7. INSTRUMENTATION AND CONTROL SYSTEMS

This chapter of NUREG-1537, Part 2, was written for heterogeneous reactors and specifies the content of a chapter describing the reactor. To expand the use of NUREG-1537 to aqueous homogeneous reactors (AHRs) or a radioisotope production facility, the applicant should provide additional chapters in the SAR, as necessary. The result should be one or two chapters with the following titles:

- Chapter 7a1, "Heterogeneous Reactor Instrumentation and Control Systems"
- Chapter 7a2, "Aqueous Homogeneous Reactor Instrumentation and Control Systems"
- Chapter 7b, "Radioisotope Production Facility Instrumentation and Control Systems"

As of the date of this ISG, the NRC is processing revised guidance concerning digital instrumentation and control (I&C) systems for non-power reactors (NPR). This ISG updates the original reference material so that NUREG-1537 reflects the most recent issue dates. Applicants for licenses subsequent to the issuance of this ISG should check for any new guidance at the time of application.

Guidance for each of these options follows.

7a1 Heterogeneous Reactor Instrumentation and Control Systems

NUREG-1537, Part 2, Chapter 7, "should be used for guidance in reviewing this chapter. The following reference is applicable:

- ANSI/ANS 10.4, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," updated in 2008.

7a2 Aqueous Homogeneous Reactor Instrumentation and Control Systems

NUREG-1537, Part 2, Chapter 7, should be used for guidance in reviewing this chapter, as appropriate for an AHR facility. The following reference is applicable:

- ANSI/ANS 10.4, "Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry," updated in 2008.

7b Radioisotope Production Facility Instrumentation and Control Systems

Add the following guidance to NUREG-1537, Part 2 Chapter 7:

> The radioisotope production facility may require sensors, electronic circuitry, displays, and actuating devices that provide the information and the means to safely control the radioisotope production process, the special nuclear material (SNM) fuel reconditioning process (if applicable), or other operations with SNM that are conducted outside of the reactor. I&C systems may also be employed to avoid or mitigate accidents. This section should include details regarding the design and operating characteristics of these I&C systems.

7b.1 Summary Description

Each I&C system for the radioisotope production facility should be designed to perform functions commensurate with the complexity of the processes therein. The applicant should provide a summary description of the I&C systems, including the design bases; the safety, considerations, and objectives; the operational characteristics of the production facility that determine or limit the I&C design; and the ways in which the various subsystems constitute the whole and interact to contribute to its essential functions. This summary should also include schematic, logic, and flow diagrams illustrating the various subsystems.

7b.2 Design of Instrumentation and Control Systems

This section should address the following as they relate to the I&C systems for the radioisotope production and SNM fuel reconditioning processes:

- Design criteria
- Design bases
- System description
- System performance analysis
- Conclusion

The remaining subsections discuss specific information that should be included in this section for each of the systems and how the reviewer should evaluate each subsystem.

7b.3 Process Control Systems

Areas of Review

The process control systems contain most of the I&C subsystems and components designed for normal operation of the radioisotope production and SNM fuel or target reconditioning processes, if applicable. The areas of review for the process control systems should discuss the factors requested in Section 7b.2, above, but only as they relate to the radioisotope production and SNM fuel reconditioning processes. Subtopics may include, but are not limited to, the following:

- Nuclear instruments—including radiation detectors and displays suitable for a particular process. Where SNM is in process in quantities above the thresholds stipulated in 10 CFR 70.24 a criticality alarm system (CAAS) must be provided.

- Process instruments—instruments designed to measure and display parameters critical to the radioisotope production process and the SNM fuel reconditioning processes, if applicable.

- Control elements—types, number, function, design, and operating features of process or reactivity control devices, or both (coordinated with the review of the section of Chapter 6 relating to criticality control).

- Interlocks—circuits or devices to inhibit or prevent an action unless a specified precondition exists with the intention of protecting personnel or other subsystems from harm.

The areas of review for the process control systems should also include the following:

- Bases, criteria, standards, and guidelines used for the design of the process control systems.

- Description, including logic, schematic, and functional diagrams, of the overall system and component subsystems.

- Analysis of the adequacy of the design to establish conformance to the design bases and criteria for stated critical parameters.

- Application of the functional design and analyses to the development of bases of technical specifications, including surveillance tests and intervals.

- Process control system failure modes to determine whether any malfunction of the process control system could prevent any subsystem from performing its safety function.

Acceptance Criteria

The acceptance criteria, together with the use of good engineering judgment, will help the reviewer to conclude whether the process control system is designed to provide for the reliable control of the radioisotope production and SNM fuel reconditioning processes for the full range of operation. Acceptance criteria include the following:

- The range of operation of sensor (detector) channels should be sufficient to cover the expected range of variation of the monitored variable during normal and transient process operation.

- The process control system should give continuous, redundant indication of neutron flux present during the radioisotope production and SNM fuel reconditioning processes, if applicable.

- The precision and accuracy of each sensor channel should be commensurate with the importance and the level of intensity of the variable being measured. This is particularly important for those instruments and controls monitoring parameters that are significant to safety.

- The system should give reliable and redundant neutron flux level and rate of change information from detectors or sensors that directly measure neutron flux.

- The system should give reliable information about the status and magnitude of process variable necessary for the full operating range of the radioisotope production and SNM fuel reconditioning processes, if applicable.

- The system should be designed with sufficient control of reactivity for all required production and SNM fuel reconditioning process operations and should ensure compliance with analyzed requirements on excess reactivity and shutdown margins. Review of this criterion should be coordinated with the review of the section of Chapter 6 relating to criticality control.

- The process control system should not be designed to fail or operate in a mode that would prevent any subsystem from performing its designed function.

- Hardware and software for computerized systems should meet the guidelines of Institute of Electrical and Electronics Engineers (IEEE) 7-4.3.2-2010, "IEEE Standard Criteria for Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," and Regulatory Guide (RG) 1.152, Revision 1, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plans," issued January 1996 (Appendix 7.1 to Chapter 7 of NUREG-1537, Part 1, "Format and Content). Software should meet the guidelines of ANSI/ANS 10.4-2008 that apply to non-power reactor systems.

- For I&C systems that are being upgraded to systems based on digital technology, the applicant should consult U.S. Nuclear Regulatory Commission (NRC) Generic Letter 95-02, "Use of NUMARC/EPRI Report TR-102348, 'Guideline on Licensing Digital Upgrades,' in Determining the Acceptability of Performing Analog-to-Digital Replacements under 10 CFR 50.59,'" dated April 26, 1995.

- The process control system should be designed for reliable operation in the normal range of environmental conditions anticipated within the facility.

- The process control system should be designed to assume a safe state during loss of electrical power.

- The subsystems and equipment of the process control system should be readily tested and capable of being accurately calibrated.

- Technical specifications, including surveillance tests and intervals, should be based on safety analysis report (SAR) analyses and should assure availability and reliability of all safety related monitoring and control instrumentation.

- Where neutron flux is a necessary process variable that must be measured for safety or control, at least one neutron flux measuring channel should give reliable readings to a predetermined flux level. If the production facility has neutron flux as a safety limit, the measurable flux level should be above the safety limit. For production facilities without neutron flux as a safety limit, the measurable neutron flux level should be high enough to show that the basis for limiting licensed neutron flux level is not exceeded.

- The applicant should describe in the SAR the interlocks used to limit personnel hazards or prevent damage to systems during the full range of normal operations.

- If an analysis of a process indicates a hazard to the process or the production facility, direct interacting or interlocking process controls may be justified. Any such automatic limiting devices should demonstrate that a safety function of any other process control subsystem will not be compromised.

Review Procedures

This chapter of the SAR should describe the I&C subsystems that apply to all normal functions and parameters of the radioisotope production and SNM fuel reconditioning processes, if applicable. These subsystems constitute the process control system. The reviewer should confirm that this section addresses I&C information for all normal functions and systems described in the chapters of the SAR.

The process control system comprises several subsystems; therefore, the reviewer should anticipate that the information in the SAR will be further subdivided, as noted in the section describing the areas of review. The subdivisions should address all of the factors listed in Section 7b.2, above, for each subsystem and should state how and where the subsystems interact and interface and how they function as a total process control system for normal operations. The reviewer should verify that all design bases are justified and that the designs themselves accurately and completely implement the applicable bases and acceptance criteria. The reviewer should obtain the assistance of experts in the I&C branch to review computer systems.

Evaluation Findings

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

- The applicant has analyzed the normal operating characteristics of the radioisotope production and SNM fuel reconditioning processes, if applicable. The applicant has also analyzed the functions of the process control system and components designed to permit and support normal production and reconditioning process operations and confirms that the process control system and its subsystems and components will convey all necessary information to the operator or to automatic devices to maintain planned control for the full range of production and reconditioning process operations.

- The components and devices of the process control system are designed to sense all parameters necessary for facility operation with acceptable accuracy and reliability and to transmit the information with high accuracy in a timely fashion. Control devices are designed for compatibility with the analyzed dynamic characteristics for the production and SNM fuel reconditioning processes.

- The applicant has ensured sufficient interlocks, redundancy, and diversity of subsystems to avoid total loss of operating information and control and to limit hazards to personnel. The applicant has also ensured compatibility among operating subsystems and components in the event of a single, isolated malfunction of equipment.

- The process control system was designed so that any single malfunction in its components, either analog or digital, would not prevent the process and facility protection systems from performing necessary functions.

- Discussions of testing, checking, and calibration provisions, and the bases of technical specifications (including surveillance tests and intervals), provide reasonable confidence that the process control system will function as designed.

7b.4 Engineered Safety Features Actuation Systems

This chapter of the SAR should follow the guidance in NUREG-1537, Part 2, Section 7.5 as it applies to the radioisotope production and SNM fuel reconditioning processes. This section should describe the actuation systems for any engineered safety features (ESFs) discussed in Chapters 6 or 13.

7b.5 Control Console and Display Instruments

This chapter of the SAR should follow the guidance in NUREG-1537, Part 2, Section 7.6 as it applies to the radioisotope production and SNM fuel reconditioning processes, with the following exceptions:

- Anywhere “reactor control system” or “RCS” is indicated, the applicant should discuss the production process control system.

- Anywhere “reactor protection system” or “RPS” is indicated, the applicant should discuss the system or means used to prevent criticality or breach of radioactive material containment in the production facility.

7b.6 Radiation Monitoring Systems

This chapter of the SAR should follow the guidance in NUREG-1537, Part 2, Section 7.7 as it applies to the radioisotope production and SNM fuel reconditioning processes. If the radiation monitoring systems for the production facility have been described in conjunction with the radiation monitoring systems for the reactor elsewhere in the SAR, the applicant should note that in this section, but the discussion does not need to be repeated. If the CAAS is described in another chapter of the SAR it should be referenced but not repeated here.

References

ANSI/ANS 10.4 – 2008, Verification and Validation of Non-Safety Related Scientific and Engineering Computer Programs for the Nuclear Industry.

IEEE 7-4.3.2, 2010, Standard Criteria for Digital Computer Systems in Safety Systems of Nuclear Power Generating Systems.

USNRC Regulatory Guide 1.152 Rev. 3, 7/2011, Criteria for Use of Computers in Safety Systems of Power Reactors.

8. ELECTRICAL POWER SYSTEMS

This chapter of NUREG-1537 was written for heterogeneous reactors and specifies the content of a chapter describing the electrical power system for the reactor facility. To expand the use of NUREG-1537 to AHRs or a radioisotope production facility, additional chapters may be provided as appropriate. The result should be one or two chapters with the following titles:

- Chapter 8a1, "Heterogeneous Reactor Electrical Power Systems"
- Chapter 8a2, "Aqueous Homogeneous Reactor Electrical Power Systems"
- Chapter 8b, "Radioisotope Production Facility Electrical Power Systems"

Guidance for each of these options follows.

