure, temperature, radiolytic gas recombination capacity, reactivity and power oscillation, and the control philosophy and methodology for each parameter. If applicable, the applicant should analyze safety-related considerations for all requested operating modes.
- Changes in core reactivity with fuel burnup, plutonium buildup, and poisons, both fission products and those added by design. 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 also be discussed.
- Analyses of the reactor kinetic behavior and the design requirements and dynamic features of the control rods that allow controlled operation for all possible reactor conditions. This includes the expected effects of radiolysis on power oscillations resulting from formation and movement of voids, effects of malfunctions in the recombiner and the possible resulting pressure pulses causing fuel solution density changes, and the effects of temperature changes or gradients in the solution.
- Analyses of the basic reactor criticality physics, including the interacting effects of fuel, solid neutron moderators (if any) and neutron reflectors, control rods, and in-core or in-reflector components such as those used for radioisotope production. This also includes discussion of the subcritical storage and handling of the full core mass outside the reactor vessel and during transport from and to the core, the reactivity swing of the processed core material after selected fission-product removal, and any compensatory measures (such as fissile addition or dilution) to achieve or maintain criticality after reinsertion into the reactor vessel.
- Discussion of the safety considerations for different core configurations, including a limiting core configuration that would yield the highest power densities and fuel temperatures achievable with the planned fuel. This includes the power stability effects of uneven, stochastic surface frothing, as well as void formation and collapse.
- The calculated core reactivities for all core configurations, including the limiting configuration that would yield the highest possible power density.
- Discussion of the administrative and physical constraints to prevent inadvertent addition of positive reactivity.

- TS that control important design features, LCO’s, and SR’s as discussed in Chapter 14 of this standard format and content guide. The applicant should justify these TS in this section of the SAR.

4a2.5.2 Reactor Core Physics Parameters

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

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

4a2.5.3 Operating Limits

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

- Reactivity conditions, excess reactivity, and negative reactivity for combinations of control rods inserted that are analyzed for the limiting core and operating cores during the life of the reactor. The applicant should discuss operational and safety considerations for excess reactivity.
- The excess reactivity based on the worth of reactor temperature coefficients, poisons, applicable experiments, and the worth associated with the radiolytic gas formation and collapse. The applicant should justify the upper limit on excess reactivity to ensure safe reactor operation and shutdown.
- The amount of negative reactivity that must be available by control rod action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe-shutdown state. The analyses should assume that the most reactive control rod is fully withdrawn (one stuck rod), nonscramable control rods are at their most reactive

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

- The limiting core configuration that is possible with the planned fuel in this reactor. The limit should be imposed by the maximum neutron flux density and thermal power density compatible with coolant availability and maintaining operational stability (For the AHR, the limiting power density has been associated with the propensity for the core to become unstable with increasing power density. Consequently, the operating power density must be quantified and substantiated, including any margins specified). The safety limits and limiting safety system settings for the reactor should be derived from this core configuration. The detailed analyses should be included in Section 4a2.6. Normal operating conditions and credible events, such as a stuck control rod, should be considered.
- A transient analysis assuming that an instrumentation malfunction drives the most reactive control rod out in a continuous ramp mode in its most reactive region. The analysis should show that the reactor is not damaged and fuel integrity is not lost.
- The redundancy and diversity of control rods necessary to ensure reactor control for the considerations noted above.
- TS for safety limits (SL's), limited safety system settings (LSSS's), LCO's, and SR's as discussed in Chapter 14 of this standard format and content guide. The applicant should justify these TS in this section of the SAR.

4a2.6 Thermal-Hydraulic Design

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

- The various systems and approaches for removing heat from the core should be identified (e.g., cooling coils, pool heat removal, and gas management (heat removal) system including recombiner and reflux chiller condenser). The expected fraction of heat removed by each approach should be discussed. The ability of the combined systems and approaches to accommodate the varying power, from gas formation and collapse and transport, during normal and transient operation should also be discussed.
- The coolant hydraulic characteristics of the core, including cooling coil number and arrangement and coupling between coils; individual coil and total system flow rates; fuel solution and coolant pressures; pressure changes at channel exits and entrances; material compatibility and heat transfer between fuel solution and coolant coils, to include plating or precipitation of material on the surface of the coil; natural circulation within the fuel solution; temperature profile along a coil from entrance to exit; and frictional and buoyant forces. The applicant should address individual coils, as well as

the core as a whole, for all flow conditions in the primary coolant system, including temperature variations and wave propagation caused by vibration and chemistry changes resulting from breached coils. The transition from forced to natural convection flow in the cooling coils should be calculated, and the applicant should prepare calculations for an event during which normal electrical power is lost and the core decay heat must be removed. The discussion should also describe the above-core gas removal and overall pool cooling systems and the effect of the loss of these systems on core coolability and decay heat removal.

- The thermal power density distribution in the fuel and heat fluxes into the coolant of each coil and along the coil, derived from the fuel loading and neutron flux characteristics discussed above.
- Calculations and the thermal-hydraulic methodology for the transfer of heat to the coolant. The applicant should consider uncertainties in thermal-hydraulic and nuclear parameters and such engineering factors as coil thickness and the buildup of any layers both in the coil and on the outside of the coil. The calculations should be based on fuel measurements and procurement specifications, as well as operating history and conditions. The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use.
- The calculations and experimental measurements to determine the coolant conditions ensuring that fuel solution boiling does not occur. The applicant should calculate at least the limiting core configuration. The discussion should also examine the positive reactivity feedback characteristics of overcooling. Operating conditions should include steady fission power, shutdown decay heat, planned pulses, and transients analyzed in Chapter 13. The applicant should consider operational and fuel characteristics from the beginning to the end of fuel life.
- For the core geometry and the coolant thermal-hydraulic characteristics, a discussion to establish the fuel conditions that ensure vessel integrity and prevent solution boiling, such as fuel pressure, temperature, pH, solubility of fuel and fission products, radiolytic gas recombiner capacity, and temperature distributions. The discussion should show correlations among these factors and justify their use in deriving SL's and LSSS's for the TS.
- The design bases for the primary coolant system, emergency core cooling system, gas treatment system cooling and pool cooling, and other systems designed to maintain vessel and primary fuel barrier integrity and prevent solution boiling, which should also be discussed in Chapter 5, "Reactor Coolant Systems." The analyses here and in Chapter 13 should describe reduction-in-cooling scenarios for forced-flow reactors. Natural convection cooling that removes decay heat to ensure thermal stability should also be discussed. Flow blockages should be analyzed in Chapter 13.
- In the case of the AHR, the coolant flows through coils immersed in the fuel solution; thus, the breach of a coolant coil should be analyzed in Chapter 13, as should the effects of localized moderation in the vicinity of the coil.

