

## CHAPTER 3 – DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

### **Section 3.2 – Meteorological Damage**

#### **RAI 3.2-1**

*NUREG-1537, Part 1, Section 2.3.1, “General and Local Climate,” states, in part, “[t]he applicant should also estimate the weight of the 100-year return period snowpack and the weight of the 48-hour probable maximum precipitation for the site vicinity, if applicable, as specified by the USGS. Using these estimates for Chapter 3, the applicant should calculate the design loads on the roof of the reactor building, and compare them with local building codes for similar types of structures.”*

*While SHINE PSAR, Section 2.3.1.2.9, “Snowpack and Probably Maximum Precipitation (PMP),” contains an estimate of the snowpack load and probable maximum precipitation, as described in NUREG-1537, and SHINE PSAR, Section 3.2.3, “Snow, Ice, and Rain Loading,” the information developed in SHINE PSAR, Section 2.3.1.2.9, is not used to calculate the design loads.*

*Provide either additional information that explains why SHINE PSAR, Section 3.2.3, does not utilize the data developed under SHINE PSAR, Section 2.3.1.2.9, or calculate the design loads with the data from SHINE PSAR, Section 2.3.1.2.9, accordingly.*

#### **SHINE Response**

Subsection 2.3.1.2.9 of the PSAR provides an estimate for the 100-year return interval snowpack for the SHINE site and the weight of the 48-hour probable maximum precipitation (PMP) for the SHINE site vicinity. Subsection 3.2.3 of the PSAR describes the criteria used to design the SHINE facility to withstand conditions due to snow, ice, and rain loading. Subsection 3.4.2.6.3.4 of the PSAR describes the snow load used in the structural analysis of the SHINE facility, which is more conservative than the 100-year return interval snowpack for the SHINE site provided in Subsection 2.3.1.2.9. The structural analysis of the SHINE facility is provided as Attachment 3.

Snow load inputs for the structural analysis of the SHINE facility are determined in accordance with Chapter 7 of American Society of Civil Engineers (ASCE) Standard 7-05 (Reference 14), and are described in Section 2.1.4.3 of the structural analysis. In accordance with Section 7.10 of ASCE 7-05, a rain-on-snow surcharge load is not considered in the structural analysis because the SHINE facility is located in an area where the ground snow load (determined from Figure 7-1 of ASCE 7-05) is greater than 20 lb/ft<sup>2</sup>.

Subsection 2.3.1.2.9 and Subsection 19.3.2.3.6 of the PSAR contain an administrative error stating the units for the 50 and 100-year interval snowpacks are inches. In accordance with Figure 7-1 of ASCE 7-05, the units for the 50 and 100-year interval snowpacks are lb/ft<sup>2</sup>. An IMR has been initiated to address the issue.

## **Section 3.3 – Water Damage**

### **RAI 3.3-1**

*NUREG-1537, Part 1, Section 3.3, “Water Damage,” states in part, that the applicant should specifically describe “... (2) the impact on systems resulting from instrumentation and control electrical or mechanical malfunction due to water, and (3) the impact on equipment, such as fans, motors, and valves, resulting from degradation of the electromechanical function due to water.”*

*NUREG-1537, Part 2, Section 3.3, “Water Damage,” Acceptance Criteria, states:*

- The design criteria and designs should provide reasonable assurance that structures, systems, and components would continue to perform required safety functions under water damage conditions.*
- For the design the applicant should use local building codes, as applicable, to help ensure that water damage to structures, systems, and components at the facility site would not cause unsafe reactor operation, would not prevent safe reactor shutdown, and would not cause or allow uncontrolled release of radioactive material.*

*While SHINE PSAR, Section 3.3, “Water Damage,” discusses water damage and PSAR Section 3.3.1.1.2, “Compartment Flooding from Fire Protection Discharge,” deals with flooding due to malfunction of the Fire Protection System, there is no discussion of the effects of discharge of the Fire Protection System on structures, systems, and components (SSCs).*

*Provide additional information discussing the effects of discharge of the Fire Protection System on SSCs.*

### **SHINE Response**

The safety-related function(s) of structures, systems, and components (SSCs) that are subject to the effects of a discharge of the fire suppression system will be appropriately protected by redundancy, separation, and a fail-safe design of each SSC. If deemed necessary by SHINE for the purposes of asset protection, electrical equipment may be protected from unacceptable damage if wetted by fire sprinkler system discharge by sprinkler water shields or hoods, in accordance with National Fire Protection Association (NFPA) 13 (Reference 15).

The SHINE facility design will ensure that fire suppression system discharge in one fire area does not impact safety-related SSCs in adjacent fire areas.

## **Section 3.4 – Seismic Damage**

### **RAI 3.4-1**

*NUREG-1537, Part 1, Section 3.4, “Seismic Damage,” states that the applicant should include information on the facility seismic design to provide reasonable assurance that the reactor could be shut down and maintained in a safe condition or that the consequences of accidents would be within the acceptable limits in the event of potential seismic events. To verify that seismic design functions are met, the applicant should give the bases for the technical specifications.*

NUREG-1537, Part 2, Section 3.4, "Seismic Damage," states that the reviewer should find sufficient information to conclude that the design to protect against seismic damage provides reasonable assurance that the facility structures, systems, and components will perform the necessary safety functions described and analyzed.

SHINE PSAR, Section 3.4.2.2, "Soil-Structure Interaction Analysis," reports Soil-Structure Interactions are performed separately for mean, upper bound, and lower bound soil properties to represent potential variations of the in situ and backfill soil conditions surrounding the building. The Soil-Structure Interaction model is developed using the computer program Structural Analysis Software System Interface (SASSI).

- a) Provide the reference manual and revision used for SASSI.
- b) Provide additional information explaining whether the geotechnical investigations requested above also determined the dynamic soil properties used for the Soil-Structure Interaction analyses. Note: the soil dynamic properties necessary for Soil-Structure Interaction analyses are nonlinear.
- c) Provide the report with details and results for the Soil-Structure Interaction analyses.

#### **SHINE Response**

- a) The soil-structure interaction analysis for the SHINE facility was performed using the SASSI2010 software, Version 1.0. The SASSI2010, Version 1.0 User's Manual contains proprietary information and is available for NRC on-site review at Sargent and Lundy's offices.
- b) For the soil-structure interaction analysis, equivalent linear elastic material properties were used because the frequency domain analysis method in SASSI requires the use of elastic material properties. The following approach was used to determine the equivalent linear material properties for the dynamic analyses:
  - 1. Shear wave and compression wave test results from Table A-1 of the Preliminary Geotechnical Engineering Report (provided as Attachment 26 to Enclosure 1 of Reference (7)) were obtained and considered as the best estimate (mean) soil properties ( $G_{BE}$ ). The soil test results provided mechanical properties at low shear strain level.
  - 2. Free field site response analyses were conducted for input seismic motions using the SHAKE2000 computer program to determine the shear strain compatible shear modulus and damping values for the design basis seismic input motion. SHAKE2000 uses an iterative nonlinear procedure to determine strain compatible soil properties from the geotechnical investigation results.
  - 3. The strain compatible soil properties obtained from the above procedure were input to SASSI and the seismic analyses using the frequency domain method were conducted to determine seismic responses of the structure and to develop in-structure response spectra.

Only the best estimate soil case is reported for the seismic analysis in Section 3.4 of the PSAR. The variation of the soil properties will be considered in the final seismic analysis. Upper bound and lower bound soil properties ( $G_{UB}$  and  $G_{LB}$ ) will be determined in accordance with Section 3.7.2 of NUREG-0800 (Reference 16), using the following equations:

$$G_{LB} = G_{BE} / (1+COV)$$

$$G_{UB} = G_{BE} \times (1+COV)$$

The coefficient of variation (COV) will be determined in accordance with Section 3.7.2 of NUREG-0800, and the results of the final seismic analysis will be provided in the FSAR. An IMR has been initiated to track the inclusion of the final seismic analysis results into the FSAR.

- c) Calculation 2013-02413, "Soil-Structure Interaction Analysis of Shine Medical Isotope Production Facility for Design Seismic Event," Revision 0, is provided as Attachment 2.

### **RAI 3.4-2**

*NUREG-1537, Part 1, Section 3.4, "Seismic Damages," states that the applicant should specify and describe the structures, systems, and components that are required to maintain the necessary safety function if a seismic event should occur. The facility seismic design should provide reasonable assurance that the reactor could be shut down and maintained in a safe condition or that the consequences of accidents would be within the acceptable limits.*

*NUREG-1537, Part 2, Section 3.4, states that the "review should include the designs and design bases of structures, systems, and components that are required to maintain function in case of a seismic event at the facility site." The finding required is that the facility design should provide reasonable assurance that the reactor can be shut down and maintained in a safe condition.*

*Additional information is needed in SHINE PSAR, Section 3.4.2.6.1, "Description of the Structures," for the NRC staff to determine the adequacy of structures, systems, and components required to maintain necessary safety functions should a seismic event occur.*

*Provide a comprehensive description of the SHINE facility structures.*

### **SHINE Response**

The SHINE production facility building consists of an Irradiation Facility (IF), containing the irradiation units (IUs), and a Radioisotope Production Facility (RPF), which contains the supercells and tank farm. These two areas make up the Radiologically Controlled Area (RCA) of the SHINE facility. As shown in Figure 1.3-2 of the PSAR, to the west of the RCA are the control room, battery rooms, uninterruptible power supply rooms, and other miscellaneous support rooms. The RCA and areas to the west of the RCA are part of the seismic boundary and are classified as Seismic Category I. These areas contain the safety-related SSCs.

The SHINE production facility building is a box type shear wall system of reinforced concrete with reinforced concrete floor slabs. The SHINE facility main floor is also populated with below grade reinforced concrete cells. The roof of the facility is supported by a steel truss system. There are large concrete containment doors at the north and south ends of the IF and also large stepped concrete doors to access each IU cell. Concrete walls and slabs in the SHINE facility are designed for axial, flexural, and shear loads per the provisions of American Concrete Institute (ACI) 349-06 (Reference 17).

To the north and south of the seismic boundary are the shipping and receiving areas, as well as other areas that contain nonsafety-related support systems and equipment. This part of the structure is not Seismic Category I. The areas outside the seismic boundary do not contain safety-related SSCs.

During a seismic event, the forces will be transmitted through the structural reinforced concrete shear walls to the foundation mat and, ultimately, the soil. Sub-grade walls of the SHINE facility are designed to resist the dynamic soil pressure loads that may occur during a seismic event, as described in Subsection 3.4.2.6.4.7 of the PSAR. Thick concrete walls and roofs encase each IU in a concrete vault, which is seismically designed. Additionally, IUs are housed within the IF and are separated from the RPF within the seismic boundary as shown in Figure 1.3-2 of the PSAR.

*(Applies to RAIs 3.4-3 through 4)*

*NUREG-1537, Parts 1 and 2, Section 3.4, "Seismic Damage," note that acceptable seismic performance has been established in the American National Standard Institute/American Nuclear Society (ANSI/ANS)-15.7, "Research Reactor Site Evaluation." With regard to seismic design, Section 3.2(2) of ANSI/ANS 15.7 states, "[r]eactor safety related structures and systems shall be seismically designed such that any seismic event cannot cause an accident which will lead to dose commitments in excess of those specified in 3.1."*

### **RAI 3.4-3**

*SHINE PSAR, Section 3.4.2.6.5, "Structural Analysis Model," reports that a three-dimensional finite element Structural Analysis Model of the SHINE Facility structure was created using the SAP2000 computer program.*

*Provide the reference manual and revision for the SAP2000 computer program that was used.*

### **SHINE Response**

Version 14.1 of the SAP2000 computer program was used to develop the three-dimensional finite element model of the SHINE facility structure. The SAP2000 reference manual contains proprietary information and is available for NRC on-site review at Sargent and Lundy's offices.

#### **RAI 3.4-4**

*SHINE PSAR, Section 3.4.2.6.6, "Structural Analysis Results," reports Structural Analysis Results were obtained from the SAP2000 model.*

*Provide the report with details and results for the SAP2000 finite element structural analyses.*

#### **SHINE Response**

Calculation 2013-01989, "Conceptual Design of Hardened SHINE Facility Structural Elements," Revision 0, which contains the details and results of the SAP2000 finite element analysis, is provided as Attachment 3.

#### **RAI 3.4-5**

*NUREG-1537, Part 1, Section 3.5, "Systems and Components," states that the applicant should provide the design bases for the systems and components required to function for safe reactor operation and shutdown. This should include, at a minimum, the protective and safety systems; the electromechanical systems and components associated with emergency cooling systems, reactor room ventilation, confinement systems; and other systems that may be required to prevent uncontrolled release of radioactive material. The design criteria should include the conditions that are important for the reliable operation of the systems and components (e.g., dynamic and static loads, number of cyclic loads, vibration, wear, friction, and strength of materials).*

*NUREG-1537, Part 2, Section 3.5, "Systems and Components," states that the reviewer should conclude there is sufficient information to support the design bases of the electromechanical systems and components to give reasonable assurance that the facility systems and components will function as designed to ensure safe operation and safe shutdown of the facility.*

*SHINE PSAR, Section 3.4.3, "Seismic Qualification of Subsystems and Equipment," states that seismic qualification of subsystems and equipment were completed using five methods.*

*Provide the details and results for seismically qualifying the SHINE facility subsystems and components. Include an applicable explanation of whether and how the nodal accelerations (at the locations indicated in PSAR Figures 3.4-4 through 3.4-14, ADAMS Accession No. ML13172A291) are used for the dynamic analyses of equipment.*

#### **SHINE Response**

##### **Seismic Qualification of Piping Subsystems**

SHINE facility piping subsystems that are classified as safety-related will be analyzed using the dynamic analysis method identified in Subsection 3.4.3 of the PSAR, and details of the analysis will be provided in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

The dynamic analysis of safety-related piping subsystems is accomplished using the response spectrum or time history approach. Development of in-structure response spectra is discussed in Subsection 3.4.1 of the PSAR.

## Seismic Qualification of Components

Seismic qualification of components in the SHINE facility will be in accordance with the one or more of the methods described below. SHINE will update Subsection 3.4.3 in the FSAR to include a discussion of which seismic qualification method is used for which type of component. An IMR has been initiated to track the inclusion of this information in the FSAR.

- Qualification by Analytical Methods Only

Analytical calculations only may be used as a qualification method when maintaining the structural integrity is an assurance for the safety function. This method can be used for structurally simple equipment when expected response to the earthquake excitations can be characterized as linear or simple non-linear behavior.

The methods to be used for qualification, by calculations only or by calculations based on supporting test results, are stated below. These methods will depend on the type of equipment and supporting structure. The following defines some of the possible cases and associated analytical methods which may be used in each case:

- Static Analysis

The equipment, as well as its support, can be considered rigid, and may be analyzed by static analysis, if it can be shown that its fundamental natural frequency does not fall in the frequency range below the high frequency asymptote (zero period acceleration (ZPA)) of the required response spectrum (RRS).

For rigid equipment supported by a rigid structure, the equipment motion shall be the same as the floor motion without amplification. The horizontal and vertical dynamic accelerations shall be taken as the zero period acceleration from the applicable response spectrum. These acceleration values are used to perform a static analysis. In this case, the dynamic forces are determined by multiplying the mass of the subassembly or parts of the equipment by the ZPA of the RRS. These forces should be applied through the center of gravity of the subassembly or the part of the equipment. The stresses resulting from each force (in each of the three directions) should be combined by an appropriate combination method to yield the dynamic stresses. The dynamic deflections (deflections due to dynamic loads) may be calculated in the same manner. These dynamic stresses and deflections are added to all stresses and deflections resulting from all applicable loads, to obtain the final resultant stresses and deflections.

- Simplified Dynamic Analysis

A simplified dynamic analysis may be performed in cases where the equipment and support systems' natural frequency falls in the frequency range below the high frequency asymptote (ZPA) of the applicable RRS. This is similar to the static analysis described above, but requires using different values for the accelerations. The accelerations to be used are obtained from the appropriate response spectra curves at each natural frequency in the frequency range of interest. If the frequency information is not available, the simplified dynamic analysis (sometimes referred as the equivalent static

analysis) is performed using 1.5 times the maximum peak of the applicable floor response spectra. Once the dynamic forces are determined using the 1.5 times the peak acceleration values from the RRS, stresses and deformations may be computed following the same procedures used for static analysis.

- Detailed Dynamic Analysis

When acceptable justification for static or simplified dynamic analysis cannot be provided, a detailed dynamic analysis is performed. A mathematical model may be constructed to represent the dynamic behavior of the equipment. A finite element model may be constructed and analyzed using the response spectrum modal analysis or time-history analysis. The maximum inertia forces, at each mass point, from each mode, are applied at that point to calculate the modal reactions (forces and moments) and modal deformations (translations and rotations). The various modal contributions are combined by an appropriate combination method. Closely spaced modes are combined by using an approach from Regulatory Guide 1.92 (Reference 18). The stresses and deflections resulting from each of the three directions are combined to obtain the dynamic stresses and deflections. These dynamic stresses and deflections are added to all stresses and deflections resulting from all applicable loads.

- Qualification by Tests

Seismic qualification by testing is the preferred method of qualification for complex equipment not suitable for analysis, and for equipment required to perform an active function. Seismic qualification by testing shall be performed in accordance with the requirements of Regulatory Guide 1.100 (Reference 19). The vibration inputs for the seismic tests are the response spectra or required input motion (typical for line-mounted equipment) at the mounting location of the equipment.

Multi-axis, single-axis, or static pull tests may be performed within the requirements and guidance provided in the standards mentioned above. When single-axis testing is used, cross-coupling shall be addressed by monitoring responses in directions other than the excitation axis, by performing supporting tests, or by other acceptable means.

The test samples shall be mounted to simulate the recommended service mounting. If this cannot be done, the effect of the actual supporting structure shall be considered in determination of the input motion. The Project Specification will state the expected (or calculated) piping nozzle reaction loads on the equipment which shall be used in the qualification. Any other loads that may act on the component (mechanical, electrical, or instrument) during the postulated dynamic event must be simulated during the test, unless the supporting test (or calculations) show that they are insignificant.

At the completion of the tests, inspection shall be made by the test conductor to assure that no structural damage has occurred. Sufficient monitoring devices shall be used to evaluate the performance of the tested active components during the tests. For acceptability, the components shall demonstrate their ability to perform their intended safety functions when subjected to all applicable loads. A test report, which includes all test data, results and conclusions, shall be submitted.



- Combined Methods of Qualification

Based on the available information, component complexity, and functional requirements, the above mentioned analytical and test methods may be combined in various sequence and content to achieve seismic qualification of the subject components.

#### Nodal Accelerations

The nodal accelerations (at the locations indicated in Figures 3.4-4 through 3.4-14 of the PSAR) are provided in Table 3.4-3 of the PSAR. These are not used for piping subsystem and component analysis, since building time histories and spectra are available. As applicable, these spectra or acceleration time histories will be used to determine the seismic requirements at the component mounting locations for qualification purposes and for piping subsystem dynamic analysis.

#### **RAI 3.4-7**

*NUREG-1537, Part 2, Section 3.4, "Seismic Damage," states that "[t]he applicant should demonstrate that all potential consequences from a seismic event are within the acceptable limits considered or bounded in the accident analyses of Chapter 13 to ensure that conditions due to a seismic event will not pose a significant risk to the health and safety of the public."*

*The SHINE site location is near the Southern Wisconsin Regional Airport. SHINE PSAR, Section 3.4.5.1, "Aircraft Impact Analysis," outlines the methodology for conducting and evaluating small aircraft impact analyses in support of the seismic envelope design for external hazards. The potential locations for 25 aircraft impact analyses of the SHINE facility are listed. In PSAR Table 3.4-4, "Aircraft Impact Analysis Results," the aircraft impact analyses results show that the performance of all barriers are acceptable to prevent transport of radioactive materials to unrestricted areas. However, the engineering report that describes the analyses' details states that all of the results are not referenced.*

*Provide the engineering report that describes the aircraft impact analyses' details that reports the results. Additionally, provide a summary of the results.*

#### **SHINE Response**

Calculation 2013-01911, "Evaluation for Aircraft Impact," Revision 0, which contains the details and results of the aircraft impact analyses, is provided as Attachment 4.

Calculation 2013-01911 evaluates the initial design of the SHINE Medical Isotope Production Facility at Janesville, Wisconsin for effects of an accidental aircraft crash by aircraft operating through the Southern Wisconsin Regional Airport (SWRA). As described in Section 4.1 of the calculation, the Challenger 605 is selected as the design basis aircraft for performing the evaluations. Figures 4.4.1-1 (A) and 4.4.1-1 (B) of the calculation show the walls and roof slabs evaluated for the SHINE facility. Roof panels and walls are two-foot thick reinforced concrete with a design strength of 5000 psi. Roof slabs are supported on trusses spanning in the east-west direction. These trusses are supported on external walls and either on an internal wall running in the north-south direction or on a plate girder spanning in the north-south direction.

Evaluations are made for local damage in Section 5.2 of the calculation and overall damage in Section 5.3 of the calculation. Local damage evaluation results show that the two-foot thick reinforced concrete panels do not scab under the impact from the engine of the impacting aircraft. Calculation results for perforation margin show that the condition of ACI 349-06 (Reference 17), Paragraph F.7.2.3 is satisfied. Therefore, a punching shear evaluation in the overall response assessment is not necessary.

The overall damage evaluation is performed in Section 5.3 of the calculation, considering 25 cases of impact. The impact cases are described in Section 4.4.4 of the calculation. Table 6-1 of the calculation and Table 3.4-4 of the PSAR summarize the acceptability of the 25 impact cases, provided stated conditions for reinforcement size and spacing, including shear ties, and provisions for truss members, including a non-linear analysis in the future to show acceptability of inelastic deformation, are met.