8a1 Heterogeneous Reactor Electrical Power Systems

NUREG-1537, Part 2, Chapter 8, should be used for guidance in reviewing this chapter.

8a2 Aqueous Homogeneous Reactor Electrical Power Systems

NUREG-1537, Part 2, Chapter 8, should be used for guidance in reviewing this chapter.

8b Radioisotope Production Facility Electrical Power Systems

NUREG-1537, Part 2, Chapter 8, should be used for guidance in reviewing this chapter provided that a reference to the reactor should be interpreted to mean the radioisotope production facility, as appropriate. Where the reactor and production facility share a common electrical supply system, it is not necessary to duplicate the information in this chapter.

9. AUXILIARY SYSTEMS

This chapter of NUREG-1537 was written for heterogeneous reactors and specifies the content of a chapter describing the auxiliary systems for the reactor facility. To expand the use of NUREG-1537 to AHRs or a radioisotope production facility, additional chapters may be provided as appropriate. The result should be one or two chapters with the following titles:

- Chapter 9a1, "Heterogeneous Reactor Auxiliary Systems"
- Chapter 9a2, "Aqueous Homogeneous Reactor Auxiliary Systems"
- Chapter 9b, "Radioisotope Production Facility Auxiliary Systems"

Guidance for each of these options follows.

9a1 Heterogeneous Reactor Auxiliary Systems

NUREG-1537, Part 2, Chapter 9, should be used for guidance in reviewing this chapter.

9a2 Aqueous Homogeneous Reactor Auxiliary Systems

NUREG-1537, Part 2, Chapter 9, should be used for guidance in reviewing this chapter except as described in the following subsections.

9a2.1 Heating, Ventilation, and Air Conditioning Systems

This section of NUREG-1537, although intended for a heterogeneous reactor, is general enough to apply to other reactor types. The current guidance in NUREG-1537 can be used for the reviewing of the heating, ventilation, and air conditioning (HVAC) systems of AHR's.

9a2.2 Handling and Storage of Reactor Fuel

This chapter in NUREG-1537 is written primarily for a heterogeneous reactor. Therefore, the guidance should be interpreted appropriately for the different characteristics of AHR fuel. For example, any reference to cladding should be understood to mean the primary fission-product barrier; reference to handling tools should be understood to mean any fuel-handling equipment, and mention of damage to fuel should be understood to mean any compromise of the quality or the integrity of the fuel. Other than these differences, the general guidance in NUREG-1537 is applicable to AHR fuel.

9a2.3 Fire Protection and Programs

The current guidance in NUREG-1537 should be used for reviewing this section.

9a2.4 Communication Systems

The current guidance in NUREG-1537 should be used for reviewing this section.

9a2.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

The current guidance in NUREG-1537 should be used for reviewing this section.

9a2.6 Cover Gas Control in Primary Coolant Systems

The cover gas control system of an AHR, which is described in Chapter 4, is an integral part of the reactor. Additional information may be included in this chapter. The guidance in NUREG-1537 can be applied here as well.

9a2.7 Other Auxiliary Systems

The current guidance in NUREG-1537 should be used for reviewing this section as applicable.

9b Radioisotope Production Facility Auxiliary Systems

The general guidance in this chapter of NUREG-1537 can be applied to auxiliary systems for a radioisotope production facility provided that a reference to the reactor should be interpreted to mean the radioisotope production facility, as appropriate. The following subsections provide additional guidance applicable to a radioisotope production facility.

9b.1 Heating, Ventilation, and Air Conditioning Systems

The current guidance in this section of NUREG-1537 is applicable to the radioisotope production facility as well as the reactor if the HVAC systems are separate. If the HVAC system is integral and common for both facilities, that fact should be noted, and the description given for the reactor in Section 9a2.1 does not need to be duplicated in this section.

9b.2 Handling and Storage of Reactor Fuel

This section of NUREG-1537 is applicable to a radioisotope production facility provided that any reference to fuel is interpreted as SNM involved in the production process outside of the reactor facility. The applicant or licensee should clearly define that area or component in the facility that separates the reactor from the radioisotope production facility in the SAR. It should be consistent with the divergent requirements to either control a critical reactor or prevent an assembly from becoming critical. Such a reference applies to both irradiated and unirradiated SNM.

9b.3 Fire Protection Systems

This section of NUREG-1537 can be applied to the radioisotope production facility as well as the reactor if the fire protection systems are separate. If the system is integral and common to both facilities, the description given for the reactor in Section 9a2.3 does not need to be duplicated in this section.

9b.4 Communication Systems

This section of NUREG-1537 can be applied to the production facility as well as the reactor if the systems are separate. If the system is integral and common to both facilities, the description given for the reactor in Section 9a2.4 does not need to be duplicated in this section.

9b.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

This section of NUREG-1537 is applicable to the production facility as well as the reactor.

9b.6 Cover Gas Control System

The radioisotope production facility may use a cover gas in the isotope extraction process and in the irradiated SNM storage and treatment system. Gas venting and control apparatus may also be part of the radioisotope extraction and SNM processing equipment. The guidance provided in this section of NUREG-1537 can be applied to the production facility.

10. EXPERIMENTAL FACILITIES

The applicant might provide information about some or all radioisotope processing operations in Part 1 of this chapter. If this chapter is used for this purpose in the SAR, then the reviewer can use this guidance accordingly.

The current content in NUREG-1537, Part 2, is applicable without modification or augmentation to this ISG.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter provides guidance to the technical reviewer about how to conduct a review for radiation protection programs and waste management for a non-power reactor facility and a radioisotope processing facility. The introduction to this chapter in NUREG-1537 is appropriate as written with the clarification that "reactor" or "reactor facility" should be interpreted to mean both the non-power reactor and the radioisotope production facility, as appropriate.

11.1 Radiation Protection

The current wording of this section and the following subsections is adequate without modification of this ISG:

11.1.1 Radiation Sources
11.1.2 Radiation Protection Program
11.1.3 ALARA Program
11.1.4 Radiation Monitoring and Surveying
11.1.5 Radiation Exposure Control and Dosimetry
11.1.6 Contamination Control
11.1.7 Environmental Monitoring

11.2 Radioactive Waste Management

The current wording of this section and the following subsections is adequate without modification of this ISG:

11.2.1 Radioactive Waste Management Program
11.2.2 Radioactive Waste Control
11.2.3 Release of Radioactive Waste

11.3 Respiratory Protection Program

The following guidance for reviewing the respiratory protection program is added to NUREG-1537 for non-power reactors and radioisotope production facilities, as appropriate.

Areas of Review

The areas of review should include detailed information about the following two areas of the respiratory program:

(1) Establishment, maintenance, and implementation of a respiratory protection program.

(2) Design and implementation of programs to control airborne concentrations of radioactive material by using ventilation systems, containment systems, and respirators.

Acceptance Criteria

The applicant should do the following:

(1) Install appropriately sized ventilation and containment systems in areas of the plant identified as having potential airborne concentrations of radionuclides that could exceed the occupational derived air concentration values specified in 10 CFR Part 20, "Standards for Protection against Radiation," Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

(2) Describe surveillance requirements, including preventive and corrective maintenance and performance testing, to ensure that the ventilation and containment systems operate when required and are within their design specifications.

(3) Describe the criteria for the ventilation and containment systems, including minimum flow velocity at openings in these systems, maximum differential pressure across filters, and types of filters to be used.

(4) Describe the frequency and types of tests to measure the performance of ventilation and containment systems, the acceptance criteria, and the actions to be taken when the acceptance criteria are not satisfied.

(5) Establish a respiratory protection program that meets the requirements of 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls To Restrict Internal Exposure in Restricted Areas."

(6) Prepare written procedures for the selection, fitting, issuance, maintenance, testing, training of personnel, monitoring, and recordkeeping for individual respiratory protection equipment and for specifying when such equipment is to be used.

(7) Revise the written procedures for the use of individual respiratory protection equipment, as applicable, when making changes to processing, facility, or equipment.

(8) Maintain records of the respiratory protection program, including training in respirator use and maintenance.

Review Procedures

The reviewer should determine whether the respiratory protection program provides adequate protection of personnel from airborne concentrations exceeding the limits of Appendix B to 10 CFR Part 20 and the overall adequacy of the program. The methods used for the identification and evaluation of potential hazards and estimated doses should provide realistic and accurate predictions. The applicant should evaluate potential hazards and estimated doses by performing surveys, bioassays, air sampling, or other means as necessary.

As for the respiratory protection to be used, the reviewer should ensure that the equipment has been tested and certified to provide the appropriate degree of personal protection. The applicant must also commit to testing of respirators for operability before usage. The reviewer should also examine the description of respirator usage, training, fit testing, selection, storage, maintenance, repair, and quality assurance through the written procedures.

After evaluating the acceptance criteria, the reviewer will perform a safety evaluation. The reviewer will prepare an SER on the licensing action for the licensing project manager.

Evaluation Findings

The reviewer will draft an SER addressing the topic reviewed explaining why the NRC staff has reasonable assurance that the respiratory protection program is acceptable and that the health and safety of the workers is adequately protected. The NRC staff may propose license conditions to impose requirements in those areas in which the application is deficient. The NRC staff's SER will include the following kind of statement and conclusion:

> The applicant has committed to an acceptable radiation protection program that includes a program to control airborne concentrations of radioactive material with engineering controls and respiratory protection.

Change the sequential number for the "References" section.

11.4 References

References in the current NUREG-1537, Part 1, Section 11.3, apply. The following has been updated:

U.S. Nuclear Regulatory Commission, Regulatory Guide 8.13, Revision 3, "Instruction Concerning Prenatal Radiation Exposure," June 1999.

12. CONDUCT OF OPERATIONS

The guidance provided in this chapter is broad in scope to the extent that it can be applied to non-power reactors, regardless of type, and also to radioisotope production facilities. Whereas in some chapters of this ISG, multiple or different chapters were prescribed because of the specificity of the information related to the type of facility involved, the information in this chapter can be applied to all types of reactors, as well as to radioisotope production facilities.

The reviewer should verify that the applicant describes and discusses the conduct of operations at any facility captured in the scope of this ISG. The conduct of operations involves the administrative aspects of facility operations, the facility emergency plan, the security plan, the quality assurance plan, the reactor operator requalification plan, the startup plan, and environmental reports as described in NUREG-1537. Wherever the document refers to "university, corporation, or facility," it should also include "processing facility."

Note that Section 12.13, "Material Control and Accounting," has been added in this ISG for compliance with 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material."

12.1 through 12.6

The current wording of these sections in NUREG-1537 applies to a non-power reactor and radioisotope production facility without augmentation or modification to this ISG.

12.7 Emergency Planning

Emergency planning is a specialized area of review. AHRs should follow the guidance in NUREG-1537. If the facility is a combined non-power reactor and a radioisotope production facility, the NRC staff expects the applicant to provide one emergency plan for the entire site. For the review and evaluation of combined non-power reactors and radioisotope production facilities, the emergency plan review should use NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983, for the reactor and production facility. In addition, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," suggests additional information that should be included for the production facility, as described below.

Section 1.0, "Introduction"

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- Provide a detailed drawing of the site showing the following features:
  - Onsite and near offsite (within 1.61 kilometers (km) (1 mile (mi)) structures with building numbers and labels.
  - Roads and parking lots on site and main roads near the site.
  - Site boundaries showing fences and gates.

  - Major site features.

  - Water bodies within approximately 1.61 km (1 mi) of the site.

- Include a general area map covering a radius of approximately 16.1 km (10 mi), a U.S. Geological Survey topographical quadrangle (7.5-minute series, including the adjacent quadrangles if the site is located less than 1.61 km (1 mi) from the edge of the quadrangle), and a map or aerial photograph indicating onsite and near-site structures within a radius of approximately 1.61 km (1 mi). The map should include the location of sensitive facilities near the site, such as hospitals, schools, nursing homes, nearest residents, fire department, prisons, environmental sampling locations, and other structures and facilities that are important to emergency management.