4a2.7 Gas Management System

In this section, the applicant should describe the design of the system for removing radiolysis and fission-product gases from the core and cover gas of the AHR. The gas management system may also provide some reactor cooling; Section 4a2.6 and Chapter 5 of the SAR should describe this aspect of its function. The applicant should describe the major components of the gas management system, which may include the following:

- particulate trap
- radiolytic gas recombiner
- recombiner cooling system
- reflux condenser
- gas chiller/condenser
- condensate return
- chemical makeup
- compressor
- gas storage tanks
- pressure-relief valves

The essential functions of a gas management system are to remove the radiolytic hydrogen and oxygen, and subsequently recombine them, to prevent a hydrogen deflagration or detonation hazard, contain hazardous chemicals (e.g., radiolytic NOx gases) and volatile fission products (e.g., krypton, xenon, iodine) until ultimate discharge, and provide venting of any pressure transients that could result in damage to the reactor vessel or primary cooling system and result in loss of containment of the reactor fuel. The applicant should describe the gas management system features that perform these duties in sufficient detail to demonstrate that the reactor core can be operated safely and in accordance with applicable environmental release criteria. This information should include the geometric dimensions of the major components and piping (including whether it is favorable geometry), the materials of construction (including chemical compatibility with evolved gases), the composition of any trap media/filters, the pressure the equipment is designed to withstand, surge capacity for fission-product storage, and any additional passive or active devices, such as alarms and pressure-relief devices, needed to perform the system's intended function.

The recombiner may need its own cooling system because the catalyzed recombination of hydrogen and oxygen is an exothermic reaction. If this is not part of the primary cooling system of the reactor, but rather an auxiliary system, it should be discussed in Chapter 9.

The proper function of the gas management system, which is part of the primary fission-product barrier, is essential. Malfunction or failure of components in this system could cause excessive pressure that could have positive reactivity feedback to the fuel solution and operating instability. Excess hydrogen production, beyond the capacity of the recombiner, could lead to an explosive mixture in the system. Failure of any inert cover gas supply that may be part of the system could also result in an explosive hydrogen concentration. Instability or loss of cooling to the recombiner or condenser could cause containment failure due to overheating or thermal stress. All of the process variables controlling the gas management system must be analyzed and appropriate limits assigned in the TS to avoid such consequences.

Technical Rationale: The specific components in the gas management system may vary from one applicant to another; this system is designed to be general in nature. A component for recombining hydrogen and oxygen will be essential. A system for condensing water and acid and allowing it (and hopefully entrained uranium) back into the reactor core is desirable and would probably be part of any gas management system. A system for trapping entrained uranium and holding fission products until they can be safely disposed of is essential. There *would be essentially five classes of hazards that must be protected against: 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. The means of preventing these events must be described. Two of these hazards, deflagration/detonation and pressure increase, could potentially increase the density of the core and affect the power density. These potential events should be discussed in terms of reactor control.*

4a2.8 References

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

2. James A. Lane, ed., *Fluid Fueled Reactors*, "Part 1 Aqueous Homogeneous Reactors," 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 following guidance should be used for the standard format and content of applications for radioisotope production facility licenses.

4b.1 Facility and Process Description

In this section, the applicant should provide a summary description of the process and processing facilities. The description should include the principal safety considerations factored into the design. The design bases and functions of the structures, systems, and components should be presented in sufficient detail to allow a clear understanding and assurance 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 understanding of physical facility features and of the processes involved.

The summary description should include an analysis of the anticipated maximum radioactive inventory that will be in process at any particular time. The analysis should include a breakdown by radionuclide of the material in process. It should include the physical and chemical forms, volumes of solutions, and any anticipated airborne contaminants that will be present during routine operations.

4b.2 Processing Facility Biological Shield

The application should include a detailed description of the design and construction of the biological shield in which radiochemical processes will be conducted. The general layout of the processing facility relative to the reactor and other parts of the facility should be presented. The applicant should provide the shielding design basis, including any calculations that were used to prescribe the required form and substance of the shield.

The application should describe the functional design of the biological shield showing entry and exit facilities for product, process equipment and operating staff; number and location of windows, manipulators, optics, and any other accessory operating equipment. It should also describe any other penetrations into the shield that facilitate passing processing material or utilities into, out of, or among segments of the shield.

The application should also include a detailed description of the ventilation system for the biological shield structure, including the design basis and function. The description should include the design and location of vent ducting, filters, and fans. The description should include details on vent system operating limits under both normal and emergency operating conditions. The description should include the design basis and function of all filtering and/or 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, "Standards for Protection against Radiation," for the protection of the staff and the public, should be evaluated and included in the proposed engineered safety feature (ESF) in Chapter 6 and TS in Chapter 14 as applicable.

4b.3 Radioisotope Extraction System

The initial step in the fission-product radioisotope production process is the extraction of specific radioactive elements from the fission-product mixture contained in the irradiated SNM. This section of the application should describe this part of the production process in enough detail to provide assurance that the process can be performed within the regulatory limits of 10 CFR 20.1201, "Occupational Dose Limits for Adults," and 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

The application should explain the theory underlying the process design. In the case of an AHR facility, the radioisotope(s) will most likely be extracted from the reactor fuel. In the case where targets containing SNM in solid form are used in the production process, the initial step will be to prepare the target material so that specific fission-product radioisotopes may be separated from the SNM. To the extent necessary to understand the safety of the process, the application should provide processing details such as the following:

- The general method to be used for separating specific fission products from the fuel (e.g., ion-exchange, precipitation or liquid/liquid extraction from SNM in solution, or sublimation from SNM in solid form).