### **Section 3.5 – Systems and Components**

#### **RAI 3.5-3**

*SHINE PSAR, Table 3.5-1, "System and Classifications," states that Radiologically Controlled Area Ventilation Zone 1 is safety-related, QL-1, and Seismic Category I; Radiologically Controlled Area Ventilation Zone 2 is IROF, QL-2, and Seismic Category I; and Radiologically Controlled Area Ventilation Zone 3 is nonsafety-related, QL-3, and Seismic Category III. SHINE PSAR, Section 9a.2.1.1, "Radiologically Controlled Area Ventilation System" (ADAMS Accession No. ML13172A271), does not state that one normally goes through Radiologically Controlled Area Ventilation Zone 3 to get to Radiologically Controlled Area Ventilation Zones 1 or 2, but such a pathway can be inferred from PSAR Section and Figure 1.3-2. Thus, Radiologically Controlled Area Ventilation Zone 3 would be used for access and egress after a postulated event with a loss of offsite power or a design basis earthquake with a loss of offsite power.*

*Provide the basis for designating the Radiologically Controlled Area Ventilation Zone 3 nonsafety-related, QL-3, and Seismic Category III or provide a discussion of the alternate method of access/egress of the Radiologically Controlled Area Ventilation Zones 1 and 2, without causing outside contamination.*

#### **SHINE Response**

RCA Ventilation Zone 3 (RVZ3) is designated as nonsafety-related and QL-2 because it does not meet the SHINE definition of a safety-related SSC (see Part a of the SHINE response to RAI 3.5-1 (Reference 4)), as follows:

1. RVZ3 is not relied upon to maintain the integrity of the primary system boundary (PSB).
2. RVZ3 is not relied upon to shut down the target solution vessel (TSV) or maintain the target solution in a safe shutdown condition.
3. RVZ3 is not relied upon to prevent or mitigate the consequences of accidents which could result in potential exposures comparable to the applicable guideline exposures set forth in 10 CFR 20. This function, as it relates to ventilation, is accomplished by RCA Ventilation Zone 1 (RVZ1) and RCA Ventilation Zone 2 (RVZ2), the ventilation systems used in areas containing radiological material.
4. RVZ3 is not relied upon to assure that an inadvertent criticality accident is not credible.

5. RVZ3 is not relied upon to assure that acute chemical exposures to an individual from licensed material or hazardous chemicals produced from licensed material could not lead to irreversible or other serious, long-lasting health effects to a worker or cause mild transient health effects to any individual located outside the owner controlled area. This function, as it relates to ventilation, is accomplished by RVZ1 and RVZ2, the ventilation systems used in areas containing licensed material.
6. RVZ3 is not relied upon to ensure that an intake of 30 mg or greater of uranium in soluble form by any individual located outside the owner controlled area does not occur. This function, as it relates to ventilation, is accomplished by RVZ1 and RVZ2, the ventilation systems used in areas containing uranium in soluble form.

RVZ3 is classified as Seismic Category III because it is a nonsafety-related system, and does not adversely impact SSCs important to safety that are designed to withstand the effects of a design basis earthquake and remain functional.

Contamination would not be expected with a loss of off-site power because there are no external events that result in both a loss of off-site power and a radiological release. However, if the RVZ3 fans are not operating during a radiological event, then emergency response in accordance with the SHINE Preliminary Emergency Plan (Reference 20) will evaluate and control contamination.

*(Applies to RAIs 3.5-4 through 5)*

*As required by 10 CFR 50.34(a)(4), “[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.”*

### **RAI 3.5-5**

*SHINE PSAR, Section 3.5.2, “Seismic Classification,” discusses the use of Seismic Category II SSCs over Seismic Category I SSCs (Seismic II/I). SHINE PSAR, Table 3.5a-1, “Appendix A to 10 CFR 50 General Design Criteria Which Have Been Interpreted As They Apply to the SHINE Irradiation Facility” (pages 3-88 – 3-93), discusses how the facility complies with 10 CFR Part 50, Appendix A, General Design Criteria. Based on SHINE’s proposed implementation of the of the General Design Criteria, the NRC staff needs clarification on the following considerations with respect to Seismic Categories II/I:*

*General Design Criterion 1 provides that structures, systems, and components important to safety are to be designed, fabricated, erected, and tested to quality standards. Thus, General Design Criterion 1 applies to Seismic II/I since the Seismic II structures, systems, and components should be properly designed, fabricated, and installed to reduce the likelihood of a Seismic Category II structure, system, or component coming loose and falling on and damaging a Seismic Category I structure, system, or component.*

*General Design Criterion 2 provides that structures, systems, and components important to safety are to be designed to resist the effects of natural phenomena like earthquakes. General Design Criterion 2 applies to Seismic III/I because it specifies the natural phenomenon (i.e., earthquake) that must be considered in the design of these structures, systems, and components. If not considered, an earthquake could loosen a Seismic Category II structure, system, or component to the extent that it could cause an unsafe condition (i.e., fall on and damage a Seismic Category I structure, system, or component).*

*General Design Criterion 4 provides that structures, systems, and components important to safety are to be protected against the effects of internally-generated missiles. General Design Criterion 4 applies to Seismic Category II structures, systems, and components because it specifies protection against the effects of internally-generated missiles (i.e., fall on and damage of a Seismic Category I structure, system, or component).*

*Based on the considerations above, dropped loads could cause the potential release of radioactive materials, a criticality accident, or damage to essential safety equipment, which could cause unacceptable radiation exposures.*

*Provide details of the Seismic III/I Program that will be put into place, including the Seismic Category II structural integrity criteria and the Seismic Category II support criteria.*

### **SHINE Response**

As stated in Subsection 3.5.2 of the PSAR, structures, components, equipment, and systems designated as safety-related are classified as Seismic Category I. The criteria put in place for Seismic Category II SSCs to ensure that they do not interfere with Seismic Category I SSCs are described below.

#### **Criteria for Seismic Category II Structures**

Nonsafety-related structures are designed with the following two general criteria:

1. SSCs in nonsafety-related structures attached to a safety-related structure or separated from a safety-related structure by a distance less than or equal to the height of the nonsafety-related structure are designed for abnormal and extreme environmental loads if the failure of the nonsafety-related SSC could result in damage to safety-related SSCs. However, if the failure of the nonsafety-related SSC will not result in damage to safety-related SSCs, the nonsafety-related SSC need not be designed for abnormal and extreme environmental load combinations.
2. Nonsafety-related SSCs separated from a safety-related structure by a distance greater than the height of the nonsafety-related structure need not be designed for abnormal and extreme environmental load combinations.

#### **Criteria for Nonsafety-Related SSCs in Safety-Related Structures**

Nonsafety-related SSCs within safety-related structures (i.e., within the seismic boundary) are evaluated for II/I interaction. Seismic interaction effects to be considered include proximity, differential displacements, structural failure and falling, water spray or flood, and fire.

Depending on the type of component, nonsafety-related SSCs categorized as II/I are evaluated or qualified using the following methods to demonstrate that they do not impact safety-related SSCs:

a. Detailed Evaluation

Nonsafety-related piping, raceways, heating, ventilation, and air conditioning (HVAC) ducts, and their supports are designed for design basis earthquake (DBEQ) loading against increased allowable stresses, using methodologies and codes similar to those required for the corresponding safety-related SSCs. The intent is to demonstrate that gross structural collapse or failure of these components does not prevent safety-related SSCs from performing their safety functions. This is accomplished by detailed structural analysis or consideration of seismic loading in the development of allowable spans and support configurations.

Nonsafety-related SSCs such as electrical equipment cabinets and mechanical equipment (pumps, motors, and tanks) may require only anchorage design and analysis to assure that the components do not interact with adjacent safety-related SSCs or become missiles during a seismic event.

b. Qualitative Evaluation

Certain architectural items such as drop ceilings, lighting, and communications or security equipment may be evaluated for II/I by engineering judgment using considerations such as weight to assess their potential impact on safety-related components (e.g., a lightweight object impacting a large diameter pipe, or a small bore pipe impacting the rugged casing of a pump).

Layout guidelines and building zones are established to indicate that certain rooms or areas within Category I structures do not contain safety-related components. In such cases, the only evaluations required would be to ensure that failure of the nonsafety-related component would not adversely impact the Category I structure by missile impact, flooding or fire.

c. Specification

Specifications for commodity or architectural items can include requirements for performance during a seismic event (e.g., cabinet door swing limits and lighting supports designed for seismic loads).

CHAPTER 4 – IRRADIATION UNIT AND RADIOISOTOPE PRODUCTION FACILITY  
DESCRIPTION

**Section 4a2.2 – Subcritical Assembly**

**RAI 4a2.2-5**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.2.1, “Reactor Fuel,” Acceptance Criteria, states, in part, that the PSAR should include a description of the “various phenomena that result in changes to the initial fuel composition...[including] potential fuel and fission product precipitation...”*

*SHINE PSAR, Section 4a2.2.1.6, “TSV Operating Conditions,” states that there is no precipitation out of the target solution, however IAEA TECDOC-1601, “Homogeneous Aqueous Solution Nuclear Reactors for the Production of Mo-99 [Molybdenum-99] and Other Short Lived Radioisotopes,” states that as the fuel solution ages, fission products can approach solubility limits.*

*Provide information on how close the SHINE target solution will be to the solubility limits. Additionally, provide additional information discussing whether SHINE plans to use catalytic agents to mitigate precipitation, as discussed in PSAR, Section 4a2.4.1.1.*

**SHINE Response**

Scientific literature was reviewed and studies have been performed at Argonne National Laboratory (ANL) to address potential precipitating species. The results of this show that some fission product compounds will reach solubility limits, with up to [ Proprietary Information ] of potential fission product precipitates per kg of target solution based on thermodynamic modeling (without kinetics limitations) and [ Proprietary Information ] cycles of irradiation (Attachment 5). SHINE is not planning on using catalytic agents to mitigate fission product precipitation (but will use catalytic agents to prevent uranium precipitation, as described below) because the small amount of potential precipitation is expected to have an insignificant effect on reactivity in the TSV. The details of the system design to remove the precipitates from the target solution during processing (e.g., filter with differential pressure monitoring) will be provided in the FSAR.

Based on studies at ANL, provided as Attachment 6, uranyl peroxide formation is expected to be avoided by [ Proprietary Information ]. SHINE plans to use an [ Proprietary Information ] in the target solution, as specified in Table 4a2.2-1 of the PSAR ([ Proprietary Information ]). As discussed in Part 1 of the SHINE Response to RAI G-1 (Reference 4), additional testing is being performed at ANL to validate that kinetics effects of uranyl peroxide precipitation are fully understood, which is scheduled to be completed by Q3 2015.

SHINE will update Table 4a2.2-1 of the PSAR with potential fission product precipitate levels and provide details on removing potential precipitates from the target solution in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

#### **RAI 4a2.2-8**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.2.1, "Reactor Fuel," Acceptance Criteria, states, in part, that "[m]aintaining fuel barrier integrity should be the most important design objective."*

*SHINE PSAR, Section 4a2.2.1.10, "TSV Physical Structure," mentions a "credible deflagration." A strong deflagration or detonation could compromise the integrity of the primary system boundary.*

*Provide the pressure expected during a "credible deflagration," and discuss how this value was determined, as well as how it compares to the maximum pressure that each component of the primary system boundary can withstand.*

#### **SHINE Response**

Preliminary calculations indicate that the maximum pressure during a credible deflagration is less than 50 psig.

The maximum pressure during a credible deflagration was determined by investigating possible modes of failure of the TSV Off-Gas System (TOGS). SHINE determined that a complete blockage of a TOGS line where the TOGS was no longer available to control hydrogen concentrations in the TSV or provide flow for dilution was the limiting case. Following the blockage, hydrogen concentrations rise due to a short delay prior to automatic trip of the TSV, radiolytic gases contained within the void fraction of the solution evolving out, and decay hydrogen production. The trip of the TSV sends a signal to open the dump valves, but one dump valve is assumed to fail to open, which reduces the transfer rate of the solution to the dump tank. The draining of the target solution to the dump tank moves the source of the decay hydrogen generation, as well as dilutes the TSV headspace gas due to the displacement of gas between the two vessels as the dumping occurs. The calculated peak hydrogen concentration is below the detonation limit. The deflagration pressure was calculated using an adiabatic flame temperature approach, adjusted for constant volume conditions, at the point of peak hydrogen concentration. Ideal gas behavior was assumed.

SHINE will ensure the design pressure of each component of the PSB will be greater than the credible deflagration pressure determined in the final calculations, which will be performed during detailed design.

#### **RAI 4a2.2-9**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.2.1, "Reactor Fuel," Acceptance Criteria, states, in part, that the application should provide a summary of the "fuel development, qualification, and production program." This should include discussions on fuel characterization, provide information on radiolytic gas production, changes in pH, gas removal, and addition of fuel and acid to the vessel along with implications on reactivity.*

*SHINE PSAR, Section 4a2.2.1.13, "Target Solution History," briefly describes some of the history of uranyl sulfate development, but does not describe SHINE's Target Solution Qualification Program.*

*Provide a description of SHINE's Target Solution Qualification Program, including specific historical target solution data and their origin (references) that have been used for validation and safety calculations presented in the current SHINE PSAR. Include tests, experiments, and analyses that will be (or have been) performed to validate the historical data.*

### **SHINE Response**

The SHINE Target Solution Qualification Program, provided as Attachment 7, contains historical target solution data, means to produce the target solution, overview of the processes that the target solution is exposed to, limits to ensure safe and reliable target solution performance, and the tests and experiments that will be and have been performed to validate the target solution characteristics.

Part 1 of the SHINE Response to RAI G-1 (Reference 4) provides a discussion of the additional testing being performed at ANL to validate that the kinetic effects of uranyl peroxide formation are fully understood.

### **Section 4a2.3 – Neutron Driver**

#### **RAI 4a2.3-1**

*While the ISG Augmenting NUREG-1537 does not have a section dedicated to the neutron driver assembly system (NDAS), which is unique to SHINE, the PSAR should provide the same level of detail for this system as is expected for other systems and components. This is in alignment with 10 CFR 50.34(a)(4), which requires a "preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."*

*For instance, the PSAR should include information regarding corrosion control, susceptibility to radiation damage, and the physical description, including materials and physical dimensions.*

- 1) Provide the physical characteristics of the NDAS (e.g., construction materials, dimensions).*
- 2) SHINE PSAR, Section 4a2.3, "Neutron Driver," states, that "most materials will not have radiation damage concerns," but does not specify which components will have radiation damage concerns. Describe what radiation damage concerns there are for affected materials and components.*
- 3) Provide the expected activity of the NDAS due to activation of its components at the end of one irradiation cycle and at the end of its expected life.*

### **SHINE Response**

- 1) See Reference 4 for Part 1 of the SHINE Response to RAI 4a2.3-1.
- 2) See Reference 4 for Part 2 of the SHINE Response to RAI 4a2.3-1.



- 3) At the end of one irradiation down cycle, the expected activity of the Neutron Driver Assembly System (NDAS) components will be [ Proprietary Information ]. Over 90 percent of this expected activity is located in components below the pool surface (i.e., primarily the target chamber assembly), which will significantly enhance shielding.

The current expected lifetime of an NDAS is [ Proprietary Information ], which corresponds to an expected activity of the NDAS components of [ Proprietary Information ]. Over 90 percent of this expected activity is located below the pool surface.

#### **Section 4a2.4 – Target Solution Vessel and Light Water Pool**

*(Applies to RAIs 4a2.4-1 through 3)*

*As required by 10 CFR 50.34(a)(4), “[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.”*

##### **RAI 4a2.4-1**

*SHINE PSAR, Section 4a2.4.1.1, Design Considerations,” specifies that the construction of, and materials for, the TSV follow the intent of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III (ASME, 2011).*

*Provide a discussion of the applicable ASME Code, how the SHINE design meets the intent of the code, and the features of the SHINE design that prevent application of the code as written.*

##### **SHINE Response**

As described in Subsection 4a2.4.1.1 of the PSAR, the TSV will be designed and fabricated following the intent of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III.

The TSV will not be certified to Section III of the ASME BPVC because its fabrication material, Zircaloy-4, is not included in ASME BPVC Section II, Part D, “Materials Properties,” and Section III of the ASME BPVC does not provide allowances for non-ASME BPVC materials. As described in Part 1 of the SHINE Response to RAI G-1 (Reference 4), irradiation and corrosion testing is currently being performed at Oak Ridge National Laboratory (ORNL) to determine the acceptability of this zirconium alloy for the TSV. The results of the corrosion and irradiation testing will be used as input for final TSV design, including corrosion allowances and material properties following irradiation.

The intent of Section III of the ASME BPVC is to verify that the stress intensity encountered by the TSV under design loadings during the design lifetime does not exceed the allowable stresses of the TSV material. This will be determined using the results of the testing at ORNL.

and applies to external, internal, and nozzle loadings, including postulated accident loadings which are factored into the TSV design parameters. No postulated normal or accident conditions result in stress intensities that exceed the allowable stress intensities per the ASME BPVC.

Sections of the ASME BPVC listed above describe fabrication, installation, and pre-service inspection requirements. The requirements will be followed for the TSV.

#### **RAI 4a2.4-2**

*SHINE PSAR, Section 4a2.4.1.5, "Chemical Interactions and Neutron Damage," states that a materials surveillance and inspection program for the TSV and other primary system boundary (PSB) components will be described in the final safety analysis report (FSAR).*

*Provide a list of surveillance and inspection requirements, as well as information to show that the design will allow the required periodic surveillance and inspections to be performed.*

#### **SHINE Response**

The TSV, TSV dump tank, TOGS, and associated components act as the primary pressure boundary and are safety-related SSCs. SHINE plans to provide surveillance and inspection capabilities for these SSCs in order to assess mechanical integrity and verify corrosion rates are acceptable. The following is a list of inspection requirements for the primary system boundary components:

- The TSV will receive periodic visual inspection of representative portions of the vessel's interior surfaces, such as weld locations, the liquid level line, and piping connections. Access for the inspections will be provided by one or more openings in the TSV.
- The TSV dump tank will receive periodic visual inspection of representative portions of the vessel's interior surfaces, such as weld locations, the liquid level line, and piping connections. Access for the inspections will be provided by one or more openings in the TSV dump tank.
- The TOGS will receive periodic internal visual inspection of key areas or components. Access for the inspections will be provided by one or more openings in the TOGS.
- The TOGS components will receive periodic external visual inspections. The TOGS will be designed with sufficient space between components to allow external visual inspection.
- Test coupons will be placed inside the TSV to verify acceptable corrosion rate. Test coupons will be removable through an inspection opening.

#### **Section 4a2.5 – Irradiation Facility Biological Shield**

##### **RAI 4a2.5-1**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.4, "Biological Shield," Acceptance Criteria, states that "[t]he principal objective of the shield design should be to ensure that the projected radiation dose rates and accumulated doses in occupied areas do not exceed the limits of 10 CFR Part 20, 'Standards for Protection Against Radiation,' and the guidelines of the facility's ALARA (as low as reasonably achievable) program discussed in Chapter 11 of the SAR."*

*SHINE PSAR, Section 4a2.5.2.2, “Geometry and Configuration,” states that the side wall of the IU cell biological shield consists of standard density concrete that is 6.0 feet (1.8 meters) thick and that the dose rates on the external surface of the shield wall is expected to be less than 1.0 millirem/hour. PSAR Section 4a2.5.3.1, “Shielding Calculations,” notes that the Monte-Carlo N-Particle (MCNP) Transport Code was used to determine the required shield thickness. PSAR Section 4a2.5.4, “Analysis,” states, in part, that analysis is performed to:*

- *Give detailed results of both neutron and gamma-ray dose rates at locations that could be occupied as well as to the unrestricted environment.*
- *Include shield penetrations and voids, such as beamports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.*

*In order for the NRC staff to determine the adequacy of the shielding design of the IU cell, provide a list of the components inside the irradiation unit cell that are considered significant contributors (and the magnitude of these contributions) to the gamma and neutron flux and dose rates impinging on the interior shield wall. For each component describe the key assumptions included in the MCNP (or other computer code) radiation transport modeling used to determine shield wall thickness.*

### **SHINE Response**

The components inside the IU cell that are considered to be significant contributors to the gamma and neutron fluxes are the neutron driver and the Subcritical Assembly System (SCAS).

The interior surface of the IU shield wall was partitioned into two foot by two foot sections above the light water pool, and the neutron and gamma flux and dose rates in each section were calculated. The doses below the light water pool are not required for shielding purposes as this portion of the IU cell is below grade and there are no areas where personnel would normally be present below grade near the IU cell.

The magnitude of the contributions from the neutron driver during irradiation are:

- Average neutron flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]
- Peak neutron flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]
- Average gamma flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]
- Peak gamma flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]

The magnitude of the contributions from the SCAS during irradiation are:

- Average neutron flux and dose rate impinging on the interior shield wall: Not significant compared to the neutron driver
- Peak neutron flux and dose rate impinging on the interior shield wall: Not significant compared to the neutron driver
- Average gamma flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]
- Peak gamma flux and dose rate impinging on the interior shield wall:  
[ Security-Related Information ], [ Security-Related Information ]

Key assumptions and inputs included in the radiation transport modeling to determine shield wall thickness using MCNP include:

- The neutron driver is producing [ Proprietary Information ] neutrons from fusion per second, which is the upper bound of its expected output, as described in Subsection 4a2.3.7 of the PSAR.
- The photon flux contributions from decay of activated neutron driver components is not included because the flux from this decay is expected to be insignificant in relation to the total doses.
- Rebar in the concrete shield walls is #11 rebar, six inch on center in each direction, and is made of carbon steel.
- The concrete and rebar is homogenized into a single material.

## **Section 4a2.6 – Nuclear Design**

*(Applies to RAIs 4a2.6-3 through 4)*

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.5.1, “Normal Operating Conditions,” states that the PSAR should give reactivity worths for control rods, reflector units, and other in-core components for all anticipated configurations. While some information is presented on coefficients of reactivity in PSAR Section 4a2.6.4, additional information is needed to verify that the SHINE IUs will not become critical under any phase of operation.*

### **RAI 4a2.6-3**

*Compare the reactivity worths of all components in the IU to the margin to criticality in the TSV for all phases of operation.*

### **SHINE Response**

#### **Mode 1 Startup Conditions**

The TSV is designed to be filled with uranyl sulfate solution to a level that is approximately five percent by volume below critical, with an estimated  $k_{\text{eff}}$  of [ Proprietary Information ] at cold shutdown conditions, which results in an expected margin to criticality of approximately [ Proprietary Information ]. While the SHINE TSV is not a reactor, there are two moveable components identified within the nuclear assembly that have the potential for significant reactivity effects. These two components are the tritium chamber and the neutron multiplier.