- Detail the stack heights, typical stack flow rates, and efficiencies of any emission control devices.

- Describe, in general, the licensed and other major activities conducted at the facility and the type, form, and quantities of radioactive and other hazardous materials that are normally on the site, by locations (use and storage), building, and hazardous characteristics (exposure rates, pH, temperature, and other characteristics), that are important to emergency management.

- Provide certification by the plant manager (or the individual authorized by the applicant) that the applicant has met all responsibilities under the Emergency Planning and Community Right To Know Act of 1986 (Title III, Public Law 99-499), in accordance with 10 CFR 70.22 (i)(3)(xiii).

- For each general type of accident identified in the Integrated Safety Analysis (ISA) summary for which protective actions may be needed, the emergency plan should describe the following:

  - The process and physical locations where accidents could occur.

  - Complicating factors and possible onsite and offsite consequences, including releases of nonradioactive hazardous chemicals incident to the processing of licensed material that could impact emergency response efforts.

  - The accident sequence that has the potential for the greatest radiological or toxic chemical impact.

  - Figures projecting doses and toxic substance concentrations as a function of distance and time for various meteorological stability classes, including a description of how the applicant projected such doses or concentrations (e.g., computer models and assumptions).

Section 2.0, “Definitions”

The emergency plan should define words or phrases with meanings specific or unique to the plan, reactor, or production facility.

Section 3.0, “Organization and Responsibilities”

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan should describe who will take the following actions and how he or she will act promptly and effectively:
  - The decision to declare an alert or site area emergency.
  - The activation of the onsite emergency response organization during all shifts.
  - The prompt notification of offsite response authorities that an alert or site area emergency has been declared, including the licensee’s initial recommendation for offsite protective actions (normally within 15 minutes of classification).
  - The notification of the NRC Operations Center (as soon as possible and, in any case, no later than 1 hour after a declared emergency).
  - The decision regarding which onsite protective actions to initiate.
  - The decision regarding which offsite protective actions to recommend.
  - The decision to request support from offsite organizations.
  - The decision to terminate the emergency or enter recovery mode.
- The emergency plan should describe the following aspects of the applicant’s plans for adequately restoring the facility to a safe status after an accident and recovery after an emergency:
  - The methods and responsibilities for assessing the damage to and status of the facility’s capabilities to safely control radioactive material or hazardous chemicals associated with the process.
  - Key positions in the recovery organization.

Section 4.0, “Emergency Classification System”

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan classification system should include the following two classifications for the production facility:

  (1) Alert: Events that may occur, are in progress, or have occurred, that could lead to a release of radioactive material or hazardous chemicals incident to the processing of license material; however, the release is not expected to require a response by an offsite response organization to protect persons offsite.

(2) Site Area Emergency: Events that may occur, are in progress, or have occurred that could lead to a significant release of radioactive material or hazardous chemicals incident to the processing of license material and that could require a response by offsite emergency response organizations to protect persons offsite.

- The emergency plan should identify the classification (alert or site area emergency) expected for each accident identified in the emergency plan.

Section 5.0, "Emergency Action Levels"

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan should specify emergency action levels (EALs) at which an alert or site area emergency will be declared. EALs are specific conditions that require the performance of emergency response measures. The applicant's EALs should be consistent with Appendix A to Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," and should be comparable to the U.S. Environmental Protection Agency's protective action guides described in EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992. Transportation accidents more than 1.61 km (1 mi) from the facility should not be classified.

Section 7.0, "Emergency Response"

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan should describe the following aspects of the applicant's procedures to be used to promptly and effectively assess the release of radioactive material or hazardous chemicals incident to the processing of licensed material:

  – Procedures for estimating or measuring the release rate or source term.

  – Valid computer codes used to project doses or concentrations to the public or environment and their associated assumptions, along with adequate justifications to show the validity of the assumptions.

  – Types, methods, frequencies, implementation times, and other details of onsite and offsite sampling and monitoring that will be performed to assess a release of radioactive materials or hazardous chemicals incident to the processing of licensed material.

  – The method for assessing collateral damage to the facility (including items relied on for safety).

- The emergency plan should provide reasonable assurance that emergency notification procedures will enable the emergency organization to correctly classify emergencies,

notify emergency response personnel, and initiate or recommend appropriate actions in a timely manner, on the basis of the following:

  – Notification procedures minimize distraction of shift operating personnel and include concise, preformatted messages. Appropriate follow-up messages to offsite authorities are issued promptly.

  – Radiological and chemical source term data are available to the command post, technical support center, emergency operation center, and appropriate State personnel in cooperation with the NRC.

  – When available, offsite field monitoring data are logged, compared with source term data, and used in the protective action recommendation process.

  – Protective action guides are available and are used by the appropriate personnel in a timely manner.

- The emergency plan should describe the information to be communicated during an emergency, including the following:

  – A standard reporting checklist to facilitate timely notification.

  – A description of preplanned protective action recommendations to be made to each appropriate offsite organization.

  – Recommended actions to be taken by offsite organizations for each accident treated in the emergency plan.

Section 8.0, “Emergency Facilities and Equipment”

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- For each accident identified in the ISA Summary, the emergency plan should briefly describe measures and equipment to be used for safe shutdown and the mitigation of consequences to workers onsite and offsite and to the public offsite.

- For each type of accident identified, the emergency plan should describe the following:

  – The means of detecting the accident.

  – The means of detecting any release of radioactive material or hazardous chemicals incident to the processing of licensed material.

  – The means of alerting the operating staff.

- The emergency plan should list and describe onsite and offsite facilities that could be relied on in an emergency. The emergency plan should include the following:

- A list and description of both onsite and offsite emergency facilities, by location and purpose.

- A description of emergency monitoring equipment available for personnel and area monitoring, and to assess the release to the environment of radioactive or hazardous chemicals incident to the processing of licensed material.

- A description of the onsite and offsite services that support emergency response operations, including first aid personnel, firefighters, law enforcement assistance, and ambulance services.

Section 9.0, “Recovery”

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan should describe the following aspects of the applicant’s plans for adequately restoring the facility to a safe status after an accident and recovery after an emergency:

  - The procedures for promptly determining the actions necessary to reduce any ongoing releases of radioactive material or hazardous chemicals incident to the processing of licensed material and to prevent further incidents.

  - The provisions for promptly and effectively accomplishing required restoration actions.

Section 10.0, “Maintaining Emergency Preparedness”

The reviewer should verify that the emergency plan supplements the information suggested in NUREG-0849 with the following details regarding the production facility:

- The emergency plan should describe the frequency, performance objectives, and plans for the emergency response training that the licensee will provide to workers. The plan should include the following:

  - The topics and general content of training programs for the licensee’s onsite and offsite emergency response personnel to satisfy the objectives described above.

  - The administration of the training program including responsibility for training, the positions to be trained, the schedule for training, the frequency of retraining, the use of team training, and the estimated number of hours of initial training and retraining.

  - The training to be provided on the use of protective equipment such as respirators, protective clothing, monitoring devices, and other equipment used in emergency response.

  - The training program for onsite personnel who are not members of the emergency response staff.

- Any special instructions and orientation tours that the licensee would offer to fire, police, medical, and other non-licensee emergency personnel who may be required to respond to an emergency to ensure that they know the emergency plan, assigned duties, and effective response to an actual emergency.

- The emergency plan should state the applicant's commitment to conduct exercises and drills in a manner that demonstrates the capability of the organization to plan and perform an effective response to an emergency. An adequate plan should demonstrate the following:

  - Qualified individuals for each position in the emergency response organization demonstrate task-related knowledge through periodic participation.

  - Effective player, controller, evaluator, and observer pre-drill briefings are conducted.

  - Scenario data and exercise messages provided by the controllers effectively maintain the timeline and do not interfere with the emergency organization's response to exercise scenario events, except when safety considerations are involved.

  - The pre-staging of equipment and personnel is minimized to realistically test the activation and staffing of emergency facilities.

  - Emergency drills demonstrate that resources are effectively used to control the site, mitigate further damage, control radiological releases, perform required onsite activities under simulated radiation or airborne and other emergency conditions, accurately assess the facility's status during an accident, and initiate recovery.

  - Emergency drills demonstrate personnel protection measures, including controlling and minimizing hazards to individuals during fires, medical emergencies, mitigation activities, search and rescue, and other similar events.

  - Emergency drills demonstrate that onsite communications effectively support emergency response activities.

  - Emergency drills demonstrate that the emergency public information organization disseminates accurate, reliable, timely, and understandable information.

  - Provisions are made for conducting quarterly communications checks with offsite response organizations.

  - Offsite organizations are invited to participate in the biennial onsite exercise, which tests the major elements of the emergency plan and response organizations.

NOTE: The reviewer should forward emergency planning information and proposed emergency plans to the Radiation Protection and Emergency Preparedness Branch in the Office of Nuclear

Reactor Regulation (NRR), in accordance with the instructions provided in Section 12.7 of NUREG-1537, Part 2.

12.8 Security

The current wording of these sections in NUREG-1537 applies to a non-power reactor and radioisotope production facility without change.

12.9 Quality Assurance

The second sentence of the current section 12.9 of NUREG-1537 should be ignored. Instead, all quality assurance should be sent to the NRC Document Control Desk (DCD) like all other submittals. The DCD will then forward it to the assigned project manager.

12.10 Operator Training and Qualifications

The current wording of this section predominantly addresses training requirements for the non-power reactor staff. This should be titled:

12.10a "Reactor Operator Training and Requalification"

The current wording of this section in NUREG-1537 is applicable to a non-power reactor and radioisotope production facility without requiring changes in this ISG.

A new section should be added pertaining to training in the radioisotope production facility as follows:

12.10b "Production Facility Operator Training and Requalification"

Reviewers should ensure that license applications for radioisotope production facilities contain the technical qualifications, training, and licensing requirements for operators as addressed in the following paragraphs.

The Atomic Energy Act of 1954, Section 107 states: "The Commission shall prescribe uniform conditions for licensing individuals as operators of any of the various classes of production and utilization facilities licensed in this Act." As set out in 10 CFR 50.54(h) and (i) the license is subject to the provisions of the Act, and may not permit the manipulation of the controls of any facility by anyone who is not a licensed operator or senior operator pursuant to the regulations in 10 CFR Part 55, "Operators' Licenses." Although 10 CFR Part 55 only specifies the licensing requirements for utilization facility operators without specifically addressing production facilities, the NRC has determined that the same technical and safety considerations apply to operators of production facilities and so will also apply the Part 55 requirements to production facility operators.

The NRC requires facilities licensed under 10 CFR Part 70 to employ properly qualified and trained staff and to ensure staff are trained on how to respond to emergency situations. NUREG-1520 provides further guidance for training for all personnel who perform activities relied on for safety. The staff should be trained and tested so as to provide reasonable assurance that they understand, recognize the importance of and are qualified to perform those activities relied on for safety in a manner that adequately protects public health and safety and

the environment. As appropriate for their authority and responsibility, personnel should have the knowledge and skills necessary to design, operate, and maintain the facility safely.

In addition to the general and specific training requirements for licensing utilization facility operators and senior operators contained in 10 CFR Part 55, the staff of a radioisotope production facility conducting safety related operations with SNM outside of the reactor should be trained and tested in the following additional basic topics:

- Theory and principles of the radioisotope production processes involving SNM
- Theory and principles of radioisotope extraction and purification processes
- Facility design and operating characteristics
- Instrumentation and control systems
- Engineered safety features
- Technical specifications
- Criticality control features and management measures required for each process involving SNM
- Normal and emergency operating procedures
- ANSI/ANS-15.4-2007 may contain additional guidance on training and qualification of personnel applicable to production facilities.