- The SNM in terms of physical and chemical form, volume or amount in process, and radioactive inventory in process.

- The sequence of radioisotope(s) extraction and the time increments involved.

- The processing apparatus including any piping, separation columns, reagent or reaction vessels, heating or cooling apparatus, local shielding of storage or transfer vessels, and any other apparatus involved in the process. These descriptions should include information such as materials of construction, physical size and arrangement, equipment operating characteristics, process monitoring or control equipment, and associated instrumentation (if extensive remote control instrumentation is involved details should be provided in Chapter 7).

- Auxiliary equipment that will be used, such as pumps, siphons, vents, filters, spill pans, heat exchangers, or ventilation or gas management apparatus.

- Criticality control features that may be provided by the physical design parameters of components, materials of construction, and any other measures derived according to the guidance in Chapter 6, “Engineered Safety Features,” Section 6b.3, “Nuclear Criticality Safety in the Radioisotope Production Facility.”

Drawings and diagrams should provide the necessary details of the radioisotope production facility.

The license TS in Chapter 14 of the SAR should include those engineered features or administrative measures in the radioisotope extraction process that are safety related and that are required to prevent the uncontrolled release of SNM or fission products that would result in exceeding the exposure limits to individuals or exceeding the limits on effluents as prescribed by 10 CFR Part 20 , Subparts C and D, and Appendix B, respectively.

4b.4 Special Nuclear Material Processing and Storage

Possession and use of SNM are generally licensed under the regulations of 10 CFR Part 70. It is anticipated that most radioisotope production facilities will be licensed under 10 CFR Part 50. The basic licensing safety premise of 10 CFR Part 50 and 10 CFR Part 70, relating to utilization and SNM facilities, respectively, differs in that the utilization facility allows the SNM to be safely made supercritical and maintained critical, while in the SNM facility criticality is not allowed. The guidance for license applications under 10 CFR Part 50 (found in NUREG-1537) covers safety issues with SNM in the form of fuel as it is used in a reactor, but it does not specifically address processes using SNM (except when it is in small, laboratory-scale batches) that are carried on outside of the reactor context. Such operations are usually proposed and reviewed under the guidance of NUREG-1520, which pertains to all of the regulatory requirements of 10 CFR Part 70. To provide thorough guidance for composing and reviewing applications for the unique features of a radioisotope production facility, the staff has combined NUREG-1537 and NUREG-1520 in such a way that the standard format and content of NUREG-1537 have been maintained and those unique, nonduplicative parts of NUREG-1520 that are not addressed in NUREG-1537 are included in this ISG. This section dealing with SNM operations outside of the reactor containment primarily uses NUREG-1520 guidance with regard to criticality and chemical safety. The contents of this chapter deal with processing components and procedures. Other chapters (e.g., 6, 11, 12, 13, and 14) address accident analyses and the consequent requirements for accident prevention and mitigation.

4b.4.1 Processing of Irradiated Special Nuclear Material

The postextraction part of the radioisotope separation process involves either returning the fuel solution to the reactor core vessel or performing subsequent procedures to prepare it for further use as fuel or for disposal as waste. This section of the application should describe this part of the production process in enough detail to provide assurance that the process can be performed within regulatory limits.

The application should explain the technical basis underlying the process design. To the extent necessary for understanding the safety of the process, the application should provide processing details such as the following:

- Description of the SNM in terms of physical and chemical form, SNM concentration, volume or amount in process, and radioactive inventory in process.

- Description of the sequence of sampling, analyzing, and conditioning or reconditioning (including the form and amount of chemical reagents used) the fuel solution as is necessary or appropriate for its intended further use, as well as the time increments involved.

- Description of the processing apparatus including any piping, separation columns, reagent or reaction vessels, and transfer or storage vessels. The information should include dimensions, materials of construction, equipment arrangement, equipment operating characteristics, and process monitoring or control equipment (if extensive remote control instrumentation is involved details should be provided in Chapter 7).

- Description of auxiliary equipment that will be used, such as pumps, siphons, vents, filters, spill pans, heating and cooling apparatus, and ventilation and gas management systems including filters or other gas (radiolytic or fission-product) isolation features.

- Description of criticality control features that may be provided by the physical design parameters of components, materials of construction, and any other measures derived per the guidance in Section 6b.3.

Drawings and diagrams should provide the necessary details of the SNM processing facilities and procedures.

The proposed license TS in Chapter 14 of the SAR should include those engineered features or administrative measures involved in irradiated SNM processing that are safety related and that are required to prevent the uncontrolled release of SNM or fission products that would result in exceeding the exposure limits to individuals or exceeding the limits on effluents as prescribed by 10 CFR Part 20.

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. To the extent necessary for understanding the safety of the process, the application should provide details such as the following:

- Criticality accident prevention measures that are derived according to the guidance in Section 6b.3.

- The chemical and physical form of the fuel in each phase of the operations before it is used in the reactor core.

- Areas in the facility where this material will be processed and stored.

- Any processing equipment and procedures used while storing or preparing the fuel for use in the core.

- Chemical accident prevention measures as appropriate.

The license TS in Chapter 14 of the SAR should include those engineered features or administrative measures involved in unirradiated SNM processing that are safety related and that are required to prevent criticality or the uncontrolled release of SNM or fission products that would result in exceeding the exposure limits to individuals or exceeding the limits on effluents as prescribed by 10 CFR Part 20.

Appendix 4.1

Regulatory Guide 2.1, “Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors,” has been replaced by Regulatory Guide 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” issued May 2009.