Removing the neutron multiplier and flooding the tritium chamber with water from the Light Water Pool System (LWPS) were calculated to both have negative reactivity effects, thus increasing the margin to criticality. The worth of the neutron multiplier was calculated to be approximately [ Proprietary Information ] and the worth of the tritium chamber was calculated to be approximately [ Proprietary Information ] at nominal Mode 1 conditions.

#### Mode 2 Irradiation Conditions

As the SHINE system transitions from Mode 1 startup conditions to Mode 2 irradiation conditions, the  $k_{\text{eff}}$  of the system decreases to approximately [ Proprietary Information ]. This is a margin to criticality of approximately [ Proprietary Information ].

Replacing the neutron multiplier and flooding the tritium chamber with water from the LWPS were calculated to both have negative reactivity effects, thus increasing the margin to criticality. The worth of the neutron multiplier was calculated to be approximately [ Proprietary Information ] and the worth of the tritium chamber was calculated to be approximately [ Proprietary Information ] at Mode 2 irradiation conditions.

#### RAI 4a2.6-4

*The SHINE system may have a positive void coefficient for the water in the cooling system since the fuel solution is over-moderated. However, a pipe break or other means of introducing voids, lowering the coolant density in the system, could result in a reactivity insertion. Additional information is needed to determine if voiding out the cooling system could turn the TSV from a subcritical system into a critical reactor.*

*Provide the reactivity worth for voiding out the cooling system over the full range from nominal coolant temperature and density to a fully voided cooling system.*

#### SHINE Response

The SHINE subcritical assembly cooling includes a Primary Closed Loop Cooling System (PCLS) and a LWPS cooling loop. SHINE performed a calculation of the expected reactivity changes due to voiding out the cooling system from nominal coolant temperature and density to a fully voided cooling system for the PCLS and LWPS. Voids were assumed to occur in the respective cooling channels (and [ Proprietary Information ] for the PCLS) of the subcritical assembly. The calculation was performed at cold (Mode 1) startup conditions and hot (Mode 2) irradiation conditions. Results of the calculation are provided in Table 4a2.6-4-1 for the LWPS and Table 4a2.6-4-2 for the PCLS.

**Table 4a2.6-4-1: Reactivity Changes Due to LWPS Voids**

<b>Percent Void</b>	<b>Delta Reactivity (pcm)</b>	
	<b>Reactivity Change for Mode 1 Startup Conditions</b>	<b>Reactivity Change for Mode 2 Irradiation Conditions</b>
20	[ Proprietary Information ]	[ Proprietary Information ]
40	[ Proprietary Information ]	[ Proprietary Information ]
60	[ Proprietary Information ]	[ Proprietary Information ]
80	[ Proprietary Information ]	[ Proprietary Information ]
100	[ Proprietary Information ]	[ Proprietary Information ]

**Table 4a2.6-4-2: Reactivity Changes Due to PCLS Voids**

<b>Percent Void</b>	<b>Delta Reactivity (pcm)</b>	
	<b>Reactivity Change for Mode 1 Startup Conditions</b>	<b>Reactivity Change for Mode 2 Irradiation Conditions</b>
20	[ Proprietary Information ]	[ Proprietary Information ]
40	[ Proprietary Information ]	[ Proprietary Information ]
60	[ Proprietary Information ]	[ Proprietary Information ]
80	[ Proprietary Information ]	[ Proprietary Information ]
100	[ Proprietary Information ]	[ Proprietary Information ]

The results provided in Table 4a2.6-4-1 show that for the LWPS, there is a positive insertion of reactivity as the percent void increases. As shown in the table, a significant amount of void (> 20 percent) must be present to add a substantial amount of reactivity ([ Proprietary Information ]) at startup conditions. The design will prevent introduction of significant amount of void into the subcritical assembly, such as the use of a delay tank to vent entrained voids located downstream of the LWPS cooling pump. Positive pressures are expected in the piping line downstream of the LWPS cooling pump, which will prevent the introduction of void since a break in the line would result in the loss of water rather than the ingress of air. Flow meters and pressure detection on the LWPS are planned, which would also indicate if a significant amount of air had entered the pump since the output pressure and flow will decrease.

The results provided in Table 4a2.6-4-2 show that for the PCLS, there is a negative insertion of reactivity as the percent void increases.

(Applies to RAIs 4a2.6-5 through 6)

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.5.2, “Reactor Core Physics Parameters,” states, in part, “the applicant should present information on core physics parameters that determine reactor operating characteristics....”*

**RAI 4a2.6-5**

*The SHINE PSAR does not discuss the effects of xenon-135 and samarium-149 on the TSV operation irradiation cycle.*

*Provide an estimate of the reactivity due to xenon-135 and samarium-149 over the cycle and its effect on neutron multiplication and fission power, since the time required to establish equilibrium xenon and samarium is significant compared to the length of an irradiation cycle.*

**SHINE Response**

The reactivity estimate due to xenon-135 and samarium-149 over [ Proprietary Information ] cycles is presented in Figure 4a2.6-5-1. Xenon was assumed to not escape from solution, resulting in bounding reactivity worth. As shown in Figure 4a2.6-5-1, the steady-state reactivity worth of xenon was estimated at approximately [ Proprietary Information ]. The reactivity worth from samarium-149 increases over the course of [ Proprietary Information ] cycles, with a worth change per cycle estimated at less than [ Proprietary Information ].

The reactivity change due to both xenon-135 and samarium-149 over the course of the cycle could add an estimated [ Proprietary Information ] of reactivity to the subcritical assembly. This change in reactivity is estimated to reduce operating neutron multiplication and fission power by less than 10 percent relative to a system without xenon-135 and samarium-149.

**Figure 4a2.6-5-1: Xenon-135 and Samarium-149 Reactivity Worth**

Proprietary Information

#### **RAI 4a2.6-6**

*Provide an uncertainty analysis for the reactivity worths, coefficients, and  $k_{eff}$  values.*

#### **SHINE Response**

Reactivity worths, coefficients, and  $k_{eff}$  values are calculated using MCNP5, version 1.60. The MCNP code and other neutronics codes contain an overall bias error, which results in absolute  $k_{eff}$  predictions deviating from actual system  $k_{eff}$ .

Bias in the calculations is a systematic error due to uncertainties and approximations in the models and methods inherent to MCNP (e.g., thermal scattering treatment), uncertainties in the input data (e.g., cross section data), and approximations and uncertainties in the user input (e.g., simplifications of geometry, material composition uncertainties). MCNP5 calculations have been performed to compare the code against benchmark aqueous systems, and SHINE's review of the benchmark data has found that a bias and bias uncertainty of  $\Delta k = 0.0160$  bounds the expected bias for predictions of absolute  $k_{eff}$  for uranyl sulfate homogenous systems. SHINE is using this bias and bias uncertainty for calculating dimensions required for safe-by-geometry tanks and piping for the facility, which is appropriate for these systems as the code is being used to predict the absolute  $k_{eff}$  of the tanks and piping and the actual margin to criticality of these systems is not measured in the actual plant (i.e., criticality safety relies on the calculations).

SHINE does not plan to use the absolute  $k_{eff}$  predictions from MCNP as the basis to determine operating  $k_{eff}$  of the subcritical assembly. Instead, SHINE plans to use a volume margin-to-critical approach coupled with the calculated reactivity worth of that volume, as described below.

Reference (21) provides a review of the neutron source multiplication method, which is equivalent to the 1/M methodology. As described in Reference (21), the neutron source multiplication method has been used in thousands of measurements of critical masses and configurations and in subcritical neutron-multiplication measurements in situ to provide data for criticality prevention. The neutron source multiplication method has been used on aqueous, homogeneous systems, including uranyl sulfate systems. For far subcritical hydrogen moderated systems (i.e., several dollars subcritical,  $k_{eff} < 0.98$ ), effects of geometry, detector positioning, and modal effects due to the presence of the source can lead to unreliable predictions. However, as the system approaches criticality, these effects become small as the flux shape approaches the dominant modal shape. [ Proprietary Information ] For a wide variety of assemblies, including uranyl sulfate systems, as the system approaches criticality and highly multiplying configurations are achieved, source, counter, and geometry effects become relatively small and a good estimate of criticality is obtained by the 1/M method.

SHINE will also ensure that the detector, source, and subcritical assembly geometry result in conservative 1/M shapes (tailing out shape, as described in Reference (21)). This ensures that early predictions of critical volume are lower than the actual critical volume and result in lower actual  $k_{eff}$  values.



SHINE plans to use the following stepwise approach to determine  $k_{\text{eff}}$  of the subcritical assembly and its uncertainty during Mode 1 operation:

- MCNP5 models will be used to predict the desired uranium concentration to give an acceptable fill height of the TSV. MCNP5 models will be used to predict the critical solution height of the TSV at this uranium concentration.
- During facility startup physics testing, the critical solution height of the TSV will be determined by subcritical approaches and using the 1/M technique. The TSV will remain subcritical for these facility startup physics tests.
- The bias of the MCNP5 models will be determined by computing the difference in uranium concentration between tests and the model that yield the same predicted critical height. Sufficient tests will be performed to provide ensure statistical significance.
- Startup physics tests will include other parameter variations (e.g., uranium concentration, target solution temperature), as determined necessary during detailed design, and the corresponding variations of critical volume as a function of these parameters. These results will be compared to MCNP predictions of the effects of the parameter variations and the differences will be used to estimate the uncertainty of the relevant reactivity coefficients.
- TRPS trip setpoints will be validated and adjusted through the facility startup physics tests by measuring fluxes at different volume margins to critical and parameter variations (e.g., temperature) and comparing them with predictions.
- The MCNP5 models will be used to predict the reactivity worth per liter of target solution for the range of allowable uranium concentrations.
- Normal startups will be performed by using the 1/M methodology. The TSV fill process is planned to be terminated at approximately five percent by volume from critical.
- The actual margin to criticality at Mode 1 startup conditions will be determined by multiplying the measured volume margin to criticality (e.g., 10 liters) by the computed reactivity worth per liter (e.g., 30 pcm/liter of target solution). The allowable margin to criticality will be determined during detailed design to ensure that under abnormal or accident conditions the TSV does not reach criticality. The margin to criticality will be protected by the TRPS trips, which will ensure that abnormal or accident conditions do not result in the TSV reaching criticality.

As the system transitions to Mode 2 irradiation, the system reactivity decreases due to temperature and void changes. The reactivity in the system after this point will not increase above the initial Mode 1 reactivity. The reactivity during Mode 2 irradiations can be estimated by adding the total temperature defect, void defect, and other lesser reactivity changes (e.g., water holdup in the TOGS, xenon and samarium effects) to the reactivity determined in the stepwise plan above for Mode 1 operations. The uncertainty in  $k_{\text{eff}}$  during Mode 2 irradiations is then due to the combination of uncertainties from the initial reactivity from Mode 1 operations and the uncertainty in the reactivity coefficients.

MCNP has been used by Los Alamos National Laboratory (LANL) to accurately predict the behavior of homogeneous solution reactors. While not a reactor, the SHINE system modeling can be validated using these comparisons due to the homogeneous nature of the target solution, the significant subcritical multiplication of the SHINE system at the target  $k_{\text{eff}}$  values, the fact that fission in the system is predominately due to thermal neutrons, and the target solution composition is similar to previous reactors (i.e., ratio of fissile material to hydrogen concentrations within ranges of previous systems).

LANL has modeled previous solution systems and found agreement with their behavior and reactivity coefficients. Results for comparisons of the Kinetic Experiment on Water Boiler (KEWB) reactor are provided in Table 4a2.6-6-1.

**Table 4a2.6-6-1: LANL Comparison of MCNP and KEWB Reactor Measurements**

	<b>KEWB A Core (\$/°C)</b>	<b>KEWB B Core Reflected (\$/°C)</b>	<b>KEWB B Core Unreflected (\$/°C)</b>
MCNP Temperature Coefficient of Reactivity	-0.0468	-0.0395	-0.0554
Experiment Temperature Coefficient of Reactivity	-0.041	-0.037	-0.053
Difference	14%	7%	5%

LANL has also compared the temperature coefficient reported for SILENE against MCNP models at various temperatures. From the available data, the MCNP temperature coefficient, combining spectral and density effects, for SILENE was estimated to match the experimental value within 10 percent.

Given the ability to successfully model other homogeneous systems at LANL, the validation of the MCNP model against homogeneous benchmark systems, and the accurate transient modeling of these homogeneous systems (Reference 22), SHINE expects that the reactivity coefficients and reactivity worths of the SHINE subcritical assembly will be predicted by the MCNP models with an uncertainty of approximately 20 percent, which was determined by adding additional margin to the KEWB and SILENE comparisons performed by LANL. The final uncertainty in the neutronics models predictions of reactivity coefficients and reactivity worths will be determined during facility startup physics testing.

Due to the statistical nature of MCNP, results from MCNP also contain a statistical error. This statistical error will be included in calculations of uncertainty, and is expected to be small in comparison with other sources of error.

The uncertainty in the measured  $k_{\text{eff}}$  during Mode 1 startup will be a combination of instrumentation uncertainties (e.g., fill volume, temperature, neutron flux) and reactivity coefficient/worth uncertainties. As described above, the  $k_{\text{eff}}$  during Mode 1 startup will be determined by multiplying the volume margin to critical by the reactivity worth per volume of solution. The uncertainty in the volume margin to critical is determined from the instrument uncertainties used for the 1/M plot. The uncertainties in the reactivity coefficients/worths are discussed above.

For example, if the overall uncertainty resulting from the combination of these uncertainties was 30 percent, and the reactivity margin to critical was calculated to be [ Proprietary Information ], then the uncertainty in  $k_{\text{eff}}$  would be approximately [ Proprietary Information ]. Even with this uncertainty of 30 percent, the system is maintained below critical with significant margin.

The validation plan, to be performed during facility startup to validate the neutronics predictions, will be included with the SHINE Operating License (OL) Application.

(Applies to RAI 4a2.6-7 through 8)

The ISG Augmenting NUREG-1537, Part 2, Section 4a2.5.1, “Normal Operating Conditions,” states that there should be systems that are “sufficiently redundant and diverse to control all proposed excess reactivity safely and to safely shut down the reactor and maintain it in a shutdown condition.”

The SHINE irradiation unit system relies on dumping the solution to the TSV dump tank under abnormal conditions. SHINE PSAR, Section 4a2.6.3.6, “Redundancy and Diversity of Shutdown Methods,” states that the dump system has redundant dump valves.

#### **RAI 4a2.6-8**

*Provide additional information on the design of the dump valves related to:*

- a) The design drain rate of the TSV when the dump valves are open.*
- b) The delay time from when the conditions would trigger a dump signal until the dump valves start to open.*
- c) The duration of time it takes for the dump valves to open.*

#### **SHINE Response**

- a) The TSV drain system is designed to drain a minimum of 20 gpm when the dump valves are open and liquid height in the TSV is [ Security-Related Information ], which is approximately the minimum fill height for irradiation as described in Subsection 4a2.4.1.2 of the PSAR. The design drain rate is conservatively based on only one drain line available.
- b) The delay time between the conditions that would trigger a dump signal and the start of the dump valves opening will be a maximum of one second.
- c) The duration of time it takes for the dump valves to open will be less than five seconds.

#### **RAI 4a2.6-9**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.5.1, “Normal Operating Conditions,” Acceptance Criteria, states, in part, “[t]he reactivity impacts of radiolytic gas and void formation, fission product gas removal, fuel solution and acid addition, and condensate return to the core should be provided.” This analysis should also include the evaporation of water.*

*SHINE PSAR, Section 4a2.6.1.1, “Gas Management System Effects,” states, in part, “[t]he radiolysis of water in the system causes an anticipated increase in reactivity during operation...”*

*The SHINE PSAR infers that water is constantly leaving the TSV through radiolysis and evaporation. A certain amount of water will be held up outside the TSV as it goes through the recombination and condensation process before it is returned to the TSV, increasing the reactivity in the system.*

*Provide quantitative estimates of the water inventory outside of the TSV, the reactivity increase caused by removing that water from the TSV, and the increase in fuel solution concentration.*

### **SHINE Response**

SHINE estimates that up to [ Proprietary Information ] of water may be held up outside of the TSV in the TOGS. This is approximately [ Proprietary Information ] of the TSV volume. This estimate was performed by calculating the uniform film thickness expected on the TOGS components that are exposed to condensing or wet sweep gas for the bounding evaporation and condensation rates.

The reactivity increase caused by removing that water from the TSV during Mode 2 irradiation has been estimated to be [ Proprietary Information ]. This water removal results in an increase uranium concentration in the target solution of approximately [ Proprietary Information ]. The reactivity increase is small in comparison to the expected subcritical reactivity of the SHINE system during Mode 2 irradiation, which is approximately [ Proprietary Information ].

## **Section 4a2.7 – Thermal-Hydraulic Design**

### **RAI 4a2.7-1**

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.6, “Thermal-Hydraulic Design,” states that the applicant should discuss possible system “instability following perturbation to the system (including from radiolytic gas generation).”*

*Provide linear stability analysis of the full system and an analysis and discussion of the expected bounds of any expected oscillations.*

### **SHINE Response**

A linear stability analysis of subcritical accelerator driven fissile solution systems is provided in Attachment 8. As described in Attachment 8, subcritical accelerator-driven fissile solution systems were found to be unconditionally stable in the linear approximation.

Perturbations from the TOGS can result in pressure changes in the TSV. However, these pressure changes only serve to impose a reactivity feedback term, and do not alter the Attachment 8 conclusions.

Oscillations in TSV fission power are expected to be the result of coupled system oscillations, principally due to potential pressure variations in the TOGS and source oscillations from the neutron driver. Pressure variations in the TOGS at the TSV are expected to be less than 0.5 psi, and this pressure change has been estimated to affect TSV reactivity during irradiation by less than [ Proprietary Information ]. This change in reactivity is estimated to result in approximately a [ Proprietary Information ] change in TSV power.

Oscillations in the neutron driver neutron output are expected to be less than three percent from the target neutron output. With a subcritical assembly and no feedback effects, a change of three percent in the neutron source term would result in a three percent change in the output of the assembly. Due to negative temperature and void reactivity coefficients, the oscillations from this source variation will be less than three percent.

Total oscillations in TSV fission power are expected to be less than [ Proprietary Information ] due to potential superposition of these sources of oscillation. As described in Subsection 4a2.6.1.2 of the PSAR, results of transient modeling of power oscillations using a dynamic model and reactivity feedback effects will be provided with the FSAR.

## **Section 4a2.8 – Gas Management System**

*(Applies to RAIs 4a2.8-1 through 6)*

*The ISG Augmenting NUREG-1537, Part 2, Section 4a2.7, “Gas Management System,” Review Procedures, states that “[t]he reviewer should confirm that the design of the gas management system and the associated analysis are sufficient to provide reasonable assurance of safe operation of the reactor and compliance with all applicable chemical and radiological release criteria.” SHINE PSAR, Section 4a2.8, discusses the gas management system.*

### **RAI 4a2.8-1**

*The capacity of the TSV off-gas recombiner system may be sensitive to the conditions under which it will have to operate.*

*Provide the TSV operating condition envelope and design assumptions for the TSV off-gas recombiner system, including assumed design margins.*

### **SHINE Response**

The following is the TSV operating envelope and TOGS design assumptions for the gas leaving the TSV headspace during normal operation:

- Gas temperature: SHINE will calculate the range for gas temperature during detailed design. The calculation will be based on the return gas temperature from TOGS and the heat transfer rate between the sweep gas and the target solution at a minimum temperature of 68°F and a maximum temperature of 176°F.
- Gas pressure: -1.5 psig to 0 psig (regulated by TOGS)
- Hydrogen production rate: up to approximately 1.2 standard cubic feet per minute (scfm)
- Relative humidity: SHINE will calculate the range for relative humidity during detailed design. The calculation will be based on the return gas humidity from TOGS and the evaporation rate of the target solution at a minimum temperature of 68°F and a maximum temperature of 176°F.

The following are the assumed design margins of the TOGS components:

- TSV Off Gas Condenser: >15% margin for the heat transfer area
- TSV Off Gas Zeolite Beds: >10% margin for the sweep gas flow rate while still maintaining a minimum of 95% efficiency per pass
- TSV Off Gas Blower: >10% margin for the sweep gas flow rate
- TSV Off Gas Recombiner Beds: >25% margin for the recombination rate
- TSV Off Gas Recombiner Condenser: >25% margin for the heat transfer area

#### **RAI 4a2.8-2**

*Provide the basis for an “alert to the operator” at a hydrogen concentration of 2.5 percent and automatic shutdown of the neutron driver at 3 percent. Discuss whether there is sufficient margin to the deflagration limits at these values. Provide information indicating where the measurement of the hydrogen concentration is taken.*

#### **SHINE Response**

Normal operation of the TOGS maintains hydrogen concentrations at or below two percent in the off-gas. The alarm setpoint of 2.5 percent is slightly higher than normal operating conditions to provide advanced warning of abnormal conditions to the operator prior to reaching the trip setpoint, while not resulting in excessive alarms that distract the operators in the Control Room. The hydrogen concentration trip point of three percent provides approximately 33 percent margin to the lower flammability limit of four percent. The hydrogen concentration trip point ensures that the initial hydrogen concentration in the TOGS is sufficiently low in the event of an abnormal condition, such as a blower malfunction.

The margin to the deflagration limit is sufficient because it is not expected that the hydrogen concentration would reach four percent in the event of a failure of a single active component (i.e., a blower) if the initial concentration is below three percent. After a blower failure is detected, such as through reduced flow, the TSV Reactivity Protection System (TRPS) would trip the TSV and neutron driver. A peak hydrogen concentration of 3.9 percent is estimated to occur in the case of a failure of a single active component (blower). This estimate assumes the following:

- The initial hydrogen concentration in the TSV and TOGS is uniformly three percent.
- One second of full-power operation occurs after the blower failure to allow for detection and tripping the safety-related breakers supplying power to the neutron driver high voltage power supply.
- The hydrogen gas contained in the target solution void bubbles out at a rate proportional to the remaining hydrogen in the void.
- Additional hydrogen generated by decay radiolysis enters the TSV headspace as it is produced.