Regulation 10 CFR 50.54(i-1) requires that within 3 months after an operating license is issued, the licensee have in effect an operator requalification program, which at a minimum meets the requirements of 10 CFR 55.59(c). Regulation 10 CFR Part 55 applies specifically to utilization facilities. With regards to Production Facilities, the operators should comply with the same requirements in 10 CFR 50.54(i-1).

Areas of Review

- Conduct of on-the-job training

- Evaluation of training effectiveness

The review of the training and qualification should address the following areas:

- Organization and management of the training function

- Analysis and identification of functional areas requiring training

- Position training requirements

- Development of the basis for training, including objectives

- Organization of instruction and use of lesson plans and other training guides

- Evaluation of trainee learning

- Personnel qualification

- Provisions for continuing quality assurance, including the needs for retraining or reevaluation of qualification

Acceptance Criteria

The reviewer should verify that the operator training and requalification program is acceptable regarding personnel training and qualification based on the following criteria:

- The program should include the following commitments regarding organization and management of training:
  - Line management is responsible for the content and effective conduct of the training.
  - The program clearly defines job function, responsibility, authority, and accountability of personnel involved in managing, supervising, and implementing training.
  - The program uses performance-based training as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.
  - The program documents and implements procedures to provide reasonable assurance that all phases of training are conducted reliably and consistently.
  - The program ensures that training documents are linked to the configuration management system to provide reasonable assurance that the training reflects design changes and modifications.
  - The program maintains both programmatic and individual training records. These records support management information needs and provide required data on each individual's training and qualification.
- The program should provide formal training for each position or activity that is relied on for safety. Training may be both classroom based and on the job. The application should state the training that will be conducted and identify the personnel that will be required to complete it. The application should also demonstrate the following:
  - The program ensures that each activity selected for training (initial or continuing) from the facility-specific activities is correlated with supporting procedures and training materials.
  - The program reviews facility-specific activities selected for training and compares training materials on an established schedule, updating them as necessitated by changes in procedures, facility systems and equipment, or job scope. The applicant monitors and evaluates change actions (e.g., procedure changes, equipment changes, facility modifications) for their impact on the development or modification of initial and continuing training and incorporates such change actions in a timely manner.
- The program should contain commitments regarding personnel qualification for managers, supervisors, designers, technical staff, construction personnel, facility operators, technicians, maintenance personnel, and other staff who perform regulated activities.

- The program should contain commitments regarding minimum qualifications for personnel. Minimum qualifications should be commensurate with the assigned functional responsibility and authority of the respective personnel as detailed below:

  – Managers should have a bachelor of science (B.S.), bachelor of arts (B.A.), or equivalent degree. Each manager should have either management or technical experience in a facility similar to the facility identified in the application.

  – Supervisors should have at least the qualifications required of personnel being supervised.

  – Technical professional staff whose actions or judgments are critical to satisfying the performance requirements identified in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," should have a B.S., B.A., or equivalent degree in the appropriate technical field.

  – Construction personnel, facility operators, technicians, maintenance personnel, and other staff whose actions are required to comply with NRC regulations should have completed the applicant's training process or have equivalent experience or training.

  – The program should require candidates for process operators to meet the minimum qualifications described in the application. The applicant should require candidates for job functions other than process operators to meet minimum qualifications, but the application need not describe these minimum qualifications.

- The program should include training objectives that state the knowledge, skills, and abilities that the trainee should acquire; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity.

- The program should include lesson plans and other training guides that provide guidance to ensure the consistent conduct of training activities based on required learning objectives derived from specific job performance requirements.

- The program should include standards for evaluating acceptable trainee performance. The evaluation of trainee accomplishment is acceptable if the applicant evaluates trainees periodically during training to determine their progress toward full capability to perform the job requirements and at the completion of training to determine their capability to perform the job requirements.

- The program should establish review and approval requirements for all lesson plans or guides and other training materials before their issuance and use.

- The program should describe any on-the-job training used for activities relied on for safety.

- The program should include on-the-job training using well-organized and current training materials. Designated personnel who are competent in the program standards and training methods should conduct the training.

- The program should use actual task performance to complete on-the-job training. When the actual task cannot be performed and is, therefore, "walked down," (simulated or subjected to a dry run) the conditions of task performance, references, tools, and equipment should reflect the actual task to the extent possible.

- The program should include provisions for continuing assurance of personnel training and qualification through periodic requalification of personnel by training or testing or both, as necessary, to provide reasonable assurance that personnel continue to understand, recognize the importance of, and be qualified to perform activities that are relied on for safety.

- The program should evaluate training effectiveness, that its relationship to job performance remains acceptable, and that there is reasonable assurance that the training conveys the required skills and knowledge based on the performance of trained personnel in the job setting. The application should also demonstrate the following:

  – Qualified individuals should periodically conduct a comprehensive evaluation of individual training to identify strengths and weaknesses. The applicant should use feedback from trainee performance during training and from former trainees and their supervisors to evaluate and refine the training.

  – The applicant should initiate, evaluate, track, and incorporate improvements and changes to initial and continuing training to correct training deficiencies and performance problems.

Review Procedures

Recognizing that the training objectives and methods and the required personnel qualifications may be graded to correspond to the hazard potential of the facility, the reviewer will perform a safety evaluation using the acceptance criteria described above. In particular, the reviewer should:

- Evaluate the adequacy of training and qualification on the basis of how well it fulfills the applicant's training objectives, especially when human factors are relied on for safety.

- Determine whether the applicant has adequately planned for the training and personnel qualification to be accomplished and whether necessary policies, procedures, and instructions will be in place and appropriate training and qualification will be accomplished before personnel begin activities relied on for safety.

- Focus on the training and qualification of personnel who will perform activities relied on for safety.

- Become familiar with the applicant's personnel training and qualification commitments and determine whether ongoing activities correspond to them.

- Determine whether there is reasonable assurance that the applicant's personnel training and qualification efforts will result in only properly trained and qualified personnel performing activities relied on for safety.

Evaluation Findings

A statement similar to the following should be included in this section:

> Based on its review of the license application [insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable], the NRC staff concludes that the applicant has adequately described and assessed its personnel training and qualification in a manner that satisfies the acceptance criteria and is consistent with the guidance in this ISG.
>
> Reasonable assurance exists that implementation of the described training and qualification will result in personnel who are qualified and competent to design, construct, start up, operate, maintain, modify, and decommission the facility safely. The staff concludes that the applicant's plan for personnel training and qualification meets the requirements of 10 CFR Part 50.
>
> Members of the facility operations staff who manipulate the controls of the production facility or who perform other duties that are required to meet the performance requirements stipulated in 10 CFR 70.61(b), 10 CFR 70.61(c), and 10 CFR 70.61(d) are fully qualified and licensed, in accordance with the requirements of 10 CFR 50.54(h) and 10 CFR Part 55, "Operator's Licenses," as augmented by this ISG.

12.11 Startup Plan

The current wording of this section in NUREG-1537 applies to a non-power reactor and radioisotope production facility without change.

Section 12.12 Environmental Reports is due from NRR environmental group however, was not available at the time this DRAFT ISG was issued. Section 12.12 will be issued at a later date. It will be incorporated in the FINAL ISG.

12.13 Material Control and Accounting Plan

The reviewer should ensure that a complete plan is filed in accordance with the requirements of 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material." This section should include the following subsections:

Areas of Review

The reviewer should verify that the application should includes information regarding compliance with Subparts A through E of 10 CFR Part 74, as appropriate, for the class of facility involved. All general and class-specific information should be included as applicable to the following:

- General requirement for all facilities

- Facilities with quantities of low strategic significance
- Facilities with quantities of moderate strategic significance
- Facilities with formula quantities of strategic SNM

Acceptance Criteria

The review shall verify that the material control and accounting plan must contains all of the information prescribed in 10 CFR Part 74 for the specific class of facility contained in the application.

Review Procedures

The reviewer should ascertain that the plan contains a clear, accurate, and thorough account of all inventory, measurement, recordkeeping, and reporting requirements prescribed by the regulations.

Evaluation Findings

The reviewer should be able to conclude the following from the submitted plan:

> The applicant has provided a complete plan that will ensure that all SNM in the facility will be properly accounted for and that will enable the licensee to achieve the specific objectives prescribed in the regulations for the relevant class of material that will be possessed at the facility.

12.14 References

Reference ANSI/ANS 15.1-1990 has been replaced with ANSI/ANS 15.1-2007.

Reference ANSI/ANS 15.4-1988 has been replaced with ANSI/ANS 15.4-2007.

Reference ANSI/ANS 15.8-1976 has been replaced with ANSI/ANS 15.8-2009.

Reference ANSI/ANS 15.11-1993 has been replaced with ANSI/ANS 15.11-2004.

Reference ANSI/ANS 15.16-1978 has been replaced with ANSI/ANS 15.16-2008.

The NRC issued Revision 1 of Regulatory Guide 2.5, “Quality Assurance Program Requirements for Research and Test Reactors,” in June 2010.

13. ACCIDENT ANALYSIS

The current version of NUREG-1537, Chapter 13, addresses accident analyses specifically related to a non-power heterogeneous reactor facility. This ISG augments NUREG-1537 to broaden its application to AHRs and radioisotope production facilities. For this purpose, this ISG chapter is divided into the following subsections:

- Section 13a1, "Heterogeneous Reactor Accident Analysis"
- Section 13a2, "Homogeneous Reactor Accident Analysis"
- Section 13b, "Radioisotope Production Facility Accident Analysis"

Guidance for each of these options follows.

13a1 Heterogeneous Reactor Accident Analysis

Chapter 13 of NUREG-1537, Part 2, applies to heterogeneous non-power reactors and can be used without further guidance.

13a2 Homogeneous Reactor Accident Analysis

13a2.1 Introduction

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

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

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

The MHA is used to demonstrate that the maximum consequences of operating the reactor at a specific site are within acceptable limits. Therefore, a MHA is postulated that results in consequences bounding those of any credible accident likely to occur over the life of the facility. The applicant may perform a sensitivity analysis of the assumptions of the MHA. For example, reactor operating time before accident initiation may be examined to determine the change in MHA outcome if a more realistic assumption is made. Assumptions made in the accident analysis may form the basis for technical specification limits on the operation of the facility. For example, if the accident analysis assumes that the reactor operates for 5 hours a day, 5 days a week, this may become a limiting condition for operation.

The information in this chapter should achieve the objectives stated in this chapter of NUREG-1537, Part 1, by demonstrating that the applicant has considered all potential accidents at the reactor facility and adequately evaluated their consequences. Each postulated accident should be assigned to one of the following categories or grouped consistently according to the type and characteristics of the particular reactor. For AHRs, the following categories are applicable:

- MHA
- Insertion of excess reactivity
- Reduction in cooling
- Mishandling or malfunction of fuel
- Loss of normal electric power
- External events (include natural hazards and manmade events)
- Mishandling or malfunction of equipment
- Large and undamped power oscillations
- Detonation or deflagration of flammable gas mixtures
- Unintended exothermic chemical reactions other than explosion
- Facility system interaction events

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

13a2.2 Areas of Review

Areas of review should include systematic analysis and discussion of credible accidents for determining the limiting event in each category. The applicant may have to analyze several events in a particular accident category to determine the limiting event. This limiting event should be analyzed quantitatively. NUREG 1537, Part 1, Chapter 13, suggests the steps for the applicant to follow once the limiting event is determined for a category of accidents.

13a2.3 Acceptance Criteria and Dose Limits

The safety analysis must meet the requirements set forth in 10 CFR 50.34, “Contents of application; technical information.” In particular, a construction permit application must include a safety analysis report as described in 10 CFR 50.34(a), “Preliminary safety analysis report;” an operating license application must include a safety analysis report as described in 10 CFR 50.34(b), “Final safety analysis report.”