5 REACTOR COOLANT SYSTEMS

This chapter of NUREG-1537 was written for heterogeneous reactors, providing guidance on what should be contained in the chapter describing the reactor coolant system. The coolant system for an AHR is significantly different than a traditional reactor coolant system. The reactor may or may not function as a radioisotope production facility. This chapter may consist of multiple parts depending on the function of the facility. If the application is for a combined reactor and radioisotope production facility, it should contain two parts, each for the appropriate type of facility involved. These may be titled as follows:

- Chapter 5a1, “Heterogeneous Reactor Coolant Systems”
- Chapter 5a2, “Aqueous Homogeneous Reactor Cooling System”
- Chapter 5b, “Radioisotope Production Facility Cooling Systems”

Guidance for each of the reactor options follows.

5a1 Heterogeneous Reactor Coolant Systems

NUREG-1537, Chapter 5, Part 1, provides guidance for preparing this chapter.

5a2 Aqueous Homogeneous Reactor Cooling System

Replace NUREG-1537, Chapter 5, Part 1, in its entirety with the following guidance.

In this chapter of the SAR, the applicant should give the design bases, descriptions, and functional analyses of the AHR cooling systems. The principal purpose of the cooling systems is to safely remove the fission and decay heat from the reactor and dissipate it to the environment. The discussions should include all significant heat sources in the reactor and should show how the heat is safely removed and transferred to the environment.

The 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 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 gas. It is expected that during normal operation most, if not all, of the heat generation will occur in the core. Decay heat, however, can be produced by two separate sources within the reactor vessel: by soluble and insoluble decay products that remain in the core, and by gaseous decay products and any other decay products entrained by those gases into the gas space above the core. It may be necessary to provide separate systems to remove the decay heat from each of these regions.

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. Heat from an AHR core is expected to be removed through one or more cooling coils immersed in the core, and this system is expected to remove most of the heat produced in the core. This guidance does not pertain to cooling systems in which the fuel solution is transported outside the vessel in an external loop. Supplemental cooling systems may be necessary to remove heat from the fission gases above the core and heat produced in the recombiners. If the AHR is designed to operate at such low power levels that no significant temperature increases will occur during normal operation, an engineered primary cooling system is not required. For such a design, the applicant should, in Chapter 4, “Reactor Description,” of the SAR, discuss the disposition of the heat produced, estimate potential temperature increases during operation, and justify why an engineered primary cooling system for heat removal is not required. In this chapter, the applicant should summarize those considerations and conclusions.

The applicant should also describe and discuss systems to remove and dispose of the waste heat in this chapter. The design bases of the reactor cooling systems for the full range of normal operation should be based on ensuring acceptable reactor conditions established in Chapter 4 of the SAR. The design bases of any features of the 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.” In this chapter, the applicant should discuss and reference the TS where analyses are used as the basis for a requirement.

The “secondary cooling systems” for an AHR are those systems and components that transfer heat from the primary cooling systems to the environment or heat sink(s). These secondary cooling systems may involve additional heat exchangers and pumps to circulate the coolant. In this chapter, the applicant should identify and discuss reactor cooling systems, including auxiliary and reactor core subsystems that remove heat from the reactor. The description should include, for example, information on core cooling coils (which are the primary cooling system) and the partition of heat removal by additional reactor cooling systems that remove heat directly, such as the gas management system, or passively through the reactor vessel walls. These additional reactor cooling systems should be summarized in Section 5a2.1 and discussed in detail in Chapter 4, if reactor core systems, such as a gas management system, are involved. Details of auxiliary systems using coolant other than the primary cooling system, such as passive core cooling by the pool surrounding the vessel, should be discussed in Chapter 9, “Auxiliary Systems.”

In this chapter, the applicant should describe all auxiliary and subsystems that use and contribute to the heat load of either the primary or secondary cooling systems. Any auxiliary systems using coolant from other sources should be discussed in Chapter 9.

5a2.1 Summary Description

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

- type of coolant: liquid, gas, or solid (e.g., conduction to surrounding structures).
- type of cooling system.
- type of coolant flow in the primary cooling system: forced convection, natural convection, or both.
- type(s) of secondary cooling system(s), if present, and the method of heat disposal to the environment.

- description of capability to provide sufficient heat removal to support continuous operation at full licensed power.

- special or facility-unique features.

For an AHR, the applicant should provide additional information in this section on the reactor cooling systems unique to the principal features of AHRs, including supplementary core heat removal pathways. If the primary cooling system is not the sole means of heat removal and the core heat removal is partitioned between supplementing pathways, these additional pathways should be mentioned. The energy partitioning should be given. These other means of heat transport from the core should be summarized, including the corresponding amount of heat transported from the core and fraction of total core heat removed.

5a2.2 Primary Cooling System

The basic requirements and design bases of the primary cooling system are to maintain reactor facility conditions within the range of design conditions and accident analyses assumptions derived from other chapters of the SAR, especially Chapters 4 and 13. The applicant should show the interrelationships among all SAR chapters and the way the designed primary cooling system provides all necessary functions. The following information should be included:

- Design bases and functional requirements of the primary cooling system.

- Schematic and flow diagrams of the system, showing such essential components as the heat source (i.e., reactor core), heat sink (i.e., heat exchanger), pumps, piping, valves, control and safety instrumentation, interlocks, and other related subsystems.

- Tables of allowable ranges of important design and operating parameters and specifications for the primary cooling system and its components, including the following:

  – coolant material

  – coolant flow rates

  – inlet and outlet temperatures and pressure throughout the system

  – elevation of components and water levels relative to the reactor core

  – construction materials of components

  – fabrication specifications of safety-related components

  – coolant quality requirements for operation and shutdown conditions, including pH and conductivity at a minimum

- Discussions and analyses keyed to drawings showing how the system provides the necessary cooling for all heat loads and all potential reactor conditions analyzed in the thermal-hydraulics section of SAR Chapters 4 and 13, including the following:

- Removal of heat from the fuel and waste gases by all modes of heat transfer that apply. Discussion and analyses of the effect of the size, shape, and structural features of the primary vessel or surrounding pool on cooling characteristics; the function of the pool as a heat reservoir; and the effect of water depth on natural thermal convection cooling.

- Transfer of heat from the primary coolant to a secondary coolant system for all reactor conditions. This discussion should include any heat exchanger design and operating conditions. Some AHRs may have only a primary cooling system that functions as a heat reservoir. For such systems, the analyses should include any factors that limit continuous operation, such as pool water temperature, and the proposed TS that ensure operation within the analyzed limits.