In order to reduce the magnitude of hydrogen concentration changes in the TSV (such as described above), reduce gas flow rate through the headspace, and reduce the potential for target solution droplet entrainment, SHINE has increased the headspace of the TSV by increasing the TSV internal height from approximately [ Security-Related Information ] to approximately [ Security-Related Information ]. Neither the geometry of the target solution during irradiation nor the height of the overflow tubes has been changed. SHINE will update the description of the TSV dimensions in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

The hydrogen concentration measurements for safety-related measurements are expected to be taken in the TOGS after the TSV off-gas condenser and TSV off-gas demister, but before the TSV off-gas recombiner beds. Figure 4a2.8-1 of the PSAR provides the location of the TSV off-gas condenser, TSV off-gas demister, and TSV off-gas recombiner beds in the

TOGS loop. This area was chosen because the hydrogen concentration is sampled prior to recombination and in the expected location of the highest concentration in the TOGS. This location also allows the instruments to be located away from the higher radiation fields near the TSV.

SHINE will perform calculations during detailed design that will ensure there is sufficient margin to deflagration limits. An IMR has been initiated to track completion of these calculations.

#### **RAI 4a2.8-4**

*SHINE PSAR, Table 4a2.8-1, "TSV Off-Gas System Major Components" (page 4a2-69), states that the condenser in the TSV off-gas condenser has a greater than 15 percent heat transfer margin. The vapor pressure of water changes rapidly with temperature in the vicinity of 140 degrees F. For example, increasing the water temperature from 140 degrees F to 150 degrees F increases the vapor pressure by approximately 33 percent. Noncondensable gas can significantly degrade the condensation efficiency in comparison to the condensation of pure steam.*

*Provide the TSV and off-gas system operating conditions and assumptions used to calculate the 15 percent margin.*

#### **SHINE Response**

The following is the TSV operating envelope and TOGS design assumptions for the gas leaving the TSV headspace during normal operation:

- Gas temperature: SHINE will calculate the range for gas temperature during detailed design. The calculation will be based on the return gas temperature from TOGS and the heat transfer rate between the sweep gas and the target solution at a minimum temperature of 68°F and a maximum temperature of 176°F.
- Gas pressure: -1.5 psig to 0 psig (regulated by TOGS)
- Hydrogen production rate: up to approximately 1.2 scfm
- Relative humidity: SHINE will calculate the range for relative humidity during detailed design. The calculation will be based on the return gas humidity from TOGS and the evaporation rate of the target solution at a minimum temperature of 68°F and a maximum temperature of 176°F.

Analysis of the TSV off-gas condenser heat transfer capabilities will be performed during the final design of the system. The condenser analysis will document the inputs and assumptions to the design of the TSV off-gas condenser, including consideration for bounding operating pressures and temperatures, corresponding steam vapor pressure, expected non-condensable gas concentrations, the impact on condenser performance, and operational degradation during the life of the condenser. The required TSV off-gas condenser specifications will be determined based on the bounding inputs and conservative assumptions. Then, an additional 15 percent design margin will be applied to the heat transfer area.

An IMR has been initiated to ensure that the analysis of the TSV off-gas condenser heat transfer capabilities is performed in accordance with the conditions and assumptions described above.

#### **RAI 4a2.8-5**

*SHINE PSAR Section 4a2.8.5 states that a pressure safety valve is connected to the TOGS piping to passively prevent an overpressurization within the PSB, which may cause structural damage to the IU. The setpoint of the pressure safety valve will not exceed the design pressure of the PSB components. This setpoint value will be provided in the FSAR. The TOGS system contains radioactive fission products.*

*Provide information indicating whether the relief valve discharge passes through a system capable of filtering or scrubbing out radioactive fission products. Provide a description of such a system if it exists. If such a system does not exist, provide a discussion of why it is not necessary in relation to meeting radioactive release and dose requirements of 10 CFR Part 20.*

#### **SHINE Response**

SHINE plans on connecting the pressure safety valve connected to the TOGS piping to the Noble Gas Removal System (NGRS). Final design of the NGRS will ensure the system contains a relief volume capable of receiving gas from TOGS in the event of an over-pressurization. Relief gases in the NGRS will be sampled and held for decay. Upon completion of an appropriate decay period, the gases in the NGRS will again be analyzed for radioactivity, and released to the Process Vessel Vent System (PVVS). In the PVVS, the off-gas is mixed with other process vessel exhaust gases, scrubbed through an acid-gas scrubber (caustic scrub solution), and vented to RVZ1, and exhausted out the facility stack following high efficiency particulate air (HEPA) and charcoal filtration. This process will ensure that the radioactive release and dose requirements of 10 CFR 20 are met.



## CHAPTER 5 – COOLING SYSTEMS

### **Section 5a2 – Irradiation Unit Cooling System**

*(Applies to RAIs 5a2.2-1 through 2)*

*The ISG Augmenting NUREG-1537, Part 1, Section 5a2, “Aqueous Homogeneous Reactor [AHR] Cooling System,” states, in part, that “the applicant should give the design bases, descriptions, and functional analyses of the AHR cooling systems. The principal purpose of the cooling systems is to safely remove the fission heat and decay heat from the reactor and dissipate it to the environment. The discussions should include all significant heat sources in the reactor and should show how the heat is safely removed and transferred to the environment.” Additionally, Section 5a2.2, “Primary Cooling System,” specifies discussion of leak detection and allowable leakage limits, if any, and specifies the inclusion of schematic and flow diagrams of the system, showing such essential components as the heat source, heat sink, pumps, piping, valves, control and safety instrumentation, interlocks, and other related subsystems.*

#### **RAI 5a2.2-1**

*In SHINE PSAR, Section 5a2.2.9, “Secondary Cooling System Interaction,” Section 5a2.3.5, “RPCS [Radioisotope Process Facility Cooling System] Cooling Functions and Operation,” and Section 5a2.3.9, “Instrumentation and Control,” pressure, flow, temperature, conductivity, and radiation detection instrumentation are discussed, with pressure being the apparent measurement used to identify system leaks. Additional information is needed for the NRC staff to determine the adequacy of pressure measurement to identify system leaks.*

*Discuss the ability of pressure measurements to identify the presence of small leaks and address how the location of leaks would be determined.*

#### **SHINE Response**

SHINE is not planning to use pressure measurements to detect the presence of small leaks in the Radioisotope Process Facility Cooling System (RPCS).

The pressure in the RPCS is greater than the pressure in the PCLS and LWPS to prevent the transfer of contaminated liquid in the event of a heat exchanger leak. The pressure measurement instrumentation on the PCLS, LWPS, and RPCS ensures that this function is maintained. An alarm will notify the operators if pressures for the PCLS, LWPS, or RPCS are outside of their allowable ranges.

Small leaks out of RPCS will be detected through one of the following methods:

- Rise in the level in the expansion tank (PCLS) or pool (LWPS) for that system over time.
- Periodic sampling and analysis for contaminants is performed on the PCLS and LWPS. The presence of contaminants (such as corrosion inhibitor agents from the RPCS) implies possible leakage and will be investigated.
- Other small leaks from the cooling system to the building environment would be detected and located visually during walkdowns and tours of accessible areas.

These three methods also assist in identifying the specific leaking heat exchanger or component.

### **RAI 5a2.2-3**

*The ISG Augmenting NUREG-1537, Part 2, Section 5a2.2, "Primary Cooling System" Acceptance Criteria, states, in part, that "[t]he primary coolant should provide a chemical environment that limits corrosion of the primary coolant barrier, control and safety rod surfaces, reactor vessels or pools, and other essential components."*

*Chemicals are commonly added to nuclear plant water systems to adjust nuclear reactivity (e.g., boric acid), to control pH (e.g., lithium hydroxide, ammonia/amines), to remove oxygen (e.g., hydrazine), as a biocide (e.g., chlorine), etc. SHINE PSAR, Section 5a.2.2.2, "PCLS [Primary Closed Loop Cooling System] Process Functions," indicates that water quality will be maintained to reduce corrosion and scaling, but this section does not indicate how this will be done. Additional information is needed for NRC staff to understand the impact of potentially toxic additives used to maintain water quality on corrosion and scaling.*

*Provide a list of all potentially toxic chemicals expected to be on the SHINE site for water quality control or for other purposes, including locations and quantities.*

### **SHINE Response**

SHINE will not be using chemical additives with the primary coolant systems, the PCLS and the LWPS. Filters and ion exchange resins will be used to remove contaminants and to maintain water quality parameters.

As described in the SHINE Response to Proposed Action Request #7 (Reference 7), the potentially toxic chemicals used to maintain water quality in the secondary water systems may include non-phosphate buffers (e.g., lithium hydroxide, boric acid, sodium sulfite, sodium lauroyl sarcosinate, or others to be determined during detailed design).

The quantities of chemicals needed for the secondary water systems are expected to be small (i.e., less than five pounds), and will be stored in appropriate chemical storage areas, segregated from incompatible chemicals, in accordance with their Safety Data Sheets. The chemical storage areas are shown in Figure 1.3-2 of the PSAR.

Toxic chemicals used for other purposes are described in Subsection 13b.3.2.2. Those chemicals and amounts are provided in Table 13b.3-1 of the PSAR.

## CHAPTER 6 – ENGINEERED SAFETY FEATURES

### **Section 6a2.1 – Summary Description**

#### **RAI 6a2.1-1**

NUREG-1537, Part 1, Section 6.1, “Summary Description,” states:

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

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

NUREG-1537, Part 2, Section 6.1, “Summary Description,” states:

*In this section of the SAR, the applicant should briefly describe all the ESFs in the facility design and summarize the postulated accidents whose consequences could be unacceptable without mitigation. A specific postulated accident scenario should indicate the need for each the ESF. The details of the accident analyses should be given in Chapter 13 of the SAR and the detailed discussions of the ESFs in Section 6.2 of the SAR. These summaries should include the design bases, the performance criteria, and the full range of reactor conditions, including accident conditions, under which the equipment or systems must maintain function.*

*The applicant may submit simple block diagrams and drawings that show the location, basic function, and relationship of each ESF to the facility. The summary description should contain enough information for an overall understanding of the functions and relationships of the ESFs to the operation of the facility. Detailed drawings, schematic diagrams, data, and analyses should be presented in Section 6.2 of the SAR for each specific ESF.*

*The ISG Augmenting NUREG-1537, Part 1, Section 6a2, “Aqueous Homogeneous Reactor Engineered Safety Features,” states, in part: “... the guidance in this section is general enough to apply to any type of reactor facility, as long as the unique features of each are addressed and appropriate ESFs are provided to ensure that operations are conducted within safe limits.”*

*SHINE PSAR, Section 6a2.1, “Summary Description,” contains a description of the ESFs for the IF, but does not contain enough information for an overall understanding of the functions of the ESFs and the conditions under which the equipment or systems must function.*

- a) *Provide a description of the conditions under which each ESF must function.*
- b) *Provide block diagrams and drawings to show the location, basic function, and relationship of each the ESF to the facility.*

- c) *Specify whether the target solution preparation systems (TSPSs) are part of the irradiation facility or the radioisotope production facility.*
- d) *Specify whether any valves or piping located in the target solution preparation system room are considered part of the confinement boundary for either or both the irradiation facility or the radioisotope production facility.*

### **SHINE Response**

- a) In the IF, engineered safety features (ESFs) consist of active components (e.g., isolation valves, isolation dampers) and passive components (e.g., IU cells) that serve confinement functions during normal conditions of operations and following a design basis accident (DBA).

DBAs that determine what conditions under which the ESFs must function include:

- Mishandling or malfunction of target solution;
- Mishandling or malfunction of equipment affecting the PSB; and
- Tritium Purification System (TPS) DBA.

Based on the postulated accidents, radioactive materials would be released in the IU cells, TOGS shielded cells, or TPS gloveboxes. The postulated releases result in potentially high radiation fields in the IU cells, primary cooling (PCLS and LWPS) rooms, TOGS shielded cells, and TPS gloveboxes. The physical structures and components that comprise the passive boundary, including penetration seals, will be designed to perform their confinement function under the expected radiation exposure for the duration of the accident. Wherever practical, the active components will be located in areas of lower radiation levels (e.g., the bubble-tight dampers are to be located outside shield walls). The active components will be designed to perform their confinement function under the postulated radiation exposure for the duration of the accident.

Some ESF components in the IU cells and primary cooling rooms, such as PCLS isolation valves, have the potential to be exposed to acidic target solution in the event of release of target solution to the PCLS. These ESF components must function under this acidic environment, and will be chemically compatible with the target solution.

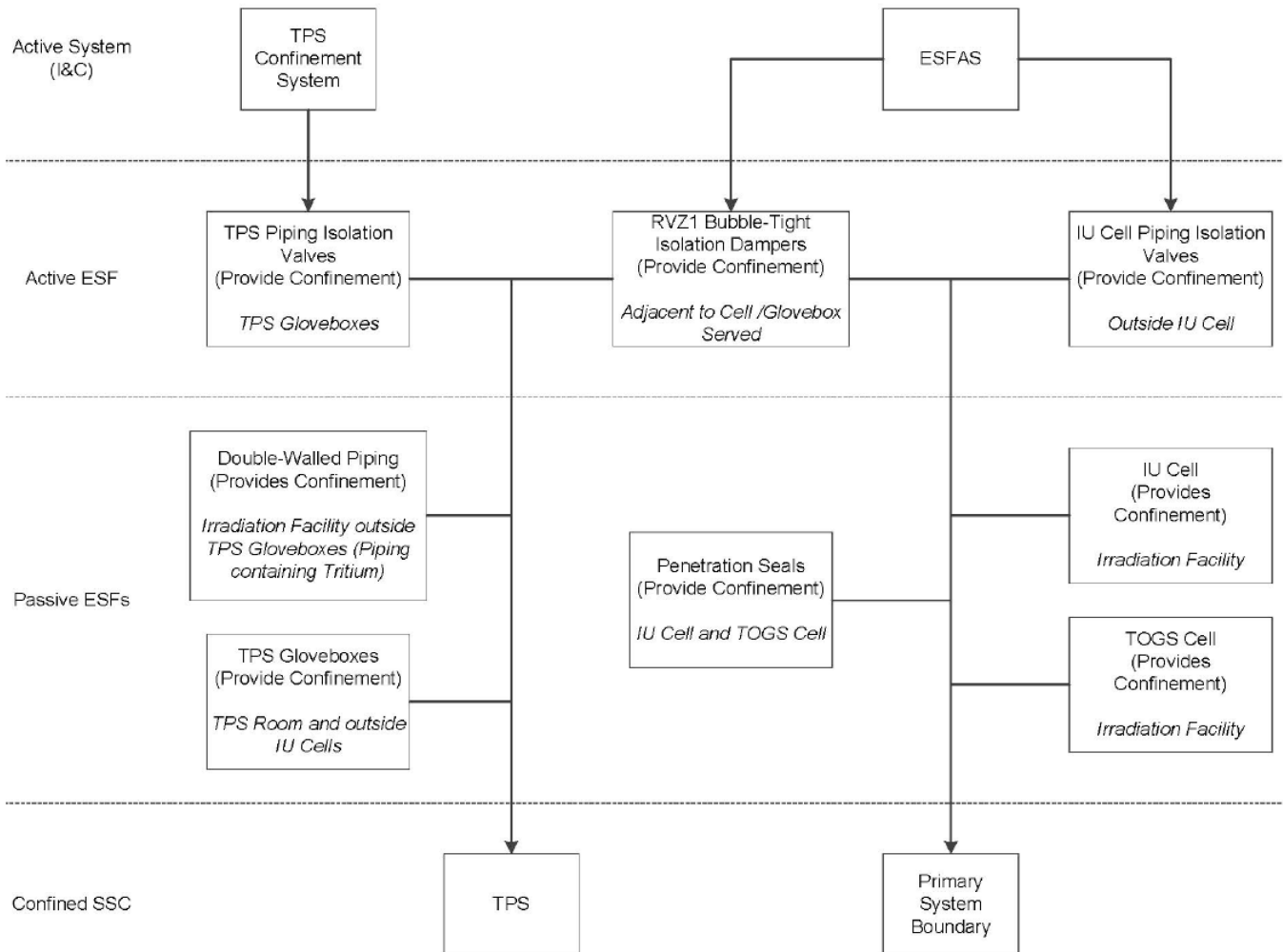
Due to the low pressure and low temperature nature of the SHINE IF processes, the IF ESF components only need to function under near ambient pressure and temperature conditions, except if credited for isolation for fire scenarios. For IF ESF components that are credited to perform isolation functions during postulated fires, they will be capable of performing the required level of isolation in the potential environments (e.g., elevated temperature) determined from the fire hazards analysis (FHA).

- b) A block diagram for the IF ESFs is provided as Figure 6a2.1-1-1. This block diagram shows the basic function of the SSCs providing the confinement ESF in the IF portion of the SHINE facility, the relationship to the facility, and the relationship between each ESF SSC.

Figure 6a2.1-1-1 provides a block diagram representation of Table 6a2.1-1 of the PSAR, which summarizes the IF DBAs and the ESFs provided for mitigation of those DBAs. The ESF credited with DBA mitigation in the IF is confinement. The SSCs providing in the confinement ESF are:

- IU cells, including penetration seals
- RVZ1 ductwork up to bubble-tight isolation dampers
- Bubble-tight isolation dampers
- Isolation valves on piping systems penetration the IU cells
- TOGS shielded cells, including penetration seals
- Engineered Safety Features Actuation System (ESFAS)
- Double-walled pipe used for the TPS
- TPS gloveboxes
- TPS confinement system

**Figure 6a2.1-1-1: Block Diagram of Irradiation Facility Confinement Engineered Safety Features**



- c) See Reference 4 for Part c of the SHINE Response to RAI 6a2.1-1.
- d) See Reference 4 for Part d of the SHINE Response to RAI 6a2.1-1.

## **Section 6a2.2 – Irradiation Facility Engineered Safety Features Detailed Description**

*(Applies to RAIs 6a2.2-1 through 9)*

*NUREG-1537, Part 2, Section 6.2, “Detailed Descriptions,” states, in part: “In this section of the SAR; the applicant should discuss in detail particular ESF systems that may be incorporated into the reactor design.”*

*NUREG-1537, Part 1, Section 6.2.1, “Confinement,” states, in part: “The applicant should discuss in detail the confinement and the associated HVAC [heating, ventilation, and air conditioning] systems that function as ESFs.”*

*NUREG-1537, Part 2, Section 6.2.1, “Confinement,” states, in part: “If the HVAC and any air exhaust or liquid release systems associated with the confinement are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13 and thereby mitigate its consequences, they should be considered part of the confinement ESF and should be discussed in this section of the SAR.”*

### **RAI 6a2.2-1**

*SHINE PSAR, Section 6a2.2.1, “Confinement,” discusses a system called the “tritium purification system (TPS) confinement system,” but the section did not provide sufficient information for the staff to understand the entire system.*

*Provide additional information that describes and defines the “TPS confinement system,” including system boundaries and interfaces, with references to the appropriate diagram(s).*

### **SHINE Response**

The TPS confinement system consists of the passive and active features of the tritium purification system that are used to limit potential tritium releases to acceptable levels. The TPS confinement system boundaries consist of:

- The TPS gloveboxes, including the pressure protection bubbler and glovebox airlock;
- The outer jacket of double-walled piping that transfers tritium outside of gloveboxes; and
- The TPS confinement isolation valves that isolate portions of the tritium system upon loss of integrity to limit tritium releases.

The double-walled piping annulus is open to the glovebox atmosphere in order to allow for detection and cleanup if a release from the inner pipe occurred. The necessary locations of isolation valves will be determined from the accident analysis during detailed design. Isolation valves will include valves that isolate the NDAS from the TPS should a leak in NDAS be detected.

The TPS confinement system will interface with following systems:

- RVZ1 (exhaust gas from the glovebox is purged to the ventilation system)
- NDAS (tritium supply and contaminated return)
- Inert Gas Control System (IGS) (supply of nitrogen for glovebox atmosphere)
- Facility Inert Gas System (FIGS) (supply of liquid nitrogen for operation of the isotope separation process)
- Normal Electrical Power Supply System (NPSS) (for electrical supply to components within the glovebox)
- ESFAS (for actuation of the TPS confinement isolation valves)

A diagram of the TPS showing the gloveboxes and double-walled tritium piping is provided in Figure 9a2.7-1 of the PSAR. Isolation valve locations will be determined during detailed design and provided in the FSAR. An IMR has been initiated to track the inclusion of the isolation valve locations in the FSAR.

### **RAI 6a2.2-3**

*SHINE PSAR, Section 6a2.2.1.2, "Confinement System and Components," states, in part: "A failure of the TPS outside the glovebox is mitigated by the TPS confinement system. The TPS confinement system uses isolation valves to stop a tritium leak outside the glovebox when a leak is detected."*

*Additional information is needed for the NRC staff to determine the adequacy of the design of the TPS confinement system.*

*Provide additional information on the design and function of the TPS confinement system, including the ability of the system to stop tritium leaks outside of the glovebox.*

### **SHINE Response**

As described in the SHINE Response to RAI 6a2.2-1, the TPS confinement system boundaries consist of:

- The TPS gloveboxes, including the pressure protection bubbler and glovebox airlock;
- The outer jacket of double-walled piping that transfers tritium outside of gloveboxes; and
- The TPS confinement isolation valves that isolate portions of the tritium system upon loss of integrity to limit tritium releases.

Tritium outside the glovebox is contained in piping normally under vacuum, which assists in reducing the potential for releases. Pressure detection is expected to be used to monitor for a leak in these lines, since an unexpected increase in pressure indicates a potential leak. Once the pressure rises past the allowable setpoints, isolation valves will close in order to reduce the potential amount of tritium released. The confinement isolation valve closure time will be accounted for in the accident analysis and the assumed value will be bounding.

The only significant portion of the tritium inventory that is not in a confinement area or double-walled piping is the tritium in the neutron drivers. The evaluation of the release of tritium from the neutron drivers is described in Subsection 13a2.2.12.3. Isolation valves will isolate the NDAS from the TPS should a leak in NDAS be detected.

Isolation valve locations will be determined during detailed design and provided in the FSAR. An IMR has been initiated to track the inclusion of the isolation valve locations in the FSAR.