The dose limits in 10 CFR Part 20, Subparts C and D apply. Applicants may reference 10 CFR Part 100, as applicable (power and test reactors), for AHR siting criteria. The applicant should discuss why the MHA is not likely to occur during the operating life of the facility.

13a2.4 Review Procedures

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

- The credible accidents were categorized, and the most limiting accident in each group was chosen for detailed analyses.

- The reactor was assumed to be operating normally under applicable technical specifications before the initiating event. However, the reactor may be in the most limiting technical specification condition at the initiation of the event.

- Instruments, controls, and automatic protective systems were assumed to be operating normally or to be operable before the initiating event. Maximum acceptable non-conservative instrument error may be assumed to exist at accident initiation.

- The single malfunction that initiates the event was identified.

- Credit was taken during the scenario for normally operating reactor systems and protective actions and the initiation of ESFs required to be operable by TSs..

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

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

- The radiation source terms were presented or referenced.

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

The reviewer should confirm that the integrity of the primary boundary will be maintained under all credible accidents analyzed. The primary boundary consists of all structures that prevent the release of fuel and fission products in solution and the fission gases generated during operation. Specifically, they include the vessel containing the fuel; the offgas systems and the waste gas

holding systems; the cooling coils in the vessel; and the associated pumps, valves, heat exchangers, and piping.

The reviewer should determine whether the applicant has categorized and analyzed all credible accidents, in terms of the limiting phenomena identified below, that could pose a challenge to the integrity of the primary boundary in different locations in the facility. The reviewer should also determine whether the applicant has identified the limiting sources and amounts of radionuclides that could be released within the facility or to the outside environment, thereby exposing the facility staff or the general public to radiation. The limiting phenomena for AHRs are expected to be:

- Bulk boiling of fuel solution (i.e., change of phase occurs as liquid evaporates into gas)
- Precipitation of fission products
- Precipitation of fuel (uranium)
- Detonation or deflagration of combustible gas mixtures
- Excessively high radiolytic gas release

The limiting phenomena are analogous to phenomena for heterogeneous non-power reactors that set operational and safety limits. For example, departure from nucleate boiling has been identified as a phenomenon that greatly increases the likelihood of cladding failure in light-water reactors and is useful in deriving quantitative operating and safety limits. Additionally, regulatory limits such as cladding oxidation are tied to the phenomenon of loss of cladding ductility. While the specific safety limits will depend on the reactor design, the reviewer should confirm that the applicant has addressed these limiting phenomena in its definitions of operating and safety limits as well as accident analyses.

Reactivity limits and the functional designs of control and safety-related systems should prevent loss of primary boundary integrity during credible accidents involving insertion of some fraction of excess reactivity. The analyses should include applicable reactivity feedback coefficients and automatic protective actions.

The reviewer should confirm that the applicant has analyzed potential power instabilities, including unstable (growing) power oscillations that are large and undamped. These large, undamped power oscillations could result from positive feedback. The reviewer should confirm that the reactor will return to a stable state such that the integrity of the primary boundary is not challenged.

The reviewer should confirm that loss of normal electrical power and consequent reduction in cooling will not lead to a challenge to the primary boundary. A loss of normal electric power should not compromise safe reactor shutdown.

13a2.5 Evaluation Findings

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

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

This section of the SAR should contain sufficient information to support the types of conclusions given below. The staff's SER will include those conclusions. The appropriate number for the reactor under evaluation should replace the notation "xx" and "yy." The reviewer should modify these conclusions to conform to the reactor design under consideration.

13a2.5.1 Maximum Hypothetical Accident

The reviewer should determine whether the following finding is applicable:

> The applicant has considered the consequences to the public of all credible accidents at the reactor facility. An MHA, which is an accident that would release fission products from fuel and would have consequences greater than any credible accident, has been analyzed. The MHA scenario is credible but the combination of bounding conditions analyze are, not credible. However, the MHA serves as a bounding accident analysis for a non-power reactor. [The MHA is specific to the reactor design and power. The reviewer may have to evaluate an MHA that differs from the suggested list of MHAs below.]

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

- Energetic dispersal of the contents of the primary boundary with bypass of any scrubbing capacity (e.g., a pool surrounding the fuel vessel).

- Detonation of hydrogen in the recombiner resulting in waste gas tank failure and release of some or all of the fuel and fission product contents in aerosolized form.

- Complete loss of fuel inventory (e.g., vessel break).

Possible MHAs for a multireactor AHR facility could be one or a combination of the following events:

- A manmade external event that breaches the primary boundary of more than one unit.
- A facility-wide external event that breaches various systems containing radioactive fluids.

The reviewer should modify the following paragraphs, as appropriate:

The air handling and filtering systems (i.e., confinement or containment) are assumed to function as designed, and radioactive material is held up temporarily in the reactor room and then released from the building. Realistic methods are used to compute external radiation doses and dose commitments resulting from inhalation by the facility staff. Realistic but conservative methods are used to compute potential doses and dose commitments to the public in the unrestricted area. Methods of calculating doses from inhalation or ingestion (or both) and direct shine of gamma rays from dispersing plumes of airborne radioactive material are

applicable and no less conservative than those developed in Chapter 11 of the SAR. The duration of the accident considered for the calculation of doses for the facility staff is (*xx)* and for the public it is (*yy)-"*

The calculated doses for the MHA scenario are the following:

Licensee staff – xx mrem

Maximum exposed member of the public – yy mrem

Nearest residence – xx mrem

These doses and dose commitments are within the acceptable limits [state limits]. Because the assumptions of the scenario are bounding, the doses calculated will likely not be exceeded by any accident considered credible. The applicant has examined more realistic assumptions about operating time and release fractions that decreased the source term by "xx" percent of the one calculated, lowering the maximum doses by that factor (if applicable). Thus even for the MHA, whose consequences bound all credible accidents possible at the facility, the health and safety of the facility staff and the public are protected.

13a2.5.2 Insertion of Excess Reactivity

The reviewer should determine if the following finding is applicable:

> The applicant has considered the following initiators that could insert excess reactivity in AHRs:
>
> - Pressurization of the fuel fluid
> - Excessive cooldown via heat sink malfunction
> - Moderator injection resulting from cooling system malfunction (e.g., cooling tube rupture)
> - Fuel injection
> - Realistic, adverse geometry changes (e.g., those caused by "sloshing" of the fuel solution in the vessel
> - Reactivity insertion from moderator lumping effects (e.g., voiding in the cooling coil)
> - Inadvertent introduction of other material into the fuel solution (e.g. excessive acid addition)
> - Control rod removal or ejection, or system or experiment malfunction

The limiting reactivity insertion event has been identified and analyzed and the consequences of this event are within the dose acceptance limits and bounded by the MHA. The reactor attains a stable condition following the limiting event. Radiation doses to the public and staff are thus within acceptable limits and the safety and health of the staff and public are adequately protected.

13a2.5.3 Reduction in Cooling

The reviewer should determine if the following finding is applicable:

> The applicant has considered postulated events that lead to reduction or loss of cooling. The following initiators have been analyzed:
>
> - Loss of electrical power
> - Failure of active components in the normal heat removal system
> - Cooling coil or heat exchanger tube rupture
> - Flow obstruction in heat exchangers
> - Loss of forced circulation
> - Recombiner burnout

The consequences of reduction in cooling events have been analyzed and shown to be bounded by the MHA. Radiation doses to the public and staff will be within acceptable limits, and the safety and health of the staff and public will be adequately protected.

13a2.5.4 Mishandling or Malfunction of Fuel

The reviewer should determine if the following finding is applicable.

The applicant considered the consequences of fuel solution mishandling events, such as excessive leakage or spillage that could potentially initiate an unintended criticality event in an area or location where it could pose a threat to facility staff. The MHA bounds the accidental dose consequences of such postulated events. Therefore, doses to the staff and the public will be within acceptable limits, and the health and safety of the staff and public will be adequately protected.

The applicant considered the consequences of fuel malfunction events for an AHR. These events include failure to control pH, temperature, or pressure of the fuel solution, which can impact the physical or chemical form of the fuel or solvent resulting in adverse chemical effects, such as fuel precipitation or excessive corrosion. The MHA bounds the accidental dose consequences of such postulated events. Therefore, doses to the staff and the public will be within acceptable limits, and the health and safety of the staff and public will be adequately protected.

13a2.5.5 Loss of Electrical Power

The reviewer should determine if the following finding is applicable:

The applicant's analysis considered the effects of radiolytic decomposition of the fuel solution and the formation of fission gases, addressed the system response to gas formation, and evaluated the potential for the decomposed gases to react explosively.

The applicant's analyses were carried out for a sufficient duration to demonstrate that the reactor reaches a stable state and that the MHA bounds the accidental dose consequences of such postulated events. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and public are adequately protected.

13a2.5.6 External Events

The reviewer should determine if the following finding is applicable:

Chapters 2 and 3 of the SAR discuss the design of the reactor facility and its ability to withstand external events and the potential associated accidents. The reactor facility is designed to accommodate these events by shutting down, which would not pose undue risk to the health and safety of the public. For events that cause facility damage, the damage is within the bounds discussed for other accidents in this chapter. Therefore, exposure to the staff and the public is within acceptable limits, and external events do not pose an unacceptable risk to the health and safety of the public.

13a2.5.7 Mishandling or Malfunction of Equipment

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

13a2.5.8 Large Undamped Power Oscillations

Reactivity feedback coupled with plant response could yield conditions that are not inherently stable. For example, positive feedback due to radiolytic gas formation and vessel pressurization under conditions where the recombiner capacity of an AHR is exceeded could result in conditions that are not inherently stable. Conditions where positive feedback is possible should be examined to determine if the reactor remains stable; or, if the reactor becomes unstable and the power oscillations grow over time, that these unstable power oscillations can be acceptably detected and suppressed.

The reviewer should determine if the following finding is applicable.

The applicant has evaluated potential unstable (growing), large undamped power oscillations and demonstrated that these oscillations are either not possible, or, if they develop, can be readily detected and suppressed so that the reactor reaches a stable state. The applicant considered the potential for positive feedback to arise due to system interaction. Power oscillations should be stable or can be readily detected and suppressed so that the consequences are bounded by the MHA. Therefore, doses to the staff and the public are within acceptable limits and the health and safety of the staff and public are adequately protected.

13a2.5.9 Detonation and Deflagration

The reviewer should determine if the following finding is applicable:

The applicant has evaluated the consequences of potential deflagration or detonation of combustible gases within the primary boundary. The assumptions regarding the impact of potential explosions on primary boundary integrity are valid. The following items have been evaluated: mechanical impact of the explosion in terms of primary barrier integrity; the core response in terms of dynamic fuel response (including potential fuel aerosolization); and reactivity response to any impinging pressure waves. The MHA bounds the consequences of the limiting credible detonation within the primary boundary. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13a2.5.10 Unintended Exothermic Chemical Reactions Other Than Explosion

The reviewer should determine if the following finding is applicable:

The applicant has evaluated the consequences of potential unintended exothermic chemical reactions (other than explosions) that could occur within the primary boundary. As a precursor to this event, an excess of gases could accumulate in the primary boundary and subsequently react with oxygen to release heat. The heat could increase pressure within the primary boundary or induce thermal stress on the primary boundary. The applicant identified the types of possible exothermic reactions and assessed the relative consequences.

The MHA bounds the consequences of the limiting unintended exothermic chemical reaction (other than detonations as discussed in Section 13.5.9) within the primary boundary. Therefore, doses to the staff and public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13a2.5.11 Facility System Interaction Events

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

For radioisotope production facilities, the reviewer should determine if the following finding is applicable:

The applicant's analysis has considered potential system interactions between the reactor and the isotope production facility. The MHA bounds the accidental dose consequences of these postulated events. Therefore, doses to the staff and the public are within acceptable limits, and the health and safety of the staff and public are adequately protected.