- Safe reactor shutdown, including passive removal of decay heat from the fuel. This discussion should include the methods of removal of decay heat in the event of loss of offsite electrical power.

- Locations, designs, and functions of essential components such as primary cooling coils. 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. In addition, locations, designs, and functions of such essential components as drains, siphon-breaks, pumps, isolation valves, and check valves should be identified. Radiological effects of potential coolant releases should primarily be analyzed in Chapter 11, "Radiological Protection Program and Waste Management."

• Discussion of the control and safety instrumentation, including location and functions of sensors and readout devices. The scram or interlock functions that prevent safety limits from being exceeded should be shown and discussed, including those related TS.

• Description and function of any special features of the primary cooling system.

• Brief description and functions of special features or components of the primary coolant system that affect or limit personnel radiation exposures from such radionuclides as nitrogen-16 and argon-41 and from radioactive contaminants.

• Description of radiation monitors or detectors incorporated into the primary cooling system and discussion of their functions.

• Brief discussion and references to detailed discussions in later sections of auxiliary systems using primary cooling, such as coolant cleanup, makeup water, emergency core cooling, cooling of experiments or targets, and biological shield cooling. The direct effect of these auxiliary systems on the design and functioning of the primary cooling system should be discussed.

• Discussion of leak detection and allowable leakage limits, if any.

- Discussion of normal primary coolant radiation concentration limits, including sampling frequency, isotopes of interest, and actions to be taken if limits are exceeded.

- For reactors that have closed systems, a discussion of allowable hydrogen limits in air spaces that are in contact with the primary coolant.

- Discussion of TS requirements for parameters of the primary cooling system, including the bases and surveillance requirements.

5a2.3 Secondary Cooling System

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

- The design bases and functional requirements of the secondary cooling system, including whether the system is designed for continuous full-power reactor operation and whether it is shared with other reactors within the facility.

- Schematic and flow diagrams of the secondary cooling system, showing such essentials as how the heat exchanger connects the primary cooling system (i.e., the heat source) to the secondary cooling system, pumps, piping, valves, control and safety instrumentation, interlocks, and interface with the environment for ultimate release of the heat.

- Tables of the range of important design and operating parameters and specifications of the secondary cooling system, including the following:

  - coolant material and its source

  - coolant flow rates

  - type of heat dissipation system, such as cooling tower, refrigerator, radiator, or body of water

  - location of heat dissipation system in relation to the reactor and the heat exchanger

  - construction materials and fabrication specifications of components

  - heat dissipation specifications related to environmental factors (e.g., temperature and humidity)

  - specifications and limitations on coolant quality and corrosion of the secondary cooling system components, including the environmental effects of the use of secondary cooling chemicals

- Discussion and functional analyses keyed to the drawings showing how the system provides the necessary cooling for all potential reactor conditions. These discussions should address the following:

  – Inlet and outlet temperatures and pressures throughout the system, including the pressure differential between the primary and secondary cooling systems in the heat exchanger. The applicant should discuss how the pressure in the secondary cooling system is maintained above that in the primary cooling system for all operating conditions, or analyze the radiological effect of leakage of contaminated primary coolant into the secondary cooling system. Isolation of the heat exchanger during shutdown periods is an acceptable method to control potential primary-to-secondary system leakage if secondary cooling system pressure is lower than primary cooling system pressure only during periods of system shutdown. The applicant does not need to analyze primary-to-secondary-system leakage if secondary cooling system pressure is lower than primary cooling system pressure for only short periods for system testing or repair. If the transfer of primary coolant into the secondary cooling system is caused by an abrupt event, such as a tube rupture in the heat exchanger, the analysis should be given in Chapter 13 of the SAR and summarized here.

  – Control of heat removal from the secondary cooling system necessary to maintain temperatures in the core and primary cooling system within the limits derived in the thermal-hydraulics analyses in Chapters 4 and 13 of the SAR.

  – Removal of heat from the heat exchanger and release to the environment when the primary cooling system operates in all anticipated and licensed modes, as applicable.

  – Safe reactor shutdown and removal and dissipation of decay heat.

  – Secondary cooling system response to the loss of primary coolant.

  – Locations, designs, and functions of such essential components as drains, sumps, pumps, makeup water, and check valves that ensure contaminated primary coolant is not inadvertently transferred to the secondary coolant system and released to the environment.

- Discussion of control and safety instrumentation, including locations and functions of sensors and readout devices and interlocks or safety capabilities.

- Descriptions of functions of any radiation monitors or detectors incorporated into the secondary cooling system. Discussion of surveillance methods to measure secondary coolant radioactivity including frequency, action levels, and action to be taken.

- Brief comments and reference to detailed discussion in other sections of auxiliary cooling systems that transfer heat to the secondary coolant system.

- Discussion of TS requirements, as appropriate, for the secondary cooling system, including the bases and SRs.

5a2.4 Primary Coolant Cleanup System

In the AHR, the primary coolant is separated from the fuel solution by a material barrier, such as a cooling tube wall, which isolates the mobile fission products from the coolant system components. For an AHR, the inner wall surface of an immersed cooling coil (e.g., a tube) is the primary coolant boundary, analogous to defining the outer surface of fuel cladding surrounding solid fuel as a primary coolant boundary. Experience has shown that integrity of cladding and presumably of other metal boundaries is improved if corrosion is reduced by maintaining high chemical purity of the coolant. Thus, purity of the primary coolant should be maintained as high as reasonably possible for the following reasons:

- to limit the chemical corrosion of primary coolant barrier, 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 could be made radioactive by neutron irradiation.

The applicant should include the following information:

- The design bases and functional requirements of the primary coolant cleanup system. Experience at nonpower reactors has shown that with a well-planned water cleanup system and good housekeeping practices, primary coolant quality can be maintained within the following ranges:

  - electrical conductivity <5 μmho/cm
  - pH between 5.5 and 7.5

  The design bases should be consistent with the discussions in Chapter 4 of the SAR.