#### **RAI 6a2.2-4**

*SHINE PSAR, Section 6a2.2.1.3, "Functional Requirements," states, in part, "Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced." Additional information is needed for the NRC staff to determine the adequacy of the SHINE design to withstand and mitigate adverse environments.*

*Provide information on the assumed "adverse environments" and how components are designed to accommodate for them.*

#### **SHINE Response**

The assumed adverse environments for the active confinement components in the IF are due to the release of radioactive materials. These assumed adverse environments are high radiation fields in the IU cells, primary cooling (PCLS and LWPS) rooms, TOGS shielded cells, and TPS gloveboxes, and acidic environments in the PCLS and LWPS. The active confinement components (e.g., isolation dampers, isolation valves) will be designed to perform their functions under the expected radiation conditions for the duration of the accident. The active confinement components that may be exposed to acidic environments (e.g., isolation valves on PCLS and LWPS) due to target solution release will be chemically compatible with the target solution.

Except for dampers credited for isolation during postulated fires, ESF components in the IF are not expected to experience adverse temperature or pressure environments due to the low temperature and low pressure nature of the SHINE IF processes. Dampers that are credited to perform isolation functions during postulated fires will be capable of performing the required level of isolation in the potential environments (e.g., elevated temperature) determined from the FHA.

#### **RAI 6a2.2-7**

*SHINE PSAR, Section 6a2.2.1.4, "Confinement Components," indicates that the details of the TPS confinement system will be left to the FSAR. Additional information is needed for the NRC staff to determine the adequacy of waiting to provide details of the TPS confinement system in the FSAR.*

*Provide the rationale for leaving the details of TPS confinement to the FSAR.*

#### **SHINE Response**

The TPS confinement system is described in the SHINE Response to RAI 6a2.2-1. Details of the components that comprise the TPS are described in Subsection 9a2.7.1 of the PSAR.



Isolation valve locations will be determined during detailed design and provided in the FSAR. The specific valve locations will be dependent on the final design and layout of the system. However, the valves locations and number will be sufficient to limit consequences from accidents to less than 10 CFR 20 limits. An IMR has been initiated to track the inclusion of the isolation valve locations in the FSAR.

#### **RAI 6a2.2-9**

*SHINE PSAR, Table 6a2.2-1, "Irradiation Facility Confinement Safety Functions" (page 6a2-9), references isolation valves on piping systems, but the applicant does not identify the valves, provide a list of the valves or reference a schematic which details the isolation valves.*

*Provide a list, schematic or reference to a list of the isolation valves.*

#### **SHINE Response**

Subsection 6a2.2.1.4 of the PSAR describes confinement components of systems normally open to the IU cells, TOGS shielded cells, and TPS glovebox. The IU cells and TOGS shielded cells will have RVZ1 inlet and outlet bubble-tight isolation dampers to achieve the confinement boundary. A schematic of RVZ1 is shown in Figure 9a2.1-1 of the PSAR.

As shown in Figure 4a2.1-1 of the PSAR, the IU cell has penetrations for the PCLS and LWPS. These systems provide cooling to the TSV, and both systems will be provided with isolation valves for confinement.

TPS confinement is achieved by the TPS Confinement System. The TPS will also contain isolation valves. As described in the SHINE Response to RAI 6a2.2-1, the necessary locations of isolation valves will be determined from the accident analysis during detailed design, and will include valves that isolate the NDAS from the TPS should a leak in NDAS be detected. Additional detail regarding the TPS isolation valves is also provided in the SHINE Response to RAI 6a2.2-3.

Additional isolation valve details will be developed during detailed design, and the FSAR will be updated with a list, details, or locations of these isolation valves. An IMR has been initiated to track the inclusion of this information in the FSAR.

### **RAI 6a2.2-11**

*NUREG-1537, Part 1, Section 6.2.1, "Confinement," states, in part: "The discussion of mitigative effects should contain a comparison of potential radiological exposures to the facility staff and the public with and without the ESF"*

*NUREG-1537, Part 2, Section 6.2.1, "Confinement," Evaluation Findings, states, in part: "This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:*

- The scenarios for all potential accidents at the reactor facility have been analyzed by the applicant and reviewed by the staff. Mitigation of consequences by a confinement system has been proposed in the SAR analyses for any accident that could lead to potential unacceptable radiological exposures to the public, the facility staff, or the environment.*
- The staff has reviewed the designs and functional descriptions of the confinement ESF; they reasonably ensure that the consequences will be limited to the levels found acceptable in the accident analyses of Chapter 13 of the SAR.*
- The designs and functional descriptions of the confinement ESF reasonably ensure that control of radiological exposures or releases during normal operation will not be degraded by the ESF."*

*SHINE PSAR, Section 6a2.2.1, "Confinement," does not contain a comparison of potential radiological exposures to the facility staff and the public with and without the ESF.*

*Provide the comparative study or reference the section of SHINE PSAR, which provides this information.*

### **SHINE Response**

Table 6a2.1-1 of the PSAR provides the IF DBAs that are mitigated by the IF confinement ESF. The IF maximum hypothetical accident (MHA) is also mitigated by the same confinement ESFs, as it is an extension of the mishandling or malfunction of target solution.

A comparative study of the ESFs for mitigating these IF DBAs is contained in Table 6a2.2-11-1. For the unmitigated consequences, SHINE assumed none of the active ESFs functioned. The only active ESF components in the confinement credited for mitigating these accidents are the bubble-tight isolation dampers. The remaining assumptions and input values used to calculate the doses are the same between the unmitigated and mitigated cases.

The unmitigated and mitigated radiation doses to workers at the SHINE facility are the same, as no active ESF components are credited in the accident analysis for the workers.

The leak path factors in Chapter 13 of the PSAR for the DBA of the TPS were incorrect in relation to the provided doses. SHINE has revised the TPS DBA described in Chapter 13 of the PSAR to correct the discrepancy and provide the reduced public doses. A mark-up of the PSAR changes is provided in Attachment 9. The non-public (proprietary) version of the PSAR, incorporating the changes provided in Attachment 9, is provided in Enclosure 3. The public (non-proprietary) version of the PSAR, incorporating the changes provided in Attachment 9, is provided in Enclosure 4.

**Table 6a2.2-11-1: Comparison of Unmitigated and Mitigated Radiological Doses for IF DBAs**

Event	Unmitigated Public Dose (rem)				Mitigated Public Dose (rem)			
	Site Boundary		Nearest Resident		Site Boundary		Nearest Resident	
	TEDE	Thyroid	TEDE	Thyroid	TEDE	Thyroid	TEDE	Thyroid
Target Solution Release into the IU Cell (IF Postulated MHA)	1.65E+00	1.58E+00	2.30E-01	2.21E-01	1.65E-02	1.58E-02	2.30E-03	2.21E-03
Mishandling or Malfunction of Target Solution	2.19E-01	1.58E+00	3.06E-02	2.21E-01	2.19E-03	1.58E-02	3.06E-04	2.21E-03
Mishandling or Malfunction of Equipment Affecting the PSB	1.59E+00	7.03E-02	2.23E-01	9.84E-03	1.59E-02	7.03E-04	2.23E-03	9.84E-05
TPS Design Basis Accident	5.6E-02	---	8E-03	---	5.6E-04	---	8E-05	---

## **Section 6b.1 – Summary Description of Engineered Safety Features**

### **RAI 6b.1-1**

*NUREG-1537 Part 1, Section 6.1, “Summary Description,” states:*

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

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

*NUREG-1537 Part 2, Section 6.1, “Summary Description,” states, in part:*

*In this section of the SAR, the applicant should briefly describe all the ESFs in the facility design and summarize the postulated accidents whose consequences could be unacceptable without mitigation. A specific postulated accident scenario should indicate the need for each ESF. The details of the accident analyses should be given in Chapter 13 of the SAR and the detailed discussions of the ESFs in Section 6.2 of the SAR. These summaries should include the design bases, the performance criteria, and the full range of reactor conditions, including accident conditions, under which the equipment or systems must maintain function.*

*The applicant may submit simple block diagrams and drawings that show the location, basic function, and relationship of each ESF to the facility. The summary description should contain enough information for an overall understanding of the functions and relationships of the ESFs to the operation of the facility. Detailed drawings, schematic diagrams, data, and analyses should be presented in Section 6.2 of the SAR for each specific ESF.*

*SHINE PSAR, Section 6b.1, "Summary Description Engineered Safety Features," contains a description of the ESFs for the Radioisotope Production Facility but does not contain enough information for an overall understanding of the functions of the ESFs and the conditions under which the equipment or systems must function.*

- a) *Provide a description of the conditions under which the system must function.*
- b) *Provide block diagrams and drawings to show the location, basic function, and relationship of each ESF to the facility.*
- c) *SHINE PSAR, Section 6b.1 states, in part: "The confinement systems provide for active isolation of piping and HVAC systems penetrating confinement boundaries in certain post-accident conditions." Explain what is meant by the word "certain" in this context.*

### **SHINE Response**

- a) In the RPF, ESFs consist of active components (e.g., isolation valves, isolation dampers) and passive components (e.g., hot cells) that serve confinement functions during normal conditions of operations and following a DBA.

DBAs that determine what conditions under which the ESFs must function include:

- Critical equipment malfunction; and
- Accidents with hazardous chemicals.

Based on the postulated accidents, radioactive materials could be released in the RPF inside of hot cells, process piping shielded subfloor trenches, shielded tank vaults, or the noble gas storage cell. The physical structures and components that comprise the passive boundary, including penetration seals, will be designed to perform their confinement function under the expected radiation exposure for the duration of the accident. Wherever practical, active components will be located in areas of lower radiation levels (e.g., bubble-tight dampers are to be located outside shield walls). The active components will be designed to perform their confinement function under the postulated radiation exposure for the duration of the accident.

Additionally, based on the postulated accidents, hazardous chemicals may be released in confinement areas during an accident (e.g., sulfuric acid from target solution releases, nitric acid from Uranyl Nitrate Conversion System (UNCS) releases). The physical structures and components (e.g., cells) that comprise the passive boundary will be designed to perform their confinement function given the potential chemical exposure for the duration of the accident. Active components will also be designed using suitable materials for the potential chemical exposures to ensure they perform their confinement function for the duration of the accident. Some ESF components, such as piping system isolation valves, have the potential to be exposed to a variety of acidic, basic, oxidizing, and flammable chemicals.

These ESF components must function under these environments, and will be chemically compatible with the potential exposures.

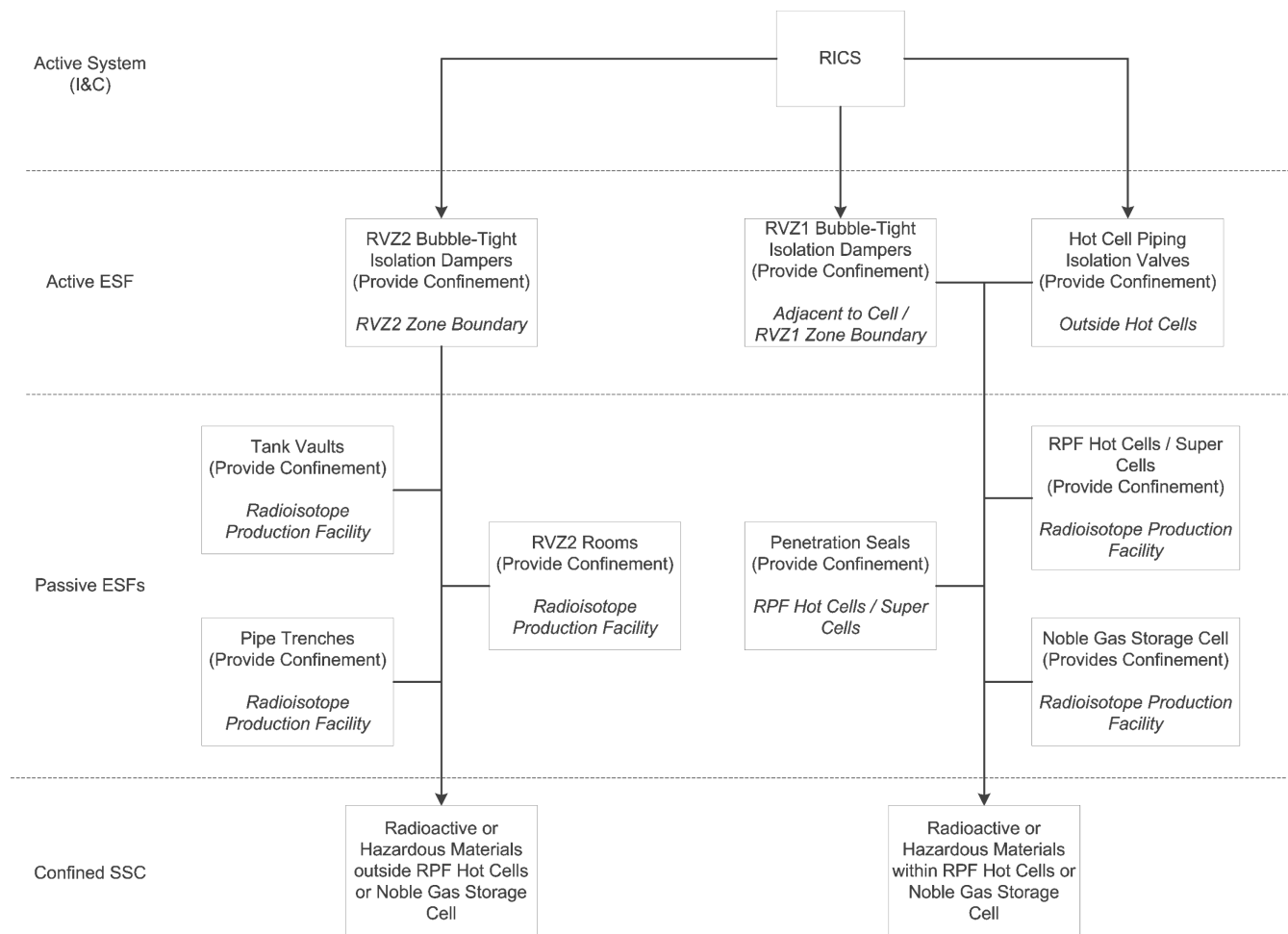
Except if credited for isolation during a fire, the RPF ESF components only need to function under near ambient pressure and temperature conditions due to the low temperature and pressure of the systems requiring confinement. For RPF ESF components that are credited to perform isolation functions during postulated fires, they will be capable of performing the required level of isolation in the potential environments (e.g., elevated temperature) determined from the FHA. Slightly elevated pressures may be present in the noble gas shielded cell following NGRS accidents due to the release of pressurized gases. ESF isolation dampers serving the noble gas shielded cell will be designed to function under the potential pressures.

- b) A block diagram for the RPF ESFs is provided as Figure 6b.1-1-1. This block diagram shows the basic function of the SSCs providing the confinement ESF in the RPF portion of the SHINE facility, the relationship to the facility, and the relationship between each ESF SSC.

Figure 6b.1-1-1 provides a block diagram representation of Table 6b.1-1 of the PSAR, which summarizes the RPF DBAs and the ESFs provided for mitigation of those DBAs. The ESF credited with DBA mitigation in the RPF is confinement. The SSCs providing the confinement ESF are:

- Hot cells, including penetration seals
- RVZ1 (including ductwork up to filters and filters) and RVZ2
- Bubble-tight isolation dampers
- Tank vaults
- Radiological Integrated Control System (RICS)
- Isolation valves on piping systems penetrating hot cells

**Figure 6b.1-1-1: Block Diagram of Radioisotope Production Facility Confinement Engineered Safety Features**



c) See Reference 4 for Part c of the SHINE Response to RAI 6b.1-1.

## Section 6b.2 – Radioisotope Production Facility Engineered Safety Features

*(Applies to RAIs 6b.2-1 through 4)*

*NUREG-1537, Part 2, Section 6.2, “Detailed Descriptions,” states, in part: “In this section of the SAR; the applicant should discuss in detail particular ESF systems that may be incorporated into the reactor design.”*

*NUREG-1537, Part 1, Section 6.2.1, “Confinement,” states, in part: “The applicant should discuss in detail the confinement and the associated HVAC systems that function as ESFs.”*

*NUREG-1537, Part 2, Section 6.2.1, "Confinement," states, in part: "If the HVAC and any air exhaust or liquid release systems associated with the confinement are designed to change configuration or operating mode in response to a potential accident analyzed in Chapter 13, and thereby, mitigate its consequences, they should be considered part of the confinement ESF and should be discussed in this section of the SAR."*

#### **RAI 6b.2-1**

*SHINE PSAR, Section 6b.2.1.3, "Functional Requirements," states, in part: "Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced."*

*However, the section does not discuss the postulated adverse environments in detail. Provide detailed information on the postulated adverse environments and how components are designed to accommodate for them.*

#### **SHINE Response**

The assumed adverse environments for the active confinement components in the RPF are due to the release of radioactive materials and the release of hazardous chemicals. These assumed adverse environments are high radiation fields in the hot cells, process piping shielded subfloor trenches, shielded tank vaults, or the noble gas storage cell, and hazardous chemical exposure in the hot cells, shielded tank vaults, and process piping shielded subfloor trenches. The active confinement components will be designed to perform their functions under the expected radiation conditions for the duration of the accident. The active confinement components will be designed to perform their functions under the potential chemical conditions of the process to which they may be exposed.

Except if credited for isolation during a fire, the RPF ESF components only need to function under near ambient pressure and temperature conditions due to the low temperature and pressure of the systems requiring confinement. For RPF ESF components that are credited to perform isolation functions during postulated fires, they will be capable of performing the required level of isolation in the potential environments (e.g., elevated temperature) determined from the FHA. Slightly elevated pressures may be present in the noble gas shielded cell following NGRS accidents due to the release of pressurized gases. ESF isolation dampers serving the noble gas shielded cell will be designed to function under the potential pressures.

#### **RAI 6b.2-6**

*NUREG-1537, Part 1, Section 6.2.1, "Confinement," states, in part: "The discussion of mitigative effects should contain a comparison of potential radiological exposures to the facility staff and the public with and without the ESF."*

*NUREG-1537, Part 2, Section 6.2.1, "Confinement," Evaluation Findings, states, in part: "This section of the SAR should contain sufficient information to support the following types of conclusions, which will be included in the safety evaluation report:*

- *The scenarios for all potential accidents at the reactor facility have been analyzed by the applicant and reviewed by the staff. Mitigation of consequences by a confinement system has been proposed in the SAR analyses for any accident that could lead to potential unacceptable radiological exposures to the public, the facility staff, or the environment.*

- *The staff has reviewed the designs and functional descriptions of the confinement ESF; they reasonably ensure that the consequences will be limited to the levels found acceptable in the accident analyses of Chapter 13 of the SAR.*
- *The designs and functional descriptions of the confinement ESF reasonably ensure that control of radiological exposures or releases during normal operation will not be degraded by the ESF."*

*SHINE PSAR, Section 6b.2.1, "Confinement," does not contain the confinement ESF effectiveness comparison in the discussion of mitigative effects. Provide the comparative study, or reference the section of SHINE PSAR, which provides the information.*

### **SHINE Response**

Table 6b.1-1 of the PSAR provides the RPF DBAs that are mitigated by the RPF confinement ESF. The RPF MHA is also mitigated by the same confinement ESFs, as it is an extension of a critical equipment malfunction scenario. Section 6b.2 of the PSAR contains an administrative error stating that an RPF fire does not have consequences that require mitigation by ESFs. An IMR has been initiated to track correction of this statement in the FSAR.

A comparative study of the ESFs for mitigating these RPF DBAs is contained in Table 6b.2-6-1. For the unmitigated consequences, SHINE assumed none of the active ESFs functioned. The only active ESF components in the confinement credited for mitigating these accidents are the bubble-tight isolation dampers. The remaining assumptions and input values used to calculate the doses are the same between the unmitigated and mitigated cases.

The unmitigated and mitigated radiation doses to workers at the SHINE facility are the same, as no active ESF components are credited in the accident analysis for the workers.

**Table 6b.2-6-1: Comparison of Unmitigated and Mitigated Radiological Doses for RPF DBAs**

Event	Unmitigated Public Dose (rem)				Mitigated Public Dose (rem)			
	Site Boundary		Nearest Resident		Site Boundary		Nearest Resident	
	TEDE	Thyroid	TEDE	Thyroid	TEDE	Thyroid	TEDE	Thyroid
Release of Inventory Stored in NGRS Storage Tanks (RPF MHA)	8.20E-01	---	1.15E-01	---	8.20E-02	---	1.15E-02	---
Critical Equipment Malfunction: Loss of Piping or Tank Integrity	2.19E-01	1.58E+00	3.06E-02	2.21E-01	2.19E-03	1.58E-02	3.06E-04	2.21E-03
Critical Equipment Malfunction: Inadvertent Release from NGRS	8.17E-01	---	1.14E-01	---	8.17E-02	---	1.14E-02	---
RPF Fire	8.77E-03	1.60E-01	1.23E-03	2.24E-02	8.77E-04	1.60E-02	1.23E-04	2.24E-03



## **Section 6b.3 – Nuclear Criticality Control**

*(Applies to RAIs 6b.3-1 through 20)*

*As required by 10 CFR 50.34(a)(4), “[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.”*

*As stated in the ISG Augmenting NUREG-1537, Chapter 13, the NRC staff has determined that the use of integrated safety analysis (ISA) methodologies as described in 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material and NUREG-1520, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,” Revision 1, May 2010, application of the radiological and chemical consequence and likelihood criteria contained in the performance requirements of 10 CFR Section 70.61, designation of IROFS, and establishment of management measures are acceptable ways of demonstrating adequate safety for the medical isotopes production facility. Applicants may propose alternate accident analysis methodologies, alternate radiological and chemical consequence and likelihood criteria, alternate safety features, and alternate methods of assuring the availability and reliability of the safety features. As used in this ISG, the term “performance requirements,” when referencing 10 CFR Part 70, Subpart H, is not intended to mean that the performance requirements of Subpart H are required for a radioisotope production facility license, only that their use as accident consequence and likelihood criteria may be found acceptable by NRC staff.*

### **RAI 6b.3-1**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, “Nuclear Criticality Safety for the Processing Facility,” states, in part that “[c]riticality process safety controls should be provided for criticality safety, and a description of their safety function should be described. The applicant should use enough safety controls to demonstrate that, under normal and abnormal credible conditions, all nuclear processes remain subcritical” and that “NCS [nuclear criticality safety] limits on controlled parameters will be established to ensure that all nuclear processes are subcritical, including an adequate margin of subcriticality for safety.”*

*For example, the applicant could commit to base the safety limits on validated calculation methods. These methods should be industry-accepted and peer-reviewed. Also, the applicant should commit to ensuring that methods used to develop NCS limits will be validated to confirm that they are used within acceptable ranges and that the applicant used both appropriate assumptions and acceptable computer codes.*

*In multiple places in SHINE PSAR (e.g., pgs. 1-4, 6b-17, and 14b-2), the applicant implies safety limits are determined utilizing MCNP and validated methods. Also, while Section 6b.3.1 provides NCS criteria, this information is insufficient.*

*a) State explicitly if safety limits are determined utilizing MCNP and validated methods.*

- b) *Provide additional clarification as to exactly what methods and assumptions are proposed for use in determining if NCS criteria are met. Include summary description of a documented, reviewed, and approved validation report or reference manual (by NCS function and management) for each methodology that will be used to perform an NCS analysis (e.g., experimental data, reference books, hand calculations, deterministic computer codes, probabilistic computer codes). Additionally, provide the validation report and reference manual referred to in the PSAR.*

### **SHINE Response**

- a) Preliminary criticality scoping safety assessments for the SHINE facility used to determine criticality safety limits use a verified installation of Monte Carlo N-Particle (MCNP), which has been validated within the area of applicability of the processes being evaluated. SHINE does not currently plan to use other methods for determination of criticality safety limits.
- b) The methods and assumptions used to determine that nuclear criticality safety (NCS) criteria are met are provided in the validation reports supporting the SHINE project. A general validation report, provided as Attachment 10, and a project-specific validation report, provided as Attachment 11, has been created. The general validation report discusses the use of MCNP5 using ENDF/B-VI cross sections for homogeneous uranium systems, and the project-specific report discusses the use of MCNP5 using ENDF/B-VII cross sections for uranyl sulfate systems.