13b Accident Analyses of Operations with Special Nuclear Material, Radioisotopes, and Hazardous Chemicals outside of the Reactor

13b.1 Introduction

In this section, the applicant should present a methodology for reviewing the systems and operating characteristics of the facility that could affect safe operation or shutdown. The methodology should be used to identify limiting accidents, analyze the evolution of the

scenarios, and evaluate the consequences. The analyses should include the assumed initiating event and the effects on designed barriers, protective systems, and operator responses. The mitigating features should also be examined. The endpoint should be a stable facility. The potential radiological consequences to the public, the facility staff, and the environment should be analyzed. The information and analyses should show that facility system designs, safety limits, limiting control settings, limiting conditions for operation, and surveillance requirements (technical specifications) were selected to ensure that the consequences of analyzed accidents do not exceed acceptable limits.

For a radioisotope production facility, the safety analysis must meet the requirements set forth in 10 CFR 50.34, "Contents of application; technical information." In particular, a construction permit application must include a safety analysis report as described in 10 CFR 50.34(a), "Preliminary safety analysis report;" an operating license application must include a safety analysis report as described in 10 CFR 50.34(b), "Final safety analysis report."

For a radioisotope production facility, the applicant should discuss and analyze an accident scenario with consequences exceeding all credible accidents (i.e., the MHA). The accident analyses of processes involving SNM, radioisotopes, and chemicals outside of the shall, as required by 10 CFR Part 70, Subpart H, and consistent with the guidance found in NUREG-1520, include ISAs for processes involving greater than critical mass (as defined in 10 CFR 70.4) quantities of SNM. The performance criteria in 10 CFR 70.61, "Performance Requirements," categorize accidents according to severity of consequences. Accidents resulting in high consequence, as defined by certain radiological doses or adverse health effects from chemical toxicity, could be considered analogous to an MHA. These performance criteria require that high-consequence accidents to be rendered highly unlikely to occur through the application of in-depth preventive and mitigative measures. Other accidents with less than high consequences, as defined in 10 CFR 70.61, require fewer or less-strict protective measures.

Performance of ISAs of potential accident sequences should identify structures, systems, components, and management measures that will become items relied on for safety (IROFS). The licensees may identify additional protective measures that they become aware of outside of the ISA process that should be designated as IROFS. The license technical specifications should include IROFS that are identified under 10 CFR 70.61(e) and other protective or mitigative measures that meet the criteria for technical specifications, as defined in 10 CFR 50.36, "Technical Specifications."

13b.1.1 Operations Conducted outside of the Reactor

The information in this section (13b, part 2) should provide the reviewer the assurance that the objectives stated in Part 1 of this section in NUREG-1537, Part 1, have been achieved. All potential accidents at the facility have been considered and their consequences adequately evaluated. At a minimum, the applicant has considered postulated accidents in all of the categories identified below. The information for a particular facility may show that some of the categories do not apply or that certain operations in a facility may warrant the assignment of additional categories.

Processes that are conducted outside of the reactor in a radioisotope production facility and that should be analyzed under accident conditions are divided into three general categories:

(1) Operations with SNM

– Irradiated fuel processed for radioisotope extraction
– Irradiated fuel processed for reuse in the reactor or for waste disposal
– Operations with unirradiated SNM

(2) Radiochemical operations
(3) Operations with hazardous chemicals

13b.1.2 Accident Initiating Events

The ISAs for the above operations and any other planned operations with SNM should include the following initiating events:

- Criticality accident (could be MHA)
- Loss of electrical power
- External events (meteorological, seismic, fire, flood)
- Critical equipment malfunction
- Operator error

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

Areas of Review

Areas of review should include a systematic analysis and discussion of credible accidents for determining the limiting event in each category. The applicant may have to analyze several events in a particular accident category to determine the limiting event. This limiting event should be analyzed quantitatively. The steps suggested for performing the analysis of accidents are given in NUREG-1537, Part 1.

The reviewer should determine the following:

- The applicant is fully committed to implementing a safety program, including ISAs (in accordance with the requirements of 10 CFR 70.62, “Safety Program and Integrated Safety Analysis,”) and the management measures prescribed in Section 12.1.6 of this ISG.

- The applicant has addressed all credible accidents involving internal facility and process abnormal events and process deviations and credible external events that could result in serious adverse consequences to the staff, the facility, the public, and the environment.

- The applicant has identified designated engineered and administrative IROFS necessary to provide preventive or mitigative measures that give reasonable assurance that the performance requirements of 10 CFR 70.61.

Acceptance Criteria

For a radioisotope production facility, the results of the accident analysis must meet the occupational dose limits in 10 CFR 20.1201, “Occupational Dose Limits for Adults,” and public dose limits in 10 CFR 20.1301, “Dose Limits for Individual Members of the Public.”

The applicant should discuss why the MHA is not likely to occur during the operating life of the facility.

NUREG-1520, Section 3.4, provides additional criteria for adherence to the safety program and ISA performance.

Review Procedures

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

- The credible accidents were categorized, and the most limiting accident in each group was chosen for detailed analyses.

- The process was assumed to be operating normally under applicable technical specifications before the initiating event. However, the process may be in the most limiting technical specification condition at the initiation of the event.

- Instruments, controls, and automatic protective systems were assumed to be operating normally or to be operable before the initiating event. Maximum acceptable non-conservative instrument error may be assumed to exist at accident initiation.

- The single malfunction that initiates the event was identified.

- Credit was taken during the scenario for normally operating process systems. Protective actions were initiated by either the operating staff, control systems, or ESFs.

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

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

- The radiation source terms were presented or referenced.

- The potential radiation consequences to the facility staff and the public were presented and compared with acceptance criteria (see previous section).

Evaluation Findings

It is essential that all credible accidents at a non-power reactor and radioisotope production facility be considered and evaluated during the design stage. The safety margins should be adequate to prevent the release of radioactive materials, in amounts exceeding regulatory limits, to uncontrolled areas as a result of credible accidents. For potential accidents and the MHA that could cause a release, the acceptance criteria and review procedures discussed above are

sufficiently comprehensive. However, the potential consequences, detailed analyses, evaluations, and conclusions are facility and accident specific. Typical findings for the accident categories are presented below. These findings are examples only. The appropriate number for the facility under evaluation should replace the notation “xx” and “yy.” The reviewer should modify the actual wording of the findings as appropriate to the situation under review.

This section of the SAR should contain sufficient information to support the types of conclusions given below. The staff's SER will include those conclusions.

13b.1.3 Evaluation Findings for Specific Accident Scenarios

This section presents examples of some typical accident scenarios that the SER should include. The SER should include other credible accident situations identified by the applicant as they pertain to any other unique processes conducted in the facility and that may have radiological consequences.

Maximum Hypothetical Accident

> The applicant has considered the consequences to the public and the staff of all credible accidents at the radioisotope production facility. A maximum hypothetical accident (MHA), which is an accident that would release fission products from fuel in process (or some other accident with equal or worse consequences) with consequences greater than any credible accident, has been analyzed. The MHA, however, is not considered to be a credible event for this facility.

This section of the SAR (ISA) should contain sufficient information to support the types of conclusions suggested below, which the SER will include. The reviewer should modify conclusions appropriately to conform or relate to the specifics of the accident condition under review.

> The containment-confinement-engineered safety features are assumed to function as designed and radioactive material is held up temporarily in the facility and then released under controlled conditions. Realistic but conservative methods are used to compute potential doses and dose commitments to the public in uncontrolled areas and to compute external radiation doses and dose commitments resulting from inhalation by the facility staff. . Methods of calculating doses from inhalation or ingestion (or both) and direct exposure to gamma rays from dispersing plumes of airborne radioactive material are applicable and no less conservative than those developed in Chapter 11 of the SAR. The exposure time for the staff is “xx” and for the public it is “yy”.
>
> The calculated maximum effective doses for the MHA scenario are the following:
>
> > “xx” mrem total effective dose equivalent (TEDE) for the staff;
> > “yy” mrem TEDE for the public (maximally exposed individual at the controlled area boundary)
>
> These doses and dose commitments are within the acceptable limits [state limits]. Because the assumptions of the scenario are conservative, the postulated accident would not be likely to occur during the life of the facility. The applicant has examined more realistic assumptions about operating time and release fractions that decreased the source term by “xx” percent of the one calculated, lowering the maximum doses by that factor [if applicable]. Thus, even in the event of the MHA, whose consequences bound all credible accidents at the facility, the health and safety of the facility staff and the public are protected.

Accidents While Processing Irradiated and Unirradiated Special Nuclear Material

The applicant has discussed initiating events that could accidentally release fission products from irradiated fuel while in process, in storage, or while being transferred within the facility. The applicant has discussed the potential for a criticality incident with unirradiated SNM. The applicant has analyzed the event that would cause the worst radiological consequences. This event is [provide description].

The analysis shows that the consequences of this event are bounded by the incident discussed under the MHA. Therefore, doses to the staff and the public are within acceptable limits.

Accidents While Processing Fission Products or Other RAM

The applicant has discussed the types of processes that could be performed in the separation and purification of radioisotopes under the facility license and technical specifications. The discussions include events that could initiate accidents such as [list events (examples are given below)]:

- Process equipment malfunction
- Operator error
- External events

The analysis shows that the technical specifications that limit the types and amounts of RAM in process and that provide engineered and administrative features and controls give reasonable assurance that the potential consequences of these events would be less severe than those already evaluated in the section on the MHA or in other, more serious, accident scenarios.

Loss of Normal Electrical Power

The applicant has discussed the events (detailing the site-specific responses) that could result from the sudden loss of normal electrical power. All safety-related equipment is either supplied with adequate emergency power or is, by default, returned to a safe condition in a de-energized state. Any requirement for emergency cooling or ventilation functions is provided as intended in the facility design.

The reviewer should modify the following statement to apply to the actual circumstances enumerated in the SAR:

The applicant has demonstrated that the loss of normal electrical power will not result in an unsafe condition for either the facility staff or members of the public in uncontrolled areas. Chapter 8 of the SAR describes emergency power to the facility. The emergency supply will power the safety-related equipment and systems required to operate after the loss of normal power.

External Events

> Chapters 2 and 3 of the SAR discuss the design of the structures, systems, and components to withstand external events and the potential associated accidents. The reactor–radioisotope production facility is designed to withstand the effects of these events. Process operations could continue provided that there would not be undue risk to the health and safety of the staff, the public, and the environment. Consequences of natural external events that cause facility damage (e.g., seismic events that damage the confinement or containment) are within the bounds discussed for other accidents in this chapter. Therefore, exposure to the staff and the public is within acceptable limits, and external events do not pose an unacceptable risk to the health and safety of the public. [An external event could be the MHA if enough damage is done to the facility. The conclusion for the MHA above would apply.]

Mishandling or Malfunction of Equipment

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

13b.2 Chemical Process Safety for the Radioisotope Processing Facility

The staff's chemical process safety review should focus on chemical safety-related accidents, chemical safety controls, and the corresponding surveillance requirements to ensure that the applicant's equipment and facilities are adequate to protect against releases and chemical exposures of licensed material, hazardous chemicals produced from licensed material, and chemical risks of plant conditions that affect the safety of licensed material.

The 1988 memorandum of understanding between the NRC and the Occupational Safety and Health Administration directs the NRC to oversee chemical safety issues related to (1) radiation risks of licensed materials, (2) chemical risks of licensed materials, and (3) plant conditions that affect or may affect the safety of licensed materials and thus increase radiation risk to workers, the public, and the environment. The NRC does not oversee plant conditions that do not affect or involve the safety of licensed materials.