- Schematic drawings and flow diagrams of the primary coolant cleanup loop.

- Table of specifications for the cleanup system demonstrating that it is designed for the volume and throughput of the primary cooling system.

- Locations and functions of control and monitoring instrumentation, including sensors, recorders, and meters. The discussion of monitors should include methods for continuously assessing coolant quality and effectiveness of the cleanup system.

- Locations and functional designs of cleanup system components such as branch points, pumps, valves, filters, and demineralizers.

- Discussion of schedules and methods for replacing or regenerating resins and filters and disposing of resultant radioactivity to ensure that radiation exposures do not exceed the limits discussed in Chapter 11 of the SAR.

- Summary of methods for predicting, monitoring, and shielding radioactivity deposited in filters and demineralizers from routine operations. The detailed discussion should be in Chapter 11 of the SAR.

- Summary of methods for predicting and limiting exposures of personnel in the event of inadvertent release of excess radioactivity in the primary cooling system and deposition in filters and demineralizers. The detailed discussion should be in Chapter 13 of the SAR.

- Provisions in the design and operation of the cleanup system to avoid malfunctions that could lead to significant loss of primary coolant or release of contaminated coolant, which could cause radiological exposure of personnel or release to the unrestricted environment to exceed the requirements in 10 CFR Part 20 and the guidelines of the facility's ALARA program.

- Discussion of TS requirements for the primary coolant cleanup system, including the bases and SRs.

5a2.5 Primary Coolant Makeup Water System

During operations at nonpower reactors with a water-based primary cooling system, primary coolant must be replaced or replenished. Coolant may be lost as a result of radiolysis, designed leakage as from pump seals, or other operational activities. The AHR should include a system or a procedure that meets the projected needs for coolant. The makeup water system need not be designed to provide a rapid, total replacement of the primary coolant inventory, but should be able to maintain the minimum acceptable water quantity and quality for reactor operation.

The applicant should provide the following information:

- The design bases for the primary coolant makeup water system that account for all activities that could cause a decrease in the primary coolant. A large loss-of-coolant event should be analyzed in Chapter 13 of the SAR. Although a required emergency core cooling system need not be a part of the makeup water system, if it exists, it should be discussed in Chapter 6 of the SAR.

- Schematic diagrams and functional discussions that show the source of water, the methods of addition to the primary cooling system, and the requirements for pretreatment before addition.

- Locations and functions of control instrumentation, including sensors, readout displays, and interlocks. Methods should be discussed for tracking additions of makeup water to detect significant changes that might indicate leaks or other malfunctions of the primary cooling system.

- Discussion of safety systems and administrative controls to ensure that the system or procedures for adding makeup water will not lead to significant loss of primary coolant and will prevent leakage of contaminated coolant into the potable water supply.

- Discussion of TS requirements for the primary coolant makeup water system, including the bases and SRs.

5a2.6 Nitrogen-16 Control System

When ordinary oxygen is irradiated with neutrons of sufficient energy, nitrogen-16, a high-energy beta and gamma emitter with a 7-second half-life, is formed. In water-cooled reactors operated above a few hundred kilowatts, the radioactivity of this nuclide may require specific systems or procedures for limiting personnel exposure.

In reactors with forced convection cooling, the coolant carrying the nitrogen-16 out of the core may be passed through a system such as a large shielded and baffled tank. This delay allows the radioactivity to decay significantly before the coolant emerges from the shielding. Another method of radiation control is to shield the entire primary cooling system.

The applicant should analyze the potential for personnel exposure to nitrogen-16 and propose control systems or procedures that include the following:

- Design bases and functional design of the nitrogen-16 control system or procedures. The design bases should be derived from analyses in Chapter 11 of the SAR.

- Schematic drawings and system and component specifications for the nitrogen-16 control system.

- The method used by the nitrogen-16 control system to reduce exposure rates and potential doses in occupied areas. Potential doses with and without the nitrogen-16 controls should be analyzed in Chapter 11 and summarized in this section of the SAR.

- The effect of the nitrogen-16 control system on overall reactor safety and operation.

- Other reactor design features affected by the nitrogen-16 control system. For example, a large shielded decay tank may affect coolant flow parameters, pump sizes, access for surveillance or inservice testing, or other factors for the primary cooling system.

- An assessment that the nitrogen-16 control system would not lead to an uncontrolled loss of primary coolant or the release of contaminated primary coolant that exceeds the requirements in 10 CFR Part 20 and the facility's ALARA program guidelines. Methods for analyzing radiation exposures as a result of coolant release should be consistent with the analyses in Chapter 11.

- Discussion of any TS requirements for the nitrogen-16 control system, including the bases and SRs.

5a2.7 Auxiliary Systems Using Primary Coolant

In addition to the systems discussed above that are associated with the primary cooling system, other auxiliary cooling systems may require the use of primary coolant and may affect the operation or safety of the reactor. If the reactor design includes an emergency core cooling system, it should be described and discussed in Chapter 6.

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

- Design bases and functional requirements of the auxiliary systems based on discussions elsewhere in the SAR.

- Schematic drawings and flow diagrams that show the source of water, locations of sensors and instruments, and locations of the components cooled.

- Tables of the range of important parameters of the systems and specifications of materials and components.

- Discussion of components to be cooled, the source of heat, the source of the coolant water, heat transfer to the coolant, and coolant heat dissipation.

- Discussion of the provisions in the auxiliary system designs to prevent interference with safe reactor shutdown.

- Discussion of the provisions in the auxiliary system design to prevent the uncontrolled release of primary coolant or radiation exposures that would exceed the requirements in 10 CFR Part 20 and the facility's ALARA program guidelines.

- Requirements for minimum water quality.

- Discussion of any TS requirements for the auxiliary cooling systems, including the bases and SRs.

5b <u>Radioisotope Production Facility Cooling Systems</u>

A radioisotope production facility that separates fission products from irradiated SNM may involve process inventories or batches containing high concentrations of fission products that require engineered safety features (ESF's) or auxiliary systems to prevent the release of radioactive material from the process apparatus or the confinement systems. One concern about the safe handling of production process batches that should be addressed is the potential heat load contained as a result of the radioactive decay of the fission-product inventory.