The general validation report contains the statistical analysis of the comparison of the probabilistic computer code MCNP5 with benchmark criticality experiments, and determines the appropriate reactivity bias to use for homogeneous uranium systems at thermal neutron energies. The project-specific validation report confirms that MCNP5 calculations using ENDF/B-VII cross sections are sufficiently accurate to establish the SHINE safety basis when used with suitable conservative assumptions.

The reference manual will contain information such as methodology and assumptions. SHINE will create the reference manual as part of the nuclear criticality safety program during detailed design, and will be provided as part of the SHINE OL Application. An IMR has been initiated to track the inclusion of the reference manual in the SHINE OL Application.

### **RAI 6b.3-2**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," Acceptance Criteria, states, in part, that the reviewer should determine if the applicant commits to "establish[ing] and maintain[ing] NCS safety limits and operating limits for the possession and use of fissile material and to maintain[ing] management measures to ensure the availability and reliability of the controls."*

*SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," has no discussion regarding applicable management measures as required in 10 CFR 70.62(d), and as defined in 10 CFR 70.4.*

*Provide either the relevant passages in the SHINE PSAR that address applicable management measures or provide information to discuss management measures. Specifically, describe change management, configuration control, quality assurance, and procurement programs and measures for assuring long term reliability and availability of engineered controls (such as geometry, absorbers, etc.).*

### **SHINE Response**

Administrative controls for the SHINE facility will ensure that engineered controls and control systems that are identified as NCS controls are designed, implemented, and maintained, to ensure they are available and reliable to perform their function when needed. Management measures identified through the accident analysis process, including NCS safety limits and operating limits for the possession and use of fissile material, will be included in the Administrative Controls section of the SHINE Technical Specifications (TS) or will be described in the FSAR and implemented through SHINE procedures, and will ensure that safety-related SSCs are available and reliable to perform their functions when needed. The management measures classification has been removed from the PSAR (FSAR) (see Part a of the SHINE Response to RAI 3.5-1 (Reference 4)).

Procedures will be developed and put in place prior to operation so that personnel will not change NCS controls without proper NCS review by a qualified person. As noted in Section 6b.3 of the PSAR, changes to nuclear criticality safety evaluations (NCSEs) that involve or could affect special nuclear material (SNM) will be evaluated under 10 CFR 50.59 (e.g., new design, operation, or modification to existing SSCs; computer programs; processes; operating procedures; administrative controls). The NCSE program will be integrated with a broader configuration management program that incorporates 10 CFR 50.59. This program will permit changes to the facility as described in the FSAR, changes in the procedures as described in the FSAR, and the conduct of tests or experiments not described in the FSAR if prior NRC approval pursuant to 10 CFR 50.90 is not required prior to implementation of the change.

The SHINE quality assurance program is described in the SHINE QAPD. The quality assurance program will ensure long term reliability and availability of engineered controls through robust design control. Design verification will ensure adequacy of engineered controls, and design change control will ensure that these engineered controls are both reliable and available.

SSCs identified as engineered NCS safety controls required to prevent or mitigate criticality accidents (such as geometry, absorbers, etc.) will be designated as safety-related in accordance with the SHINE definition of safety-related (see Part a of the SHINE Response to RAI 3.5-1). As described in the SHINE QAPD, safety-related SSCs receive the full measure of quality assurance.

Requirements of the SHINE configuration management program will ensure the design and licensing basis requirements of plant SSCs are properly reflected in plant documents such as drawings, calculations, equipment specifications, procedures, and software, and that these documents properly reflect the physical plant SSC configuration.

The SHINE procurement requirements are described in Section 2.4 and Section 2.7 of the SHINE QAPD. The procurement program will have established procedures to ensure that items and services that are related to the reliability and availability of safety-related engineering controls are acceptable. Additionally, verification activities for purchased items and services related to the reliability and availability of engineered controls will take place.

#### **RAI 6b.3-4**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, “Nuclear Criticality Safety for the Processing Facility,” states, in part, that the reviewer should determine “whether the margin of subcriticality for safety is sufficient to provide reasonable assurance of subcriticality.”*

*While the SHINE PSAR, including Section 6b.3, mentions a subcritical margin, additional information is needed for the NRC staff to determine the adequacy of the subcritical margin.*

*Identify and justify the use of a subcritical margin for use in NCSEs, accident analyses, and development of safety controls. This should include conservative assumptions that are incorporated into evaluations to assure that processes should be less reactive than evaluated. The NRC staff notes that this may be information that is included in the summary of the NCS reference manual.*

#### **SHINE Response**

Criticality safety limits are determined at an upper subcritical limit of  $k_{\text{eff}}$ , which includes a margin of subcriticality value of  $0.05 \Delta k$ . This margin of subcriticality is consistent with guidance contained in NUREG/CR-5661 (Reference 23) and NUREG-1520 (Reference 24). This margin of subcriticality is sufficient for NCSEs, accident analyses, and safety control development for the SHINE facility based on the well-known nature of the material and the well-defined processes being used. The reactivity bias and uncertainty value for homogeneous uranium systems from preliminary criticality scoping safety assessments of the SHINE facility is [ Proprietary Information ]. The margin of subcriticality and the reactivity bias and uncertainty value are subtracted from one to determine the upper subcritical limit that is used to determine the criticality safety limits. This upper subcritical limit is equal to [ Proprietary Information ].

Sufficient conservatisms, including bounding geometry and material assumptions, are incorporated into preliminary criticality scoping safety assessments of the SHINE facility, as appropriate. Conservatisms used in the preliminary criticality scoping safety assessments of the SHINE facility include:

- Temperature of 20°C to maximize reactivity by avoiding Doppler broadening absorption in U-238.
- Solute saturation is assumed to be unlimited in order to show peak reactivity regardless of concentration.
- No excess acid is modeled in uranyl sulfate to maximize reactivity.
- Uranium enrichment of 21 percent U-235, higher than the SHINE facility upper bound of 19.95 percent.
- Uranium dioxide maximum theoretical density of 10.96 g/cc to maximize uranium amount.
- Uranium metal maximum theoretical density of 19.05 g/cc to maximize uranium amount.
- Water theoretical maximum density of 0.9982 g/cc at minimum light water pool operation temperature of 20°C to maximize reflection.

When conservatisms on geometry and material are not appropriate or possible, criticality safety controls and limits are provided. These limits are sufficient to maintain the system at an adequately subcritical level based on the calculations which include the margin of subcriticality.

### **RAI 6b.3-5**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," Acceptance Criteria, states, in part, that the reviewer should determine whether the applicant commits to "establish[ing] and maintain[ing] NCS safety limits and operating limits...." This commitment should assume optimum credible conditions (i.e., the most reactive conditions physically possible or limited by written commitments to regulatory agencies) unless specified controls are implemented to control the limit to a certain range of values.*

*A commitment to establishing and maintaining NCS safety limits, including optimum credible conditions, is not clearly delineated in the SHINE PSAR.*

*Provide a discussion committing to establishing and maintaining NCS safety limits, including optimum credible conditions.*

### **SHINE Response**

SHINE will establish and maintain NCS safety limits and operating limits. The limits will assume optimum credible conditions unless specified controls are implemented to control the limit to a certain range of values (e.g., the liquid waste storage tanks are not analyzed at optimum credible conditions as specified controls are placed on uranium concentration entering these tanks).

### **RAI 6b.3-6**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine if, when they are relevant, the applicant considers heterogeneous effects. Heterogeneous effects are particularly relevant for low-enriched uranium processes, where, all other parameters being equal, heterogeneous systems are more reactive than homogeneous systems.*

*SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," states that "[h]eterogeneous effects are not considered applicable because the uranium enrichment is less than 20 percent."*

*Explain and justify this assumption, especially as one of the processes involves dissolution of special nuclear material (SNM) in metal form.*

### **SHINE Response**

The enrichment of the uranium being used in the SHINE facility is less than 19.95 percent U-235. However, the preliminary criticality scoping safety assessments were performed using a bounding uranium enrichment of 21 percent U-235. The preliminary criticality scoping assessments for the SHINE facility use homogenous conditions, which have been shown to be the most reactive condition (Reference 25).

Reference (25) discusses the effect of heterogeneous versus homogeneous uranium systems for various U-235 enrichments and the physics of the system when uranium is contained in heterogeneous "lattices" such as in power reactors. The reason for smaller critical masses at lower U-235 enrichments is due to the consequence of the absorbing characteristics of U-238 for neutrons having energies of a few electron volts, a property called resonance absorption. When the uranium is latticed, as in a reactor, there is a greater probability of immediate neutron

energy degradation from the high energy at which neutrons are produced by fission to less than that at which U-238 is strongly absorbing. These neutrons “escape” the U-238 resonance absorption and the probability for the escape is a measureable and calculable property of such lattices. The maximum U-235 enrichment of the uranium at which latticing (heterogeneous systems) can reduce the critical mass is estimated to be about six percent U-235. The critical mass of uranium below this enrichment can be lower for a heterogeneous system than for a homogeneous system. Since the SHINE facility is using uranium with an enrichment above six percent U-235, heterogeneous effects can be ignored and conservative homogeneous uranium conditions used.

SHINE will update Section 6b.3 of the PSAR in the FSAR to add clarification to the statement justifying the use of homogeneous uranium conditions instead of heterogeneous effects. An IMR has been initiated to track the update to Section 6b.3.

### **RAI 6b.3-7**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, states that the reviewer should determine whether the applicant’s use of geometry as a controlled parameter is acceptable if, before beginning operations, all dimensions and nuclear properties that use geometry control are verified. The facility configuration management program should be used to maintain these dimensions and nuclear properties.*

*SHINE PSAR, Section 6b.3, “Nuclear Criticality Control,” includes little discussion on the configuration management program (e.g., page 6b-16 and Table 6b.3-2).*

*Provide additional detail on the role of the configuration management program (i.e., the configuration control process, procedures addressing the process, and how the change management program will ensure that changes to the NCS basis are incorporated into procedures, evaluations, postings, drawings, other safety-basis documentation, and the ISA summary) to allow the staff to evaluate its implementation.*

### **SHINE Response**

The SHINE configuration management program will ensure that criticality safety controls defined in the NCSEs will not be changed without appropriate review by a qualified criticality safety engineer. This ensures that criticality safety controls remain in place as designed to maintain subcritical operations. Procedures will be developed and put in place prior to operation so that personnel will not change NCS controls without proper NCS review.

The change management program will include incorporating NCS controls into operating procedures and equipment drawings and identifying them as such to ensure they are not changed or removed without proper NCS review. Prior to implementing a change, an evaluation must conclude that the entire process will remain subcritical, with an approved margin for safety, under normal and credible accident conditions. Changes to NCS controls will be documented in the appropriate NCSE.

As described in Section 6b.3 of the PSAR, changes to NCSEs will be evaluated under 10 CFR 50.59 (e.g., new design, operation, or modification to existing SSCs; computer programs; processes; operating procedures; or administrative controls). Should the 10 CFR 50.59 review determine prior NRC approval is required, an amendment request in accordance with 10 CFR 50.90 will be submitted to the NRC for approval prior to implementation of the change.

#### **RAI 6b.3-8**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of moderator as a controlled parameter is acceptable.*

*The SHINE PSAR does not address moderator as a controlled parameter.*

*Verify that this will not be a controlled parameter and how this will be addressed in the NCSEs.*

#### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility include optimum moderation conditions. The preliminary design of the SHINE facility does not contain systems that require moderation as the sole criticality safety controlled parameter. Therefore, there are no plans to have moderation controlled areas within the SHINE facility. In the NCSEs, moderation will be listed as not being a controlled parameter.

*(Applies to RAIs 6b.3-9 through 10)*

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of concentration as a controlled parameter is acceptable (e.g., concentrations of SNM in a process are limited unless the process is analyzed to be safe at any credible concentration; when using a tank containing concentration-controlled solution, the tank is normally closed and locked to prevent unauthorized access; when concentration needs to be sampled, dual independent sampling methods are used; and, after identification of possible precipitating agents, precautions are taken to ensure that such agents will not be inadvertently introduced).*

#### **RAI 6b.3-9**

*The SHINE PSAR does not address concentration as a controlled parameter other than to state it is a control for selected equipment.*

*Discuss the use of concentration as a controlled parameter and how this will be addressed in the NCSEs.*

#### **SHINE Response**

Preliminary criticality scoping safety assessments for the SHINE facility have been performed at optimum concentration, with the exception of the evaluation of the liquid waste processing tanks. The liquid waste processing tanks are the only tanks or vessels at the SHINE facility that are not criticality-safe by geometry where fissionable material exists outside the TSV, since

these tanks are not expected to ever contain an appreciable amount of fissionable material. Limits on uranium concentration of fissionable material are applied to these tanks to ensure criticality safety. NCS controls will be established within the NCSEs and will comply with the double contingency principle. These controls will ensure they are available and reliable such that uranium bearing solution which exceeds the NCS limit is prevented from entering the non-favorable geometry vessels.

#### **RAI 6b.3-10**

*Page 6b-17 of the SHINE PSAR states, "Each of the tanks within the scope of this section features criticality safety controls that meet the double-contingency principle...the first criticality safety control is that each tank, with the exception of the tanks associated with liquid waste processing, is criticality safe by geometry or by the combination of geometry and a layer of neutron absorbing material integral to the tank construction. The second, independent criticality-safety control is that the most reactive concentration of uranium in any tank results in  $k_{\text{eff}}$  less than or equal to 0.95."*

*Clarify how concentration is independently subcritical given the design concentrations for process equipment listed in the PSAR. Specifically, clarify how the most reactive concentration of uranium in the uranyl sulfate preparation tank, 1-TSPS-01T, independently results in  $k_{\text{eff}}$  less than or equal to 0.95.*

#### **SHINE Response**

Favorable geometry vessels for criticality safety meet the double contingency principle based on two process controls. The first control is the geometry of the vessel, which is designed to remain subcritical based on its design dimensions, the optimum concentration of fissile material, and, in some cases, the presence of neutron-absorbing materials that are integral to the tank construction under all normal and credible abnormal conditions. The second control is the SHINE configuration management program. Favorable geometry vessels have been shown to remain subcritical for credible abnormal process conditions at the optimum concentration of fissile material and changes in design dimensions during operation. The only way to cause the favorable geometry vessels to become critical would be to replace the vessels with a design that does not have favorable geometry. The replacement of a vessel with a design that has unfavorable geometry is prevented by the SHINE configuration management program. Therefore, favorable geometry vessels comply with the double contingency principle without need to control concentration.

The uranyl sulfate preparation tank (1-TSPS-01T) has been designed to have a favorable geometry. The tank may contain uranyl sulfate at any concentration and remain below the upper subcritical limit, which includes a 0.05  $\Delta k$  margin of subcriticality. This ensures that the vessel will have a  $k_{\text{eff}}$  of 0.95 or less under all normal and credible abnormal conditions.

Subsection 6b.3.1 of the PSAR contains an administrative error, which states:

*"... and a layer of neutron absorbing material integral to the tank construction. The second, independent criticality-safety control is that the most reactive concentration of uranium in any tank results in  $k_{\text{eff}} \leq 0.95$ , based on MCNP analyses."*



SHINE will revise the above statement in the FSAR to read:

*“The second control is the configuration management program. Favorable geometry vessels have been shown to remain subcritical for all credible abnormal process conditions at the optimum concentration of fissile material and changes in design dimensions during operation. The only way to cause the favorable geometry vessels to become critical would be to replace the vessels with a design that is not favorable geometry. This process upset is prevented by the configuration management program. Therefore, all favorable geometry vessels comply with the double contingency principle without need to control concentration.”*

An IMR has been initiated to track the correction to Subsection 6b.3.1.

*(Applies to RAIs 6b.3-11 through 12)*

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, “Nuclear Criticality Safety for the Processing Facility,” Areas of Review, states, in part, that:*

- Criticality accident analyses should be identified, including the assumption that all criticality accidents are high-consequence events and that the applicant’s bases and methods are based on using preventive controls.*
- Criticality process safety controls should be provided for criticality safety, and a description of their safety function should be described. The applicant should use enough safety controls to demonstrate that, under normal and abnormal credible conditions, all nuclear processes remain subcritical.*
- Criticality management measures should ensure that the reliability and availability of the safety controls are adequate to maintain subcriticality.*

#### **RAI 6b.3-11**

*SHINE PSAR, Chapter 13, (page 13b-29), states that “an inadvertent criticality event inside a shielded concrete vault within the facility is not an event of significant concern.” Also, on page 6b-3, the PSAR states that inadvertent nuclear criticality in the radioisotope production facility is a design basis accident that does not have consequences requiring mitigation by ESFs. In addition, on page 6b-11, SHINE commits to ANSI/ANS-8.10-1983 (R2005), “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement” (ANSI/ANS, 2005a).*

*Any inadvertent criticalities are reportable to the NRC and are indication of a loss of applicable process controls.*

*Provide additional information providing the bases for asserting that an inadvertent criticality event inside a shielded concrete vault is not an event of significant concern and that an inadvertent nuclear criticality in the radioisotope production facility is a DBA that does not have consequences requiring mitigation by ESFs. Include a discussion demonstrating that under all normal and abnormal credible conditions, subcriticality will be maintained.*

### **SHINE Response**

The referenced statement on Page 13b-29 of the PSAR incorrectly states that an inadvertent criticality event inside a shielded concrete vault within the facility is not an event of significant concern. This statement has been removed from the PSAR (see the SHINE Response to RAI 13b.1-1).

Any unintended criticality is an event of significant concern and is an indication of a loss of process controls. The SHINE design ensures that an unintended criticality is not a credible event. Subsection 13b.2.5 of the PSAR provides a description of safety controls that maintain subcriticality under all normal and abnormal conditions throughout the RPF. Because these safety controls prevent unintended criticality, ESFs that mitigate such an event are not required. Through the combination of controls described in Section 6b.3 and Subsection 13b.2.5 of the PSAR, subcriticality will be maintained in the RPF under all normal and abnormal credible conditions.

### **RAI 6b.3-13**

*The ISG to NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," Acceptance Criteria, states, in part, that the reviewer should determine whether "the applicant describes a program that ensures compliance with the double-contingency principle, where practicable." In a very few processes, double-contingency protection may not be practicable. In those rare instances, the applicant should provide adequate justification for why such cases are acceptable.*

*The applicant commits to the double contingency principle in multiple passages in the PSAR, including Section 6b.3, (e.g., pages 1-4, 6b-12, and 17); however, no mention is made as to whether there are any planned exceptions to the double-contingency principle.*

*Clarify that all processes will be compliant with the double-contingency principle or provide justifications for those which will not.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility ensure compliance with the double contingency principle for planned system operations. There are currently no planned exceptions to compliance with the double contingency principle.

#### **RAI 6b.3-14**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant understands/acknowledges that use of a single NCS control to maintain the values of two or more controlled parameters constitutes only one component necessary to meet the double-contingency protection.*

*Provide clarification, indicating whether a single NCS control is used to maintain the values of two or more controlled parameters, and acknowledge that any such a control constitutes only one component necessary to meet the double-contingency protection.*

#### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility do not maintain the values of two or more controlled parameters with a single NCS control. When a single NCS control is capable of controlling two or more criticality controlled parameters, another control will be identified to provide double contingency protection.

#### **RAI 6b.3-15**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that "the reviewer should review all aspects of the applicant's NCS program, including management, organization, and technical practices. The reviewer should identify and note any items or issues relating to the NCS program and commitments that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the commitments made in the license application are implemented through procedures and training."*

*While SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," commits to ANSI/ANS-8.26, "Criticality Safety Engineer Training and Qualification Program," 2007 on page 6b-11 and also on page 6b-12, this is not explicitly discussed as a requirement of the NCS program and it is somewhat confused with a more general training commitment for plant personnel.*

*Provide an explicit commitment to having NCS staff trained and qualified to this ANSI guidance. Also provide, as supplemental information, the training and qualifications of staff evaluating the processes for NCS in the initial facility design.*

#### **SHINE Response**

As described in Section 6b.3 of the PSAR, SHINE NCS staff will be trained and qualified in accordance with ANSI/ANS-8.26-2007 (R2012) (Reference 26).

The staff that have worked on the preliminary criticality scoping safety assessments for the SHINE facility are qualified with a program that is compliant with ANSI/ANS-8.26-2007 (R2012). The current SHINE criticality safety staff are contractors employed by Atkins Nuclear Solutions US (formerly Nuclear Safety Associates). The training and qualification records, including re-qualification records, of the two Atkins Nuclear Solutions US staff evaluating the processes for NCS in the initial facility design are included in Attachments 12 and 13.