Areas of Review

The staff's review should cover the following specifications:

Chemical Process Description

The chemical process descriptions and chemical accidents determined by the applicant are the bases for the chemical process safety evaluation. The reviewer should establish that the applicant's facility design, operations, and safety controls for chemical safety provide reasonable assurance that they will function as intended and ensure the safe handling of licensed material at the facility. The reviewer must verify that the applicant's proposed equipment and facilities are adequate to protect public health and safety and the environment. The reviewer should examine the mechanisms that will allow the applicant to identify and correct potential problems.

Chemical Accident Description

The applicant should describe the potential accidents caused by process deviations or other events internal to the facility and credible external events, including natural phenomena. The reviewer should assess the chemical risks to ensure that the design and the operational plans for the facility reflect the level of safety. In return, to validate the criteria used by the applicant in reporting accidents, the reviewer will make an independent judgment of the comparative risks assigned by the applicant to accidents identified. This judgment is based on risk relative to other sequences of events (competing risks), the complexity of the accidents, facility operating history, and general industry performance. Whenever possible, the applicant should use its own experience to supplement the identification of potential chemical hazards. The reviewer may consider a selected number of lower-risk chemical safety-related accident sequences that were not identified by the applicant and provided in the application

Chemical Accident Consequences

The reviewer should verify that the proposed quantitative standards used to assess the consequences to an individual and the public from acute chemical exposure are appropriate. Events with high potential consequences should be identified; controls should be used to reduce the likelihood or the consequences of the event. The reviewer should ensure that the select standards are correctly applied to the worker or the member of the public.

Chemical Process Safety Controls

The staff will review the chemical process ESFs or technical specifications, or both, to ensure their adequacy in protecting against all unmitigated accidents identified by the applicant. The reviewer should establish that the applicant's controls for chemical safety provide reasonable assurance that they will function as intended and ensure the safe handling of licensed material at the facility.

Chemical Process Surveillance Requirements

- The application should include a chemical safety element describing the methods, activities, and implementation of the overall safety program. The technical reviewer should verify the applicant's commitment to retaining records for chemical process safety compliance and reporting commitments for chemical releases. In addition, the reviewer should verify the applicant's commitment to refer any unacceptable performance deficiency to those responsible for the facility's corrective action function.

- If the applicant has applied a graded approach to safety, the reviewer should establish that the grading of controls or management measures is appropriate and sufficient to protect against chemical process risks.

Acceptance Criteria

Acceptance criteria are based on meeting the relevant requirements of the following regulations:

- The general contents of an application for chemical process safety information are in 10 CFR 50.34, “Contents of applications; technical information.” The section outlines the general information that must be included in the license application.

- The requirements for the approval of the application are in 10 CFR 50.45, “Standards for construction permits, operating licenses, and combined licenses.”

Regulatory Acceptance Criteria

The reviewer should find the applicant’s chemical process safety information acceptable if there is reasonable assurance that it adequately addresses and satisfies the acceptance criteria presented below.

Section 70.64 requires a description of each process in the facility. The applicant’s descriptions of the chemical processes are acceptable if they meet the following conditions:

- Process descriptions are sufficiently detailed to allow an understanding of the chemical process hazards (including radiological hazards caused by or involving chemical accidents) and to allow the development of potential accidents.

- Process descriptions are sufficiently detailed to allow an understanding of the theory of operation.

Review Procedures

During the safety evaluation, the reviewer determines whether the application comprehensively describes the chemical safety of the licensed activity. For deviations from the specific acceptance criteria, the staff should review the applicant’s explanation of how the proposed alternatives to the Standard Review Plan criteria provide an acceptable method of complying with the relevant NRC requirements.

During the license application review, the reviewer should identify and note any items or issues that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the applicant implemented ESFs or technical specifications, or both, through procedures and operator training.

For an existing facility, the reviewer should consult NRC inspectors to identify and resolve any issues related to the licensing review. For a planned facility, the reviewers should consult with the facility design team to gain a better understanding of the process, its potential hazards, and its safety approaches. The reviewer should coordinate these interactions through the licensing project manager.

The primary reviewer will prepare input to the SER for the licensing project manager in support of the licensing action.

Evaluation Findings

The reviewer’s input to the SER should address each topic reviewed and explain why the NRC staff has reasonable assurance that the chemical safety portion of the application is acceptable. The reviewer may propose license conditions to impose requirements for those areas in which

the application is deficient. If unable to make a finding of reasonable assurance, the reviewer will prepare input to the SER explaining the deficiencies and the reasons for denying the proposed application. In cases in which the SER is drafted in advance of resolving all outstanding chemical process safety issues, the reviewer should document the review as described below and include a list of open issues that require resolution before the staff can make a finding of reasonable assurance. For partial reviews, revisions, and process changes, the reviewer should use applicable sections of the acceptance criteria and the SER and note areas that were not reviewed and their chemical process safety significance, if any. On completion of the review, the NRC staff can impose temporary license conditions to authorize short-duration activities. For certain functions and requirements that concern safety or regulatory issues, the NRC can impose a license condition that remains in effect until removed by an amendment or license renewal.

The SER should include a summary statement of what the NRC evaluated and the basis for the reviewer's conclusions. The SER should include statements like the following:

> The NRC staff has evaluated the application using the criteria listed previously. Based on the review of the license application, the NRC staff has concluded that the applicant has adequately described and assessed accident consequences that could result from the handling, storage, or processing of licensed materials and that could have potentially significant chemical consequences and effects. The applicant has constructed a hazard analysis that identified and evaluated those chemical process hazards and potential accidents and established safety controls to provide reasonable assurance of safe facility operation. To ensure that the requirements for acceptability are met, the applicant has provided reasonable assurance that ESFs and technical specifications are available and reliable when required to perform their safety functions. The staff has reviewed these safety controls and the applicant's plan for managing chemical process safety and finds them acceptable.
>
> The NRC staff concludes that both the applicant's plan for managing chemical process safety and the chemical process safety controls meet the requirements of the regulations and provide reasonable assurance that the health and safety of the public will be protected.

Bibliography

Nonpower Reactors

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

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

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

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

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

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

Behavior of Zirconium-Hydride Fueled Reactors (TRIGA)

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

Baldwin, N.L., Foushee, F.C., and Greenwood, I.S., *“*Fission Product Release from TRIGA-LEU Reactor Fuels,” *Seventh Biennial U.S. TRIGA Users Conference, San Diego, CA,* 1980.

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

Foushee, F.C., and Peters, R.H., “Summary of TRIGA Fuel Fission Product Release Experiments,” Gulf-EES-AI0801, September 1971.

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

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

Lindgren, I.R., and Sinmad, M.T., “Low-Enriched TRIGA Fuel Water-Quench Safety Tests,” *Transactions of the American Nuclear Society*, 33:276, 1979.

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

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

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

Simnad, M.T., Foushee, F.C., and West, G.B., “Fuel Elements for Pulsed TRIGA Research Reactors,” *Nuclear Technology,* 28:31–56, 1976.

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

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

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

Behavior of Plate Fueled Reactors (MTR)

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

Forbes, S.G., et al., “Instability in the SPERT I Reactor, Preliminary Report,” 100-16309, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, October 1956.

Knexevich, M., et al., “Loss of Water at the Livermore Pool Type Reactor,” *Health Physics,* 11:481–487, 1965.

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

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

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

Shibatta, T., et al., “Release of Fission Products from Irradiated Aluminide Fuel at High Temperatures,” *Nuclear Science and Engineering*, 87:405–417, 1984.

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

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

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

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

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

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

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

Radiological Consequences

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

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

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

14. TECHNICAL SPECIFICATIONS

The current wording of this chapter in NUREG-1537, Part 2, is limited to reviewing the technical specifications for heterogeneous non-power reactors. This ISG describes changes that expand the NUREG so that it also applies to aqueous homogeneous non-power reactors and radioisotope production facilities. The basic change in this chapter involves the division of NUREG-1537, Part 2, into the following sections:

- Section 14a1, "Heterogeneous Reactor"
- Section 14a2, "Aqueous Homogeneous Reactor"
- Section 14b, "Radioisotope Production Facility"

Guidance for each of these options follows.

14a1 Heterogeneous Reactor Technical Specifications

The reviewer should follow the current guidance in Part 2 of NUREG-1537.

14a2 Aqueous Homogeneous Reactor Technical Specifications

The reviewer should use the current guidance in Part 2 of NUREG-1537 as it would apply to an AHR. For example, the following change to the evaluation findings would be appropriate:

In the affirmation in the first bullet under *Evaluation Findings*, change the NUREG reference from "NUREG-1537 (Part 1)" to "NUREG-1537 (Part 1) as supplemented by this ISG."

14b Technical Specifications of Processes Involving Special Nuclear Material, Radioisotopes, and Chemicals outside of the Reactor

This section provides guidance for reviewing and evaluating the technical specifications submitted for processes involving SNM, radioisotopes, and chemicals outside the reactor. These technical specifications include the IROFS that were identified in the SAR and the ISAs in Chapter 13. The technical specifications (safety limits, limiting control settings, and limiting conditions for operation) for a radioisotope production facility are required as license conditions to comply with the performance criteria in 10 CFR 70.61 and 10 CFR 50.36 using a graded approach, as described in the introduction to Section 13b in Part 1 of this ISG. The technical specifications also include the management measures that are established to ensure that these specified IROFS are designed, implemented, and maintained so that they are available and reliable to perform their functions when needed.

Areas of Review

The reviewer should verify that the ISA Summary provided in accordance with Section 13b, Part 1, of this ISG and the SAR. The staff should review the applicant's method of estimating the consequences of accidents so that all IROFS that should be technical specifications have been identified. The staff should also review the following:

- Safety limits proposed by the applicant for the IROFS that are technical specifications.

- Limiting control settings for active engineered IROFS that are technical specifications.
- Limiting conditions for operation.
- Surveillance requirements.
- Design features that affect the function, availability, or reliability of IROFS that are technical specifications.
- Management measures to ensure that IROFS that are technical specifications will be available when needed.
- Corrective actions proposed to be taken if any specifications are exceeded or breached.

Acceptance Criteria

Technical specifications for radioisotope production operations are acceptable if they meet the following criteria:

- The technical specifications include a list of all IROFS for all processes involving SNM, radioisotopes, and hazardous chemicals (associated with operations with SNM) that prevent or mitigate accident consequences.
- The technical specifications comply with the performance criteria of 10 CFR 70.61 and 10 CFR 50.36.
- The technical specifications are presented in the format prescribed in Appendix 14.1 to NUREG-1537, Part 1, and the guidance of this ISG.
- The technical specification limits for the facility design, construction, and operation provides reasonable assurance that the facility can be operated without endangering the environment and the health and safety of the staff and the public.

These acceptance criteria are explained in greater detail below.

The list describing the IROFS that are technical specifications documents the safety basis of all processes in the facility necessary to comply with the performance criteria in 10 CFR 70.61 and 10 CFR 50.36. This list assists in ensuring that the IROFS that are technical specifications are not degraded without a justifying safety review. Thus, the key feature of this list is that it includes all IROFS that are technical specifications. However, sets of hardware or procedures that perform the same safety function may be referred to as a single set of IROFS and do not need to be individually identified.

IROFS that are technical specifications may be hardware with a dedicated safety function or hardware with a property that is relied on to meet the performance criteria of 10 CFR 70.61 and 10 CFR 50.36. Thus, IROFS that are technical specifications may be the dimension, shape, capacity, or composition of hardware. The technical specifications need not provide a breakdown of hardware IROFS by component or identify all support systems. However, the technical specifications documentation maintained on site, such as system schematics or

descriptive lists, should contain sufficient detail about items within a hardware IROFS such that it is clear to the reviewer and the applicant what structure, system, equipment, or component is included within the hardware IROFS' boundary and would, therefore, be subject to management measures specified in the technical specifications. Some examples of items within hardware IROFS are detectors, sensors, electronics, cables, valves, piping, tanks, and dikes. In addition, ISA documentation should also identify essential utilities and support systems on which the IROFS depends to perform its intended function. Some examples of these are backup batteries, air supply, and steam supply. In some processes, the frequency of demands made on IROFS must be controlled or limited to comply with 10 CFR 70.61. In such processes, whatever features are needed to limit the frequency of demands are themselves IROFS.