License applications for radioisotope production facilities should present an analysis of the thermal characteristics of the anticipated process that considers the following:

- The operating power of the SNM during irradiation in the reactor.

- The decay time allowed after the end of irradiation in the reactor before the separation process progresses.

- The volumetric heat load and the resultant thermal flux at heat transfer surfaces of the process containment apparatus throughout the process.

- Calculations of the resultant maximum temperature of material in process with the objective of determining the need for auxiliary cooling to maintain the temperature and pressure within the processing components at safe levels to prevent the failure of the process apparatus or the containment system.

In the event auxiliary cooling is required to maintain the process temperature within acceptable limits, the applicant should provide a complete description of the cooling system. The description should include the design basis, the physical characteristics, and the operation of the system. Discussion topics should be similar to those identified in Section 5a2.7. Any safety-related operating limits should be determined and included in the appropriate TS in Chapter 14 of the SAR.

6 ENGINEERED SAFETY FEATURES

As written, this chapter of NUREG-1537 is limited to ESF's pertaining to heterogeneous reactors. This ISG augments NUREG-1537 to make it applicable to a heterogeneous reactor, to an AHR, and to a radioisotope production facility through the guidance provided in the following chapters:

- Chapter 6a1, "Heterogeneous Reactor Engineered Safety Features"
- Chapter 6a2, "Aqueous Homogeneous Reactor Engineered Safety Features"
- Chapter 6b, "Radioisotope Production Facility Engineered Safety Features"

6a1 Heterogeneous Reactor Engineered Safety Features

NUREG-1537, Part 1, Chapter 6, should be used as guidance in preparing this chapter.

6a2 Aqueous Homogeneous Reactor Engineered Safety Features

Since NUREG-1537 is basically written for a heterogeneous reactor with fuel in solid form that has cladding to serve as the primary fission-product barrier, it may be used as the basic guidance for preparing this part of an AHR SAR, provided that the basic differences between these two types of reactors are addressed. For example, any reference to fuel cladding as the primary barrier for fission products should, in the case of an AHR, be taken to mean the reactor vessel and the gas management system. Otherwise, the guidance in this section is general enough to apply to any type of reactor facility, as long as the unique features of each are addressed and appropriate ESF's are provided to ensure that operations are conducted within safe limits.

6b Radioisotope Production Facility Engineered Safety Features

The SNM in an AHR serves a dual purpose as the fuel as well as the isotope production target and, as such, will be routinely cycled through reactor operation, isotope extraction, and fuel reconditioning operations. Other types of production facilities will be processing irradiated SNM outside of the reactor as well. Certain operations with the fuel or irradiated SNM will be subject to the requirements of 10 CFR Part 70. License conditions under 10 CFR Part 70 that are derived from accident analyses are defined as items relied on for safety (IROFS). Some of these IROFS may be comparable or equivalent to ESF's that are required under 10 CFR Part 50. Section 6.0 of NUREG-1537 may be applied to guidance for a production facility SAR with the following three provisions:

(1) Wherever the term "reactor" or "nonpower reactor" appears, it is understood to mean "radioisotope production facility," as appropriate.

(2) Change the third paragraph list of bullets 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, or wind

(3) Add the following paragraph:

In addition to the radiological exposure limits prescribed in 10 CFR Part 50 and 10 CFR Part 20 (by reference in Part 50), the regulatory limits pertaining to chemical exposure prescribed in 10 CFR 70.61 will also apply to a radioisotope production facility.

6b.1 Summary Description

Section 6.1 of NUREG-1537 is applicable to a radioisotope production facility, provided that wherever the term "reactor" or "nonpower reactor" appears, it is understood to mean "radioisotope production facility," as appropriate.

6b.2 Detailed Descriptions

Section 6.2 of NUREG-1537 is applicable to radioisotope production facilities, provided that wherever the term "reactor" or "nonpower reactor" appears, it is understood to mean "radioisotope production facility," as appropriate.

6b.2.1 Confinement

Section 6.2.1 of NUREG-1537 is applicable to a radioisotope production facility if wherever the term "reactor" or "nonpower reactor" appears, it is understood to mean "radioisotope production facility," as appropriate.

6b.2.2 Containment

Section 6.2.2 of NUREG-1537 is applicable to a radioisotope production facility with the following changes:

- First paragraph: The term "reactor facility" in this sub-section means "radioisotope production facility."

- Second paragraph: The second sentence should read "A possible scenario for such a release could be a significant loss of integrity of the radioisotope extraction system or the irradiated fuel processing system."

- Second paragraph: The third sentence should read "The containment is designed to control the release to the environment of airborne radioactive material that is released in the facility even if the accident is accompanied by a pressure surge or steam release."

6b.2.3 Emergency Cooling System

Section 6.2.3 of NUREG-1537 is applicable to a radioisotope production facility with the following change:

- First paragraph: The first sentence should read “In the event of the loss of any required primary or normal cooling system, an emergency cooling system may be required to remove decay heat from the fuel to prevent the failure or degradation of the gas management system, the isotope extraction system or the irradiated fuel processing system.”

The remaining parts of this section of NUREG-1537 are applicable as written, provided that any reference to a “reactor” or “reactor facility” is understood to mean “radioisotope production facility,” as appropriate.

6b.3 <u>Nuclear Criticality Safety in the Radioisotope Production Facility</u>

This section establishes the recommended contents of the nuclear criticality safety (NCS) section of the application. In general terms, the applicant should design a facility that will provide adequate protection against criticality hazards related to the storage, handling, and processing of SNM outside of the reactor.

The application should meet the 10 CFR Part 50 requirements for the NCS-related areas. The regulatory requirements for the license application review should comply with the general requirements of an application. The license application must propose equipment, facilities, and procedures to protect health and minimize danger to life or property, and to ensure that the design provides for criticality control, including adherence to the double-contingency principle ,which basically requires that more than one safety-related engineered feature or management measure must fail or be compromised before a criticality accident can occur. Requirements for criticality monitoring and alarms also apply.