### **RAI 6b.3-16**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine if the applicant's use of mass as a controlled parameter is acceptable under stated circumstances, that is, when mass limits are derived for a material that is assumed to have a given weight percent of SNM, determinations of mass are based on either (1) weighing the material and assuming that the entire mass is SNM or (2) conducting physical measurements to establish the actual weight percent of SNM in the material; when fixed geometric devices are used to limit the mass of SNM, a conservative process density is assumed in calculating the resulting mass; and, when the mass is measured, instrumentation subject to facility management measures is used.*

*This information is not apparent in the discussion of mass as a controlled parameter in SHINE PSAR, Section 6b.3, page 6b-18.*

*Provide clarification of the use of this controlled parameter, if applicable.*

### **SHINE Response**

Specifics of the control of mass as a criticality safety controlled parameter are not yet defined and will be determined as part of the detailed design. However, if mass is used as a controlled parameter, determinations of mass will be based on either:

- 1) Weighing the material and assuming the entire mass is SNM; or
- 2) Conducting physical measurements to establish the actual weight percent of SNM in the material. When fixed geometric devices are used to limit the mass of SNM, a conservative process density is assumed in calculating the resulting mass; and, when the mass is measured, instrumentation subject to facility administrative controls is used.

### **RAI 6b.3-17**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of density as a controlled parameter is acceptable.*

*SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," does not address density as a controlled parameter. Verify whether density will be a controlled parameter and how this parameter will be addressed in the NCSEs.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility do not control density for purposes of nuclear criticality safety. Preliminary criticality scoping safety assessments involving dry and moderated fissile solids assumed theoretical density of the material. In the NCSEs, density will be listed as not being a controlled parameter.

### **RAI 6b.3-18**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of enrichment as a controlled parameter is acceptable.*

*SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," does not address enrichment or other likely SNM components (e.g., plutonium) as a controlled parameter.*

*Verify whether enrichment will be a controlled parameter and how this parameter will be addressed in the NCSEs.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility have made no requirements to control the U-235 enrichment. Preliminary criticality scoping safety assessments have been performed at the maximum bounding uranium enrichment value of 21 percent U-235. In the NCSEs, enrichment will be listed as not being a controlled parameter.

### **RAI 6b.3-19**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of reflection as a controlled parameter is acceptable.*

*SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," does not address reflection as a controlled parameter.*

*Verify whether reflection will be a controlled parameter and how this parameter will be addressed in the NCSEs.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility have made no requirements to control reflection. Preliminary criticality scoping safety assessments have been performed using worst-case credible reflection conditions. In the NCSEs, reflection will be listed as not being a controlled parameter.

### **RAI 6b.3-20**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of interaction as a controlled parameter is acceptable (i.e., the structural integrity of the spacers or racks should be sufficient for normal and credible abnormal conditions).*

*While other aspects of interaction are addressed, SHINE PSAR, Section 6b.3, does not explicitly state that use of interaction is a controlled parameter.*

*Specify whether interaction control will be used, and how it would be applied.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility do use interaction control in conjunction with geometry control related to arrays of cylindrical tanks and storage rack arrays. These geometric designs control the spacing between multiple storage locations for racks and the spacing of tanks containing fissile material. The structural integrity of the features will be designed such that proper spacing between fissile material units is maintained during normal and credible abnormal conditions. Administrative controls will be applied to ensure that NCS controls on interaction remain available and reliable.

### **RAI 6b.3-21**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," states that the reviewer should determine whether the applicant's use of volume as a controlled parameter is acceptable. SHINE PSAR, Section 6b.3, "Nuclear Criticality Control," does not appear to address volume as a controlled parameter.*

*Specify whether volume will be a controlled parameter and explain how this parameter will be addressed in the NCSEs.*

### **SHINE Response**

The preliminary criticality scoping safety assessments for the SHINE facility do use volume control as a controlled parameter. Volume control assumes an optimum concentration in a spherical geometry of fissile material as well as full water reflection and allows for criticality safety to be demonstrated without regards to geometric shape. Administrative controls will be applied to ensure that NCS controls on volume remain available and reliable.

### **RAI 6b.3-22**

*The ISG Augmenting NUREG-1537, Part 2, Section 6b.3, "Nuclear Criticality Safety for the Processing Facility," Areas of Review, states, in part, that "[c]riticality process safety controls should be provided for criticality safety, and a description of their safety function should be described. The applicant should use enough safety controls to demonstrate that, under normal and abnormal credible conditions, all nuclear processes remain subcritical."*

*SHINE PSAR, Section 6b.3.1, "Criticality-Safety Controls," provides information on nuclear criticality safety evaluations.*

*Provide a representative sample of several nuclear criticality safety evaluations to improve staff's understanding of the methods of processes being utilized.*

### **SHINE Response**

Formal NCSEs have not been performed for the preliminary design of the SHINE facility. The analysis of the preliminary design has been scoping in nature. However, credible normal and abnormal conditions have been considered in of the preliminary criticality scoping safety assessments for the preliminary design and the nuclear processes remain subcritical. The NCSEs for the final design will ensure that under credible normal and abnormal conditions, nuclear processes remain subcritical.

## CHAPTER 7 – INSTRUMENT AND CONTROL SYSTEMS

### **Section 7a2.2 – Design of Instrument and Control Systems**

#### **RAI 7a2.2-1**

*NUREG-1537, Part 1, Chapter 7, “Instrumentation and Control Systems,” Section 7.2.2, “Design-Basis Requirements,” states, in part, that the “design bases for the I&C system, subsystems, and components should include the following, as applicable:*

- The range of values that monitored variables may exhibit for normal operation, shutdown conditions, and for postulated accidents.*
- The specification of precision and accuracy requirements for the instruments, control subsystems, or components.”*

*SHINE PSAR, Table 7a2.2-2, “IF Verification Matrix Design Criteria, Bases, Description” (Sheet 9 of 10), states, in part, that “the amount and rate of reactivity increases during the fill and irradiation processes are limited through physical and control system design to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the primary system boundary greater than limited local yielding, nor, (2) sufficiently disturb the target solution vessel, its support structures or other target solution vessel internals to impair significantly the capability to drain the target solution vessel.” However, there is insufficient information supporting these assertions for the staff to determine if the design provides reasonable assurance that the design criteria will be met.*

*Provide additional information to support the assertions in this section of the PSAR, particularly supporting details on the accuracy anticipated for the reactivity control and the criteria for determining that draining of the target solution vessel is not impaired.*

#### **SHINE Response**

Part 1 of NUREG-1537 (Reference 27) states, in part, that the design bases for the instrument and control (I&C) system, subsystems, and components should include the range of values that monitored variables may exhibit for normal operation, shutdown conditions, and for postulated accidents. SHINE will monitor neutron flux and process variables during the entire fill (Mode 1) and irradiation (Mode 2) process. Neutron flux is plotted to obtain the 1/M versus the fill volume (height) during the fill process and continuously monitored during the irradiation process to ensure that measured parameters do not exceed acceptable limits. Reactivity in the TSV will only be directly controlled during the startup process and when the target solution is drained. The nominal margin to critical (e.g., five percent by volume) will be controlled through the 1/M process. The expected accuracies for the parameters used for reactivity monitoring are provided in Table 7a2.2-1-1. The expected ranges of variables for normal operation and postulated accidents are also provided in the table. The accuracy of the overall margin to critical during startup is described in the SHINE Response to RAI 7a2.3-1.



**Table 7a2.2-1-1: Reactivity Monitoring Variables, Ranges, and Accuracies**  
**(Design Basis for Reactivity Control and Protection Trips – Nominal)**

Function	Variable Monitored	Range of Variable	Expected Accuracy
Source Range	Neutron Flux	6 decades, 0.1 – 10 <sup>5</sup> counts per second	± 11% of span
Wide Range	Neutron Flux	8 decades, 10 <sup>-8</sup> to 200% on a logarithmic scale	± 12.5% of span
High Range	Neutron Flux	0 – 125% on a linear scale	± 7%
TSV	Level	0 – 100% of vessel	± 2.5%
	Flow (Fill Line)	0 – 120% of rated flow	± 1%

During the irradiation (Mode 2) process, the control system does not directly control the reactivity. The physical design and characteristics of the subcritical assembly (e.g., natural convection cooling design, nuclear feedback coefficients) determine the reactivity response.

The target solution does not require reactivity monitoring after it is transferred into the TSV dump tank during Mode 3, because it is contained in a criticality-safe geometry and is passively cooled by the light water pool.

The draining of the TSV is monitored by valve position indication on the TSV dump valves and TSV level instrumentation. Only one of the two dump valves is required to function to meet the required drain rate for the TSV. Periodic surveillance testing will be performed to verify the draining of the TSV is not impaired by observing the rate of TSV level decrease following opening of the TSV dump valves, which will be compared to the required rate. Discrepancies identified during surveillance testing (e.g., significant decrease in flow rate) will be placed into the Corrective Action Program.

## **Section 7a2.3 – TSV Process Control Description**

### **RAI 7a2.3-1**

*NUREG-1537, Part 2, Section 7.3, "Reactor Control System," Acceptance Criteria, states, in part: "The RCS [Reactor Control System] should give continuous indication of the neutron flux from subcritical source multiplication level through the licensed maximum power range."*

*SHINE PSAR, Section 7a2.3.2.1, "Mode 1 - Startup Mode," states that the startup process calculates the subcritical multiplication factor M from the neutron flux level and plots 1/M versus the fill volume (height). This is then compared to a predicted graph of acceptance values for the same parameter. However, it is not clear how bias and uncertainties associated with the benchmarking of criticality calculations, together with the expected variability in process parameters and instrumentation readings are being considered.*

*Provide additional information regarding the uncertainties in these computations, including a quantitative estimate of the expected overall uncertainty in their subcritical reactivity values during startup.*

### **SHINE Response**

The SHINE Response to RAI 4a2.6-6 discusses the uncertainties in  $k_{\text{eff}}$  and subcritical reactivity during startup. As discussed in that response, SHINE plans to use a volume margin to critical approach, which will determine the target solution volume below critical during startup and calculate the subcritical reactivity by multiplying this volume margin to critical by the reactivity worth per volume. By using this approach, the bias of criticality calculations is eliminated from the calculation process.

## CHAPTER 8 – ELECTRICAL POWER SYSTEMS

### **Section 8a2.2 –Emergency Electrical Power Systems**

#### **RAI 8a2.2-1**

*NUREG-1537, Part 1, Section 8.2, “Emergency Electric Power Systems,” states, in part: “In this section, the applicant should present a detailed functional description and circuit diagrams. In the design bases, the applicant should discuss if non-interruptible electrical power is required in the transfer from normal to emergency electrical service and if the transfer is manual or automated. The design bases should also provide voltage and power requirements for the emergency electrical power systems, the time duration over which these could be needed, and assurance that fuel will be available for the time required. The designs of the emergency electrical systems should provide that any use for non-safety-related functions could not cause loss of necessary safety-related functions. The design discussion should show how the emergency power supply system is isolated or protected, if necessary, from transient effects, such as power drains, short circuits, and electromagnetic interference.”*

*NUREG-1537, Part 2, Section 8.2, “Emergency Electrical Power Systems,” Acceptance Criteria, states in part: “Any non-safety-related uses of an emergency electrical power system should not interfere with performance of its safety-related functions.”*

*SHINE PSAR, Section 8a2.2.1, “Class 1E UPSS [Uninterruptible Power Supply System],” references SHINE PSAR Figure 8a2.2-1, “One-Line Diagram – Uninterruptible Electrical Power Supply System” (ADAMS Accession No. ML13172A298), for UPSS components configuration. SHINE PSAR, Section 8a2.1.11, “Raceway and Cable Routing,” states, in part, “[n]on-Class 1E circuits are electrically isolated from Class 1E circuits by isolation devices in accordance with IEEE [Institute of Electrical and Electronics Engineers] 384 (IEEE, 2008).”*

*SHINE PSAR, Figure 8a2.2-1 shows the Class 1E/non-Class 1E boundaries for uninterruptible power supply system Divisions A and B as horizontal dashed lines with arrows pointing upward toward what the annotation indicates is the non-Class 1E side. For both divisions, the drawing shows the Class 1E/non-Class 1E boundaries to be situated between the first load circuit breakers from the respective facility 480-Vac standby diesel generator (SDG) bus supplying each division’s Class 1E battery charger and Class 1E 480V-208Y/120V voltage-regulating transformer and the respective input/supply circuit breakers for those battery chargers and voltage-regulating transformers.*

*Class 1E isolation devices are located and designed to function to isolate non-Class 1E circuits with sustained overloads or faults from otherwise unaffected Class 1E circuits powered from a common source to preserve the continuity of power to the otherwise unaffected Class 1E circuits.*

*Because the SDG buses normally provide power to both Class 1E and non-Class 1E loads, then theoretically, all the non-Class 1E load circuit breakers from the SDG busses, or their respective local supply breakers could be considered Class 1E isolation devices that must trip open to clear faults or sustained overloads on the non-Class 1E loads in order to preserve continuity of power to the Class 1E loads.*

*However, is not clear which circuit breakers are considered Class 1E isolation devices. It is necessary to know which circuit breakers serve as Class 1E isolation devices, because even though they may be enclosed in the switchgear for non-Class 1E busses, and considered physically part of the non-Class 1E portion of the electrical power distribution system, they must perform a Class 1E function. Therefore, they must be classified as Class 1E themselves.*

*Provide additional information to explain the design approach to Class 1E isolation and to designate which circuit breakers in the electrical power distribution systems for the SHINE facility are to serve as Class 1E isolation devices. Additionally, explain the bases for those designations, how the type of circuit breakers designated as Class 1E isolation devices will be reasonably assured of meeting the specifications for such devices in accordance with IEEE Standard 384-2008.*

### **SHINE Response**

The 480 VAC standby diesel generator (SDG) buses shown on Figure 8a2.1-1 and Figure 8a2.2-1 of the PSAR are non-Class 1E buses. These buses provide the normal power supply to the Class 1E systems and are not designed to maintain continuity of power to the Class 1E systems upon a loss of off-site power or a fault on the buses. Therefore none of the circuit breakers on these buses are considered Class 1E isolation devices that must trip open to clear faults or sustained overloads on the non-Class 1E loads.

The Class 1E isolation devices are described in Subsection 8a2.2.2 of the PSAR and shown in Figure 8a2.2-1 of the PSAR as "Class 1E Battery Charger A," "Voltage Regulating Xfmr Assembly A Class 1E," "Class 1E Battery Charger B," and "Voltage Regulating Xfmr Assembly B Class 1E." These devices isolate the Class 1E 250 VDC and Class 1E 120 VAC Uninterruptible Power Supply System (UPSS) busses from the non-Class 1E 480 VAC SDG busses.

The Class 1E battery chargers and voltage regulating transformers are Class 1E isolation devices that meet the requirements of Section 6.1.2.3 of IEEE 384-2008 (Reference 28) by limiting the input current to an acceptable value under faulted conditions.

During detailed design, the Class 1E battery chargers and voltage regulating transformers are to be specified to include electrical isolation requirements in accordance with IEEE 384-2008. The suppliers of these devices are to submit test reports to demonstrate compliance with the electrical isolation requirements of IEEE 384-2008. An IMR has been initiated to track receipt of these reports.

## CHAPTER 9 – AUXILIARY SYSTEMS

### **Section 9a2.1 – Heating, Ventilation, and Air Conditioning Systems**

*(Applies to RAIs 9a2.1-1 through 2)*

*NUREG-1537, Part 2, Section 9.1, “Heating, Ventilation, and Air Conditioning Systems,” Acceptance Criteria, states, in part: “The design and operating features of the system should ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur.”*

#### **RAI 9a2.1-1**

*SHINE PSAR, Section 9a2.1.1, “Radiologically Controlled Area Ventilation System,” discusses the following systems: Radiological Controlled Area (RCA) Zone 2 Supply Air (RVZ2SA), RCA Ventilation System Zone 1 (RVZ1) Exhaust, RCA Ventilation System Zone 2 (RVZ2) Exhaust, and RCA Ventilation System Zone 3 (RVZ3). In reviewing this section with Figure 9a2.1-1, “RVZ1 Ventilation Flow Diagram,” and Figure 9a2.1-2, “RVZ2SA and RVZ2 Ventilation Flow Diagram” (ADAMS Accession No. ML13172A300), the following was noted:*

- a) The section states that RCA Zone 2 Supply Air supplies air to RCA Ventilation System Zone 2 and RCA Ventilation System Zone 3, but there is no mention of where RCA Ventilation System Zone 1 gets its supply air from. SHINE PSAR, Figure 9a2.1-2, has an arrow after the supply fans that states: “Supply Air Flows to Additional Rooms” but provides no clarification as to what rooms/areas receive the air. PSAR, Figure 9a2.1-1 has an arrow going into the irradiation unit cell and an arrow going into the hot cell. Both arrows have the following statement “Transfer Air from Zone 2.” It is not clear if the supply to the irradiation unit and hot cells is via dedicated ductwork or from ambient air drawn from the room.*

*Clarify the source of air supply for RCA Ventilation System Zone 1, the rooms/areas that receive air from the supply fans identified in Figure 9a2.1-2, and the air supply source to the irradiation unit and hot cells identified in Figure 9a2.1-1.*

- b) The section states that RCA Ventilation System Zone 3 is supplied by the RCA Ventilation System Zone 2 Supply Air subsystem, is exhausted to RCA Ventilation System Zone 2, and is maintained at a higher pressure than RCA Ventilation System Zone 2. However, this PSAR section provides no details on how this is accomplished and Figure 9a2.1-2 has the following annotations, which may or may not be associated with RCA Ventilation System Zone 3: an arrow after the supply fans is labeled - “Supply Air Flows To Additional Rooms,” but does not identify what rooms/air receive the air; an arrow to the exhaust fans states - “Exhaust Flows From Additional Zone 2 Rooms,” which infers preclusion of any air from RCA Ventilation System Zone 3; and at the two Zone 3 airlocks an “Offset Airflow” from Zone 3 to Zone 2, which may not be sufficient total exhaust airflow for RCA Ventilation System Zone 3.*

*Provide additional information on the exhaust and pressure maintenance for RCA Ventilation System Zones 2 and 3, as well as figures, including an RVZ3 flow diagram.*

## **SHINE Response**

- a) As described in Subsection 9a2.1.1 of the PSAR, the RCA Ventilation Zone 1 (RVZ1) hot cell and IU cell enclosures identified in Figure 9a2.1-1 draw ventilation air from the surrounding RCA Ventilation Zone 2 (RVZ2) spaces through HEPA filters. There is no direct RCA Zone 2 Supply Air (RVZ2SA) to the hot cell and IU cell enclosures. The pressure in RVZ1 is negative with respect to RVZ2. The air transferred from the surrounding RVZ2 spaces supply the following RVZ1 enclosures:

- IU cells
- Hot cells (i.e., supercells, uranium extraction (UREX), thermal denitration, solid waste packaging, pump transfer, waste evaporation, and liquid waste solidification hot cell)
- Noble gas storage cell
- TOGS shielded cell

The RVZ2SA air handling units identified in Figure 9a2.1-2 of the PSAR directly supply air to the following rooms/areas of the SHINE facility:

- Zone 2 Hot Cell Operating and Service Access Area
  - Uranyl Preparation and Storage Rooms
  - U.S. Food and Drug Administration (FDA) and Hot Laboratories
  - Decon Room
  - Health Physics Room
  - Tool Crib
  - RPF Airlocks
  - RCA Exhaust Filter Room
  - IF
  - IF Airlock
- b) The RCA Ventilation System (RV) maintains the RCA at the desired temperature and humidity levels and negative pressurization during normal operation. The systems have design features to control overall RCA pressure, RVZ1 exhaust header pressure, and RVZ2 exhaust header pressure, keeping them constant during normal operation. Local room differential pressure set points will be obtained by providing an offset between supply and exhaust flow rates to each space. Supply air flows will be controlled through the use of supply fan variable frequency drives (VFDs), flow measuring stations, and airflow control valves. Exhaust airflow will be controlled based on building pressure, exhaust header pressure, and flow demands, using variable flow fans controlled by VFDs.

RVZ2 and RVZ3 supply air flow rates and RCA zone pressures will be controlled under normal operation by the modulation of airflow control valves and exhaust fans as needed to maintain desired set points. RV zone pressures are negative with respect to atmosphere, and cascade in order from least negative (RVZ3) to most negative (RVZ1) relative pressure.

RVZ2 areas receive supply air from RVZ2SA and offset air from RVZ3, exhaust through general room exhausts and fume hood enclosures (where present), and transfer a portion of air to RVZ1 enclosures. The supply system and the exhaust will have airflow control valves, reacting to maintain the design differential pressure and ensuring the zone pressures are negative with respect to atmosphere, and cascade as described above. Fume hoods will be constant volume to minimize fluctuations within the zone.

RVZ3 areas will have airflow control valves on the supply side delivering air balanced at design values. Air supplied to RVZ3 will be transferred to RVZ2 spaces through engineered leakage pathways and exhausted through RVZ2 systems. SHINE does not anticipate the need for an RVZ3 exhaust system. RVZ3 areas include the RCA airlocks and additional areas which will be defined during detailed design. SHINE will identify the additional RVZ3 areas and provide an RVZ3 flow diagram in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

Figure 9a2.1-2 of the PSAR shows RVZ3 enclosures in the HVAC flow path, and provides the relationship between RVZ3, RVZ2, and Facility Ventilation Zone 4 (FVZ4).