The essential features of each IROFS needed to achieve the performance criteria of 10 CFR 70.61 and 10 CFR 50.36 must be described. Sufficient information should be provided about engineered hardware controls to permit an evaluation that ensures, in principle, that controls of this type will have adequate reliability. The likelihood of failure of items often depends on safety margins; therefore descriptions of the safety parameter controlled by the item, the safety limit on the parameter, and the margin to true failure may be needed. For IROFS that are administrative controls, the nature of the action or prohibition involved must be described sufficiently to permit an understanding that, in principle, adherence to it should be reliable. Features of the IROFS that affect its independence from other IROFS, such as reliance on the same power supplies, should be indicated.

The description of each IROFS needed to meet the performance criteria of 10 CFR 70.61 and 10 CFR 50.36 should identify its expected function, conditions needed for the IROFS to reliably perform its function, and the effects of its failure. The description of each IROFS within an ISA Summary should identify the management measures, such as maintenance, training, and configuration management that are applied to it. If a system of graded management measures is used, the grade applied to each control should be determinable from information in the ISA Summary. The reliability required for an IROFS is proportionate to the amount of risk reduction it is expected to provide. Thus, the quality of the management measures applied to an IROFS may be graded commensurate with the required reliability. The management measures should ensure that IROFS are designed, implemented, and maintained, as necessary, to be available and reliable to perform their function when needed. The degree of reliability and availability of IROFS ensured by these measures should be consistent with the evaluations of accident likelihoods. In particular, for redundant IROFS, all information necessary to establish the average vulnerable outage time is required in order to maintain acceptable availability. Otherwise, failures must be assumed to persist for the life of the facility. In particular, for IROFS whose availability is to be relied on, the time interval between surveillance observations or tests of the item should be stated, since restoration of a safe state cannot occur until the failure is discovered.

Review Procedures

The reviewer should compare the proposed technical specifications with ANSI/ANS 15.1, as it could pertain to radioisotope production structures, systems, and components, with previously accepted technical specifications for facilities of similar design and function, and Appendix 14.1 to Part 1 of NUREG-1537 as modified by this ISG.

The reviewer should be able to determine the technical specifications and bases from the analyses in the SAR and the ISAs for all radioisotope production processes. The reviewer

should determine that the technical specifications are complete and in the proper format and that each specification is supported by appropriate analyses in the SAR or the ISAs.

The review of the identification of IROFS that are technical specifications should include an acceptance review and evaluation of the ISA Summary that includes an ISA methods review, horizontal review, and vertical slice review.

Evaluation Findings

The current version of this section in NUREG-1537 applies to the radioisotope production facility with the following substitution for the last sentence in the bullet paragraph:

> The staff has reviewed the format and contents of the technical specifications using the guidance of ANSI/ANS 15.1-2007 as it would apply to radioisotope production operations, and NUREG-1537, Part 1, issued February 1996, as modified in this ISG for application to radioisotope production.

15. FINANCIAL QUALIFICATIONS

The current wording of this chapter in NUREG-1537, Part 2, is broad enough to apply to any type of reactor facility. It may also be applicable to a radioisotope production facility provided that a reference to a non-power reactor can be interpreted to mean a reactor or a production facility, as appropriate. Since the NUREG was initially issued in 1996, some changes to the regulations and guidance have occurred that are reflected in this ISG as described below.

Introductory Section

- In paragraph 2, the reference to 10 CFR 2.790 should be replaced with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," for requests for withholding.

- The first sentence of paragraph 3 should be changed as follows:

  Because the review of this information requires specialized knowledge that the non-power reactor technical reviewer usually does not possess, this financial information is submitted to expert financial reviewers (located in the Financial Analysis and International Projects Branch, which is in the Division of Policy and Rulemaking in the Office of Nuclear Reactor Regulation at the time of this writing).

15.1 Financial Ability To Construct a Non-power Reactor

Acceptance Criteria

At the end of the first bullet, add the following:

> The estimate of construction costs should provide, at a minimum, a breakdown of costs into major cost elements, such as material and labor, and include a detailed schedule of construction activity in today's dollars.

Review Procedures

Change the branch name from the "License Renewal and Environmental Review Project Directorate" to the "Financial Analysis and International Projects Branch."

15.2 Financial Ability To Operate a Non-power Reactor

Review Procedures

Change the branch name from the "License Renewal and Environmental Review Project Directorate" to the "Financial Analysis and International Projects Branch."

Evaluation Findings

Replace the text in the bullet with the following:

The applicant has supplied financial information for the estimates of operating costs and the sources of funds to cover these costs. The NRC staff has reviewed the financial ability of the applicant to operate the proposed facility and concludes that the applicant has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs. Therefore, the NRC staff concludes that the applicant has met the financial qualifications requirements under 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities regarding the facility. The NRC staff has also reviewed the proposed conduct of commercial activities at the facility and finds that it meets the requirements of [a Class 104 license pursuant to 10 CFR 50.21] or [a Class 103 license pursuant to 10 CFR 50.22.]

15.3 Financial Ability To Decommission the Facility

Areas of Review

Replace the text of the third bullet with the following:

The means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

Acceptance Criteria

Replace the text of the fourth bullet with the following:

The applicant should provide a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility based on actual changes or changes in cost indices.

Review Procedures

Change the branch name from “License Renewal and Environmental Review Project Directorate” to the “Financial Analysis and International Projects Branch.”

Evaluation Findings

Replace the text of the bullet with the following:

The applicant has supplied financial information for decommissioning costs of the facility in accordance with 10 CFR 50.75(d). The NRC staff has reviewed the decommissioning cost estimate submitted by the applicant and concludes that the cost estimate is reasonable. The applicant has indicated the method or methods to be used to provide funds for decommissioning and has provided a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The NRC has reviewed the applicant’s information provided on decommissioning funding assurance and finds that the applicant’s financial assurance method to be used to provide funds for decommissioning is acceptable, and the applicant’s means of adjusting the cost estimate and associated funding level periodically over the life of the facility

is reasonable. The NRC staff notes that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs resulting from the availability of disposal facilities. The applicant or licensee also has an obligation under 10 CFR 50.9, “Completeness and Accuracy of Information,” to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

16. OTHER LICENSE CONSIDERATIONS

The current wording of NUREG-1537 is applicable to a non-power reactor and radioisotope production facility without the need for modification or augmentation in this ISG.

17. DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

This ISG augments NUREG-1537, Part 2, for applicability to an AHR reactor and processing facility for separating FP radioisotopes from the reactor fuel.

Most of the current content of this introductory section is outdated. For the purpose of this ISG, this section should read as follows:

> The NRC has developed a systematic approach for licensee and NRC actions to terminate facility licenses. As required by 10 CFR 50.82(b), "Termination of License," an application for termination of a non-power reactor license must be preceded or accompanied by a proposed decommissioning plan (DP). The following guidance is offered to facilitate the composition and review of such decommissioning plans.

17.1 Decommissioning

A new section to NUREG-1537 is added as follows:

> 17.1.0 Decommissioning Report
>
> In accordance with the requirements of 10 CFR 50.33(k)(1), an application for an operating license or a combined license for a production or utilization facility must state, in the form of a report as described in 10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," how reasonable assurance will be provided that funds will be available to decommission the facility. This report must include a cost estimate, the proposed decommissioning method to be used, and a proposed means of projecting changes to the cost estimate. Chapter 15 of this NUREG provides additional information on funding.

17.1.1 Preliminary Decommissioning Plan

The wording in this section should be changed as follows:

> In accordance with the requirements of 10 CFR 50.75(f)(4), non-power reactor licensees must submit a preliminary decommissioning plan at or about 2 years before the projected end of operation of the facility. The plan shall include an estimate of the cost and an up-to-date assessment of the major technical factors that could affect planning for decommissioning. The factors to be considered include the following:
>
> [The current list of factors in items (1) through (5) apply to a non-power reactor and radioisotope production facility.]

The last paragraph of this section should read as follows:

> The preliminary DP only needs to address the above-listed five factors and may be substantially less detailed than the final DP. The plan should show that the licensee is aware of the technical and administrative complexities of

decommissioning. The reviewer should compare the licensee's plan with other plans for similar facilities that have been reviewed.

17.1.2 Decommissioning Plan

The wording of the current NUREG remains unchanged except, in the second paragraph, the references to sections of the regulations should be changed from 10 CFR 50.82(b)(1)(ii) to 10 CFR 50.82(b) and from 10 CFR 50.82(b)(1)(iii) to 10 CFR 50.82(b)(4)(i).

17.1.3 Review of the Decommissioning Plan

The current wording of this section applies to a non-power reactor and radioisotope production facility. A new second paragraph should be inserted to update the regulatory developments and expand the pertinent technical reference material as follows:

> Subsequent to the issuance of this NUREG, the majority of decommissioning oversight responsibility has been shifted to the NRC Office of Federal and State Materials and Environmental Management Programs (FSME), Division of Waste Management and Environmental Protection (DWMEP), Materials Decommissioning Branch. More guidance has been published on decommissioning methods, particularly on the subject of conducting surveys and satisfying acceptance criteria. NUREG-1757, Volume 1, "Consolidated Decommissioning Guidance: Decommissioning Process for Material Licensee," Rev. 2, September 2006; NUREG-1757, Volume 2, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," Rev.1, September 2006; and NUREG-1757, Volume 3, "consolidated NMSS Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness," September 2003, contains this recently developed guidance. Therefore, Appendix 17.1, "The Format and Content of DP for NPRs," may serve as the basic outline for the DP but the content should be augmented by including appropriate parts of NUREG-1757.

Areas of Review

The current wording is applicable, but a final bullet should be added as follows:

- Review Appendix 17.1 to ensure that the DP addressed all of the identified contents of a DP.

Review Procedures

The current wording is applicable. Add a sentence as follows:

> The reviewer should ensure that the applicable sections of the DP include the most current guidance (NUREG-1757).

Evaluation Findings

The current wording is applicable.

17.2 Possession-Only License

The current wording is applicable to a non-power reactor and radioisotope production facility provided the references in the second paragraph are changed from 10 CFR 50.82(a) to 10 CFR 50.82(b)(1) and from 10 CFR 50.82(b)(1)(iii) to 10 CFR 50.82(4)(i).

17.2.1 Review of the Application for Possession-Only License Application

The current wording of this section applies to a non-power reactor and radioisotope production facility.

17.2.1.1 Facility License

The current wording of this section applies to a non-power reactor and radioisotope production facility.

17.2.1.2 Technical Specifications

The current wording of this section applies to a non-power reactor and radioisotope production facility.

17.2.1.3 Emergency, Physical Security, and Operator Requalification Plans

The current wording of this section applies to an AHR or radioisotope production facility.

17.2.1.4 Possession-Only License Amendment Safety Analysis

The current wording of this section applies to an AHR or radioisotope production facility.

17.2.1.5 Changes to Facility without License Amendment

The current wording of this section applies to an AHR or radioisotope production facility.

Appendix 17.1

FORMAT AND CONTENT OF DECOMMISSIONING PLAN

For the purpose of this ISG, the content of this appendix has not been reviewed for necessary updates or changes. Regulatory updates and technical guidance have occurred since this NUREG was issued. The reviewer should keep this in mind if any license application reviews are conducted using this guide. In particular, the guidance in NUREG-1757 should be used to supplement the content of applications and reviews.

18. HIGHLY ENRICHED URANIUM TO LOW-ENRICHED URANIUM CONVERSION

The current version of NUREG-1537, Part 2, applies without modification or augmentation provided that the term “reactor” is understood to mean a reactor or a radioisotope production facility.