Although 10 CFR Part 50 does not directly require a nuclear safety program, an applicant should provide commitments pertaining to NCS in the following areas:

- Establishing and maintaining NCS safety practices and procedures.

- Establishing and maintaining NCS safety limits and procedures for determining LSSS’s and LCO’s.

- Conducting NCS evaluations to ensure that, under normal and credible abnormal conditions, all nuclear processes remain subcritical, with an approved margin of subcriticality for safety.

- Providing training in procedures for criticality-related possession and use of nuclear material and for response to an inadvertent nuclear criticality event.

- Protecting against the occurrence of any identified accident sequence that could lead to an inadvertent nuclear criticality event.

The following additional guidance may be used to supplement the contents of the NCS program:

- NUREG-1513, “Integrated Safety Analysis Guidance Document,” May 2001.

- NUREG/CR-6410, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” March 1998.

The applicant should describe a program that ensures compliance with the double-contingency principle, where practicable. Processes in which there are no credible accident scenarios that lead to criticality meet the double-contingency principle by definition. This principle, as given in American National Standards Institute/American Nuclear Society (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.

The applicant may use the guidance in ANSI/ANS-8.10-1983, “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement,” as modified by Regulatory Guide 3.71, “Nuclear Criticality Safety Standards for Fuels and Material Facilities,” in determining the consequences of criticality accident sequences. In general, such events should be considered high-consequence events (refer to 10 CFR 70.61, “Performance Requirements”) unless controls are in place to provide shielding or other isolation between the source of radiation and facility personnel. Provide justification for considering events resulting in other than high-consequence.

The application should describe the NCS program in sufficient detail to enable an understanding of its objectives, structure and organization, and administration. The NCS program described in the application should have SR’s pertinent to the following topics:

- Training (refer to Chapter 12, Section 12.10, of this ISG concerning licensing of production facility operators).

- Procedures.

- Audits and assessments.

The NCS technical practices should include evaluations performed using accepted methods to ensure that all processes remain subcritical. The margin of subcriticality for safety is an allowance for any unknown uncertainties that have not been accounted for in validation and a measure of the degree of confidence that systems calculated to be subcritical are actually subcritical. The margin is used to define an upper subcritical limit, as follows:

$$\text{k-subcritical} = 1.0 - \text{bias} - \text{bias uncertainty} - \text{margin of subcriticality for safety}$$

In general, a margin of subcriticality for safety of 0.05 has been found acceptable for typical nuclear processes involving LEU, without a detailed justification. The use of increasingly smaller margins should require increasingly more rigorous justification, and the SAR should include evaluations of other physical systems on a case-by-case basis.

For each methodology used to perform an NCS analysis, a validation report should be generated. The methodology must be sufficiently rigorous and be applied in a manner consistent with its assumptions.

Additional commitments to the NCS should include a description of measures to implement the facility change process requirements in 10 CFR 50.59. The applicant should describe a change control process that is sufficient to ensure that the safety basis of the facility will be maintained during the lifetime of the facility. The change control process should be connected to the facility's configuration management system to ensure that changes to the NCS basis are incorporated into procedures, evaluations, postings, drawings, and other safety-basis documentation.

The applicant may use standards as a means to meet regulatory requirements. Regulatory Guide 3.71 endorses the national standards in the ANSI/ANS-8 series, with some exceptions. The NRC endorsement of these standards means that they provide procedures and methodology generally acceptable to the NRC staff for the prevention and mitigation of nuclear criticality accidents. However, application of a standard is not a substitute for detailed NCS analyses for specific operations.

6b.3.1 Criticality Safety Controls

The applicant should provide a list briefly describing all criticality safety controls in sufficient detail to permit an understanding of their safety functions. The applicant could demonstrate that the likelihood of each credible criticality event will result in low overall risk. To reduce common-mode failures, the applicant should favor design features that use independent sources of motive force.

6b.3.2 Surveillance Requirements

The applicant should review Sureveillance Requirements (SR's) to ensure the availability and reliability of safety controls when they are required to perform their safety functions. SR's of controls may be graded commensurate with risk.

The applicant should state clearly how the design of the new facility or process provides for criticality control. The discussion should identify how the following were considered in the design:

- Subcriticality under normal and abnormal conditions.

- Criticality accident alarm system per the requirements of 10 CFR 70.24, "Criticality Accident Requirements."

- Implementation of double contingency.

6b.3.3 Technical Specifications

The applicant should identify those ESF's, administrative controls, and surveillance measures that are required to either prevent or mitigate the consequences of a criticality accident and include them in the license TS as prescribed in 10 CFR 50.36, "Technical Specifications," and described in the guidance in NUREG 1537, Chapter 14, as augmented by this ISG.

6b.4 References

American National Standards Institute/American Nuclear Society, ANSI/ANS-8.3-1997, "Criticality Accident Alarm System," ANS, La Grange Park, IL.

ANSI/ANS-8.7-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials," ANS, La Grange Park, IL.

ANSI/ANS-8.9-1987, “Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate,” ANS, La Grange Park, IL.

ANSI/ANS-8.10-1983, “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement,” ANS, La Grange Park, IL.

ANSI/ANS-8.23-1997, “Nuclear Criticality Accident Emergency Planning and Response,” ANS, La Grange Park, IL.

ANSI/ANS-15.7-1977, “Research Reactor Site Evaluation,” ANS, La Grange Park, IL.

Atomic Safety and Licensing Appeal Board, “In The Matter of Trustees of Columbia University in The City of New York”, 4 A.E.C. 849, (1972).

U.S. Code of Federal Regulations, Chapter I, Title 10, “Energy,” Part 70, “Domestic Licensing of Special Nuclear Material.”

U.S. Nuclear Regulatory Commission, “Integrated Safety Analysis Guidance Document,” NUREG-1513, May 2001.

U.S. Nuclear Regulatory Commission, “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” NUREG/CR-6410, March 1998.U.S. Nuclear Regulatory Commission, “Nuclear Criticality Safety Standards for Fuels and Material Facilities,” Regulatory Guide 3.71, October 2005.