#### **RAI 9a2.1-2**

*SHINE PSAR, Section 9a2.1.2, "Non-Radiological Area Ventilation System," discusses the Facility Ventilation Zone 4 (FVZ4) system. While the SHINE PSAR states that this is a nonradiological controlled area ventilation system, additional information on the potential for contamination in this area is needed for the NRC staff to determine the adequacy of the FVZ4 ventilation system.*

*Provide additional information on the FVZ4 ventilation system, including information on where the system exhausts, whether there are any radiation detectors on the exhaust, and a FVZ4 flow diagram.*

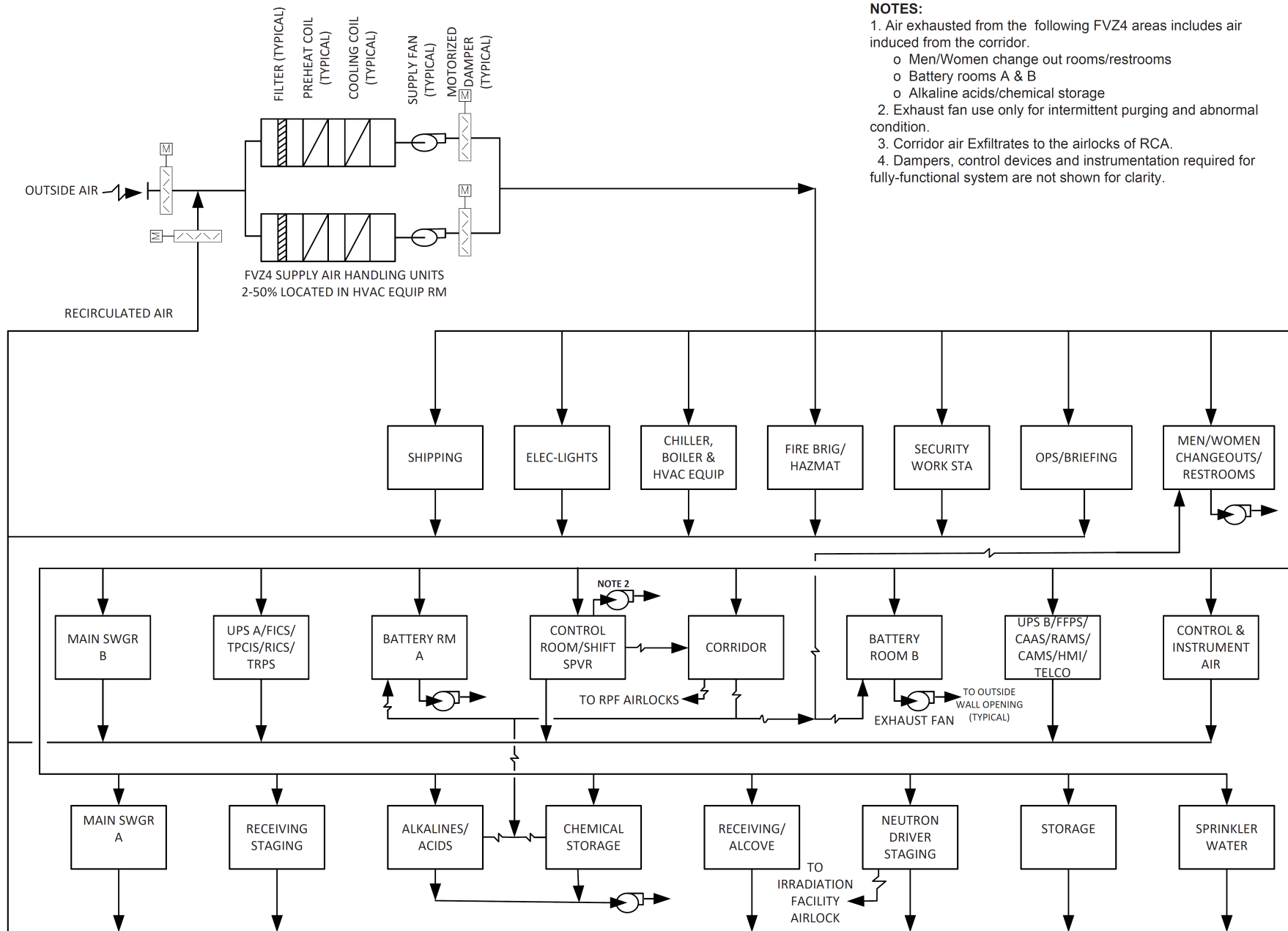
#### **SHINE Response**

FVZ4 consists of areas which are within the SHINE production facility building, but outside of the RCA. FVZ4 is completely independent of the RV. The FVZ4 supply air handling units draw at least 10 percent outside air to make up for air exhausted and exfiltrated. The outside air is mixed with recirculated air and conditioned through the air handling units before being supplied to FVZ4 areas.

FVZ4 exhaust streams exhaust directly to the outside of the SHINE facility. No radiation detectors are provided in the FVZ4 exhaust, as contamination is not expected to be present in FVZ4.

Figure 9a2.1-2-1 provides a flow diagram of FVZ4.

**Figure 9a2.1-2-1: Facility Ventilation Zone 4 Flow Diagram**



**NOTES:**

1. Air exhausted from the following FVZ4 areas includes air induced from the corridor.
  - o Men/Women change out rooms/restrooms
  - o Battery rooms A & B
  - o Alkaline acids/chemical storage
2. Exhaust fan use only for intermittent purging and abnormal condition.
3. Corridor air Exfiltrates to the airlocks of RCA.
4. Dampers, control devices and instrumentation required for fully-functional system are not shown for clarity.



## **Section 9a2.3 – Fire Protection Systems and Programs**

### **RAI 9a2.3-1**

*NUREG-1537, Section 9.3, “Fire Protection Systems and Programs,” states that the application should discuss passive design features required by the facility design characteristics to, in part, limit fire consequences. The facility should be designed and protective systems should exist to prevent the uncontrolled release of radioactive material if a fire should occur.*

*Identify which fire detection and suppression systems are necessary to prevent or mitigate high or intermediate consequence accidents in the RPF (i.e., IROFS), and describe and commit to applying management measures that will assure that these systems and components are constructed, procured, installed, and tested to ensure that they will be available and reliable to perform their intended functions when needed.*

### **SHINE Response**

The SHINE Integrated Safety Analysis (ISA) lists potential accident scenarios and identifies those controls that will prevent the occurrence of and/or mitigate the effects of scenarios with unacceptable consequences. There are no fire detection or suppression systems that are credited with prevention and/or mitigation of potential accident scenarios.

The SHINE Response to RAI 13b.1-1 and RAI 13b.1-2 provide a complete listing of the controls credited with prevention and/or mitigation for the scenarios listed in the ISA. Although there are no credited fire detection or suppression systems, all fire areas of the SHINE facility will have fire detection and/or suppression as described in Section 9a2.3 of the PSAR.

SHINE is no longer using the items relied on for safety (IROFS) or management measures classifications (see Part a of the SHINE Response to RAI 3.5-1 (Reference 4)).

The elements of the fire protection program used at the SHINE facility will be designed, constructed, and maintained in accordance with the applicable National Fire Protection Association (NFPA) standards. Adherence to these standards will ensure that these features will be available to perform their functions when required.

SHINE has added the fire protection program to the list of programs to be included in the Administrative Controls section of the SHINE TS, as provided in Section 14a2.6 of the PSAR. Including the fire protection program in the Administrative Controls section of the SHINE TS will also ensure that the fire protection program elements are available and reliable, by ensuring that testing, surveillance, and maintenance activities are conducted as required. A mark-up of the PSAR changes is provided in Attachment 9. The non-public (proprietary) version of the PSAR, incorporating the changes provided in Attachment 9, is provided in Enclosure 3. The public (non-proprietary) version of the PSAR, incorporating the changes provided in Attachment 9, is provided in Enclosure 4.

In addition, fire protection system SSCs are subject to the SHINE Quality Assurance Program, as described in the SHINE QAPD.

*(Applies to RAIs 9a2.3-2 through 3)*

*NUREG-1537, Part 1, Section 9.3, "Fire Protection Systems and Programs," states, in part, that the applicant should describe systems and programs designed to protect the reactor facility from damage by fire and discuss how the facility meets all local building and fire codes."*

#### **RAI 9a2.3-2**

*SHINE PSAR, Section 9a2.3.4.4, "Safety Evaluation of Fire Hazards," discusses egress from the SHINE facility as in compliance with the International Building Code and Life Safety Code, satisfying the requirements of Title 29 of the Code of Federal Regulations.*

*Fire Area 6 (PSAR Section 9a2.3.4.4.6.7, Figure 9a2.3-1, "Fire Area and Fire Zone Boundaries") of SHINE PSAR, Section 9a2.3, "Fire Protection Systems and Programs," is the corridor in the facility structure that wraps around the north, west, and south sides of the building.*

*A fire in this area could make all egress (except the airlock at the southeast corner of the building) inaccessible.*

*Provide information on how building personnel can evacuate the building under such conditions.*

#### **SHINE Response**

Fire Area (FA)-6 consists of the corridors outside of the airlock and several offices and miscellaneous rooms. A supervised air pre-action fire water suppression system and a fire detection system are provided in this fire area.

As described in Table 9a2.3-1 of the PSAR, the International Building Code (IBC) classification for the SHINE facility is F-1 (Special Industrial). Emergency exits are available from FA-6 within the IBC-allowed maximum travel distance of 250 ft, common path of travel does not exceed the allowed maximum of 100 ft per the IBC, and dead-end corridors do not exceed the allowed maximum of 50 ft per the IBC. These egress distances also satisfy the Life Safety Code (LSC) requirements. Therefore, egress from FA-6 is in compliance with the IBC and LSC.

In addition, while offices and miscellaneous rooms may include low amounts of ordinary combustibles, the connecting corridors are anticipated to be free of any significant combustibles. Activation of the automatic suppression system will ensure any fire in the area remains localized and only segments of the corridors may become inaccessible. Early detection and notification of the fire ensure the personnel will have sufficient time to evacuate the SHINE facility via areas and corridors not affected by the fire.

### **RAI 9a2.3-3**

*Fire Areas (FAs) 1 and 3 utilize gaseous fire suppression systems, as described in SHINE PSAR, Sections 9a2.3.4.4.6.4.3 and 9a2.3.4.4.6.2.3, respectively. Gaseous suppression systems could result in asphyxiation during a release.*

*Describe how potential asphyxiation during a release of the gaseous suppression systems has been addressed in the design of the fire protection system and in the fire protection program in accordance with local building and/or fire codes.*

### **SHINE Response**

In accordance with local building and/or fire codes, automatic gaseous fire suppression systems in use at the SHINE facility will be equipped with a pre-discharge alarm system and a discharge delay to permit personnel egress. In addition, warning signs will be affixed in appropriate locations for areas protected by an automatic gaseous fire suppression system.

*(Applies to RAIs 9a2.3-4 through 5)*

*NUREG-1537, Part 2, Section 9.3, "Fire Protection Systems and Programs," Acceptance Criteria, states, in part, that "[m]ethods to detect, control, and extinguish fires should be stated in the plan."*

### **RAI 9a2.3-4**

*In radiation areas, the smoke detection capability of ionization detectors could be adversely affected. Photoelectric smoke detector capability can be affected in areas of dust/particulates.*

*Provide the basis of choosing detectors, and what maintenance program will be used to assure that the detectors function properly.*

### **SHINE Response**

As stated in Subsection 9a2.3.4.4.2.4 of the PSAR, fire detection systems are provided throughout the SHINE facility, and are designed, installed, located, inspected, tested, and maintained in accordance with NFPA 72 (Reference 29).

SHINE will ensure that appropriate fire detection equipment is placed in radiation areas and areas of dust/particulates.

### **RAI 9a2.3-5**

*The neutron moderation capability of firefighting foam is not discussed. Additional information is needed on the moderation capabilities of firefighting foams because local fire departments may use foam as part of their firefighting repertoire.*

*Provide additional information on foam, if any, that can or will be used in the facility and what training is proposed for the fire brigade and for offsite fire departments that may provide assistance.*

### **SHINE Response**

There are no firefighting foam systems within the RCA in the current design of the SHINE facility.

SHINE has initiated discussions with Rock County Emergency Management. The Rock County 911 Communications Center will have a response information binder specific to emergency response at the SHINE facility. SHINE will ensure that the response information binder specific to emergency response at the SHINE facility does not allow off-site fire support organizations to use firefighting foam within the SHINE RCA. An IMR has been initiated to ensure the Rock County 911 Communications Center's SHINE-specific response information binder provides specific guidance on the use of firefighting foam at the SHINE facility.

Periodic training will be provided to both SHINE fire brigade members and off-site fire support organizations regarding permitted manual fire suppression techniques at the SHINE facility.

### **RAI 9a2.3-6**

*NUREG-1537, Part 2, Section 9.3, "Fire Protection Systems and Programs," Acceptance Criteria, states, in part: "The fire protection plan should discuss the prevention of fires, including limiting the types and quantities of combustible materials."*

*As shown in SHINE PSAR Figure 9a2.3-1, "Fire Area and Fire Zone Boundaries," the Boiler Room (FA-17), which has a natural gas pipeline supplying the boiler, is adjacent (i.e., shares a common wall) to the Fire Brigade/Hazmat Room (FA-16) that contains the Fire Zone Panels.*

*Provide additional information on the potential for a fire in the Boiler Room and address the effects of the pipeline gas combustible load (until the pipeline can be shut off outside the Boiler Room) on the FA-17 and on the rest of the building.*

### **SHINE Response**

The fire protection features for FA-17 and adjacent fire areas are provided in Subsection 9a2.3.4.4.6.18 of the PSAR. The fire area boundaries are identified on Figure 9a2.3-1 of the PSAR. The walls, floors, and ceilings of the FA-17 boundary have a three-hour fire resistive rating as required by a high combustible loading in the room and where an adjacent room contains equipment or systems from a different safety train. The three-hour fire barrier provides adequate time for operators to manually isolate the natural gas supply, if required.

The SHINE gas-fired boiler will comply with applicable State of Wisconsin Statutes and Administrative Codes. The potential release of natural gas into FA-17 will be limited by the installation of safety controls, as required by Wisconsin Administrative Code Chapter SPS 341.

#### **RAI 9a2.3-7**

*NUREG-1537, Part 2, Section 9.3, "Fire Protection Systems and Programs," Acceptance Criteria, states, in part: "The facility should be designed and protective systems should exist to ensure a safe reactor shutdown and prevent the uncontrolled release of radioactive material if a fire should occur."*

*SHINE PSAR, Figure 9a2.3-1 indicates that there are fire zones inside of FA-1 and FA-2. However, the fire zones are not numbered.*

*Provide information indicating whether the fire zones will be numbered, and whether the fire zone numbers will be unique. Additionally, provide information indicating whether the Fire Hazards Analysis will provide assessments of each fire zone.*

#### **SHINE Response**

Each fire zone will be uniquely numbered. As detailed design is completed, the Fire Hazards Analysis (FHA) will be revised and updated. The final FHA will provide an assessment of each fire zone. An IMR has been initiated to ensure that each fire zone is uniquely numbered, and the final FHA provides an assessment of each fire zone.

### **Section 9b.7 – Other Auxiliary Systems**

#### **RAI 9b.7-1**

*NUREG-1537, Part 2, Section 9.7, "Other Auxiliary Systems," Acceptance Criteria, states, in part, that "[t]he design, functions, and potential malfunctions of the auxiliary system should not cause accidents to the reactor or uncontrolled release of radioactivity."*

*SHINE PSAR, Section 9b.7.2, "RCA Material Handling," identifies the equipment used to move or manipulate radioactive material within the RCA and states that "the overhead cranes meet the requirements of ASME B30.2 and CMAA [Crane Manufacturers Association of America] 70."*

*Due to the size and weight of the shields and equipment that need to be moved, and the inventory of tritium and uranium onsite, provide additional assessments demonstrating the implementation of the requirements of ASME B30.2 and CMAA 70 to ensure that dropped, toppled, rolled or otherwise off-normal load events do not result in the loss of safety function or the release of radioactivity to the public.*

#### **SHINE Response**

Implementing the requirements of ASME B30.2 (Reference 30) and Crane Manufacturers Association of America (CMAA) 70 (Reference 31) will ensure that the overhead cranes are designed, built, installed, inspected, operated, and maintained with the high level of quality appropriate for the industrial use of overhead cranes. The use of these standards ensures that cranes whose failure could cause a drop of a heavy load on safety-related SSCs meet the minimum industrial specifications.

Additionally, for cranes located in the vicinity of safety-related SSCs, or where crane accidents or inadvertent operation could impact safety-related SSCs, SHINE will apply the following guidance (see Section 5.1.1 of NUREG-0612 (Reference 32)):

1. Safe load paths will be defined for the movement of heavy loads, or deviations from the defined load paths will require written procedures approved by site safety personnel.
2. Procedures will be developed to cover load handling operations for heavy loads. Procedures will include the identification of required equipment, inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; the defined safe load path; and other special precautions.
3. Crane operators will be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ASME B30.2 (applies to all crane operation).
4. Special lifting devices used in the vicinity of target solution or safety-related SSCs will satisfy the guidelines of ANSI N14.6 (Reference 33).
5. Lifting devices that are not specially designed will be installed and used in accordance with the guidelines of ANSI B30.9 (Reference 34)
6. Tests and inspections will be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency.
7. The crane will be designed to meet applicable criteria and guidelines of ASME B30.2 and CMAA 70 (applies to all cranes).

A heavy load will be defined as a load that, if dropped, may cause radiological consequences that challenge 10 CFR 20 limits. The heavy load limit, and the associated load drop analysis, will be determined during detailed design and provided in the FSAR. An IMR has been initiated to track the inclusion of this information into the FSAR.

If a load contains radioactive material or if safety-related SSCs are beneath or directly adjacent to a potential travel load path of the overhead handling systems (i.e., a path not restricted by limits of crane travel or by mechanical stops or electrical interlocks), such that an off-normal load event could result in the complete loss of a safe shutdown function or the release of radioactivity in excess of 10 CFR 20 limits, one of the following will be satisfied in addition to satisfying the above criteria (see Section 5.1.5 of NUREG-0612):

- a. The crane and associated lifting devices will be single-failure-proof.

OR

- b. If the load drop could impair the operation of equipment or cabling associated with redundant or dual safe shutdown paths, mechanical stops or electrical-interlocks will be provided to prevent the movement of loads in proximity to these redundant or dual safe shutdown equipment. Credit will not be taken for intervening floors unless justified by analysis.

OR

- c. The effects of load drops will be analyzed and the results indicate that any damage would not preclude operation of sufficient safety-related equipment to achieve safe shutdown, provide confinement, or prevent the release of radioactivity in excess of 10 CFR 20 limits.

## **RAI 9b.7-2**

*NUREG-1537, Part 2, Section 9.7, "Other Auxiliary Systems," states that the "design, functions, and potential malfunctions of the auxiliary system should not cause accidents to the reactor or uncontrolled release of radioactivity."*

*In SHINE PSAR, Section 9b.7.2, "RCA Material Handling," statements are made in Sections 9b.7.2.6, 9b.7.2.7, and 9b.7.2.8 that the equipment is designed to prevent inadvertent criticality during material handling. However, no design details are provided. In addition, Section 9b.7.2 does not mention whether there is a need for technical specifications with respect to criticality control during materials handling.*

*Due to the consequences that may result from inadvertent criticality during materials handling, provide additional details on how the equipment will be designed to prevent inadvertent criticality and provide an assessment of why technical specifications are not needed or describe preliminary plans for technical specification safety limits and surveillance requirements.*

## **SHINE Response**

An NCSE will be performed for systems that handle fissile material within the facility. The design for systems that handle fissile material within the facility will comply with the requirements for criticality safety detailed in ANSI/ANS-8.7-1998 (R2007) (Reference 35) and the PSAR. The safety controls identified and associated with the handling devices will be maintained under the configuration management program to ensure the designs and procedures associated with material handling are not changed without appropriate review of the criticality safety function. In addition, maintenance of the material handling devices will ensure that the safety controls are reliable and available when their function is needed.

For SNM outside of the TSV, safety limits are derived for criticality accident prevention based on the nuclear criticality safety program and from the ISA. Limits are specified, using the double-contingency principle, that prevent a criticality accident and are set with a conservative margin. Accordingly, safety limits that are applicable for material handling activities will be included in the TS and Surveillance Requirements (SR) will be developed to ensure criticality control is maintained during material handling activities. The TS will be provided as part of the SHINE OL Application.

## CHAPTER 11 – RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

### **Section 11.1 – Radiation Protection**

#### **RAI 11.1-1**

*10 CFR 50.34(a)(3)(i) requires that preliminary design information provided for the facility include principal design criteria.*

*SHINE PSAR, Section 11.1.1.1, “Airborne Radioactive Sources,” presents information on the management of airborne radioactive sources. It states that predicted personnel dose rates (including maintenance activity) due to airborne radioactivity and associated methodology will be presented in the Final Safety Analysis Report for the SHINE facility.*

*Provide design information in sufficient detail (including key assumptions) to demonstrate the manner in which airborne radioactive material concentrations to which workers may be exposed (especially during maintenance activities) will be controlled in order to meet the derived air concentrations contained in 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage.” Specifically, provide the following:*

- a) The expected airborne radioactive material concentrations (partitioned into noble gases, radioiodines, and particulates) associated with normal operations of the facility compared to their respective derived air concentrations in various areas that could be occupied by workers. Use definitions for airborne radioactivity areas similar to the following in terms of the derived air concentrations: Zone 1 (<0.01 – 1.0 derived air concentration); Zone 2 (1.0 - 10 derived air concentrations); and Zone 3 (>10 derived air concentrations).*
- b) The expected airborne radioactive material concentrations associated with facility accidents compared to their respective derived air concentrations in various areas that could be occupied by workers.*
- c) Key assumptions associated with (a) and (b) above, including:*
  - (i) The basis for the production rate data in PSAR, Table 11.1-9, “TSV [Target Solution Vessel], Noble Gas and Iodine Production Rates, Annual Releases, and ECL [Effluent Concentration Limits] Fraction at the Site Boundary after 960 Hours of NGRS [Noble Gas Removal System] Holdup;*
  - (ii) A description of leakage pathways (including holdup and filtration/adsorption) from the point of production to the point of worker exposure; and*
  - (iii) For the ventilation system: Key parameters and assumptions associated with the estimates of airborne radioactive material concentrations in work areas.*

#### **SHINE Response**

- a) The design of the SHINE facility maintains airborne radioactive material concentrations very low in normally occupied areas. Confinement and ventilation systems are designed to protect workers from sources of airborne radioactivity during normal operation and minimize*



worker exposure during maintenance activities, keeping with the as low as reasonably achievable (ALARA) principles outlined in 10 CFR 20.

SHINE has qualitatively assessed anticipated derived air concentrations for airborne radioactive material (noble gases, radioiodines, and particulates), above grade and within the RCA at the SHINE facility, during normal operations. Figure 11.1-1-1 provides the results of this assessment, using the following definitions:

- Zone 1 ( $< 0.01 - 1.0$  derived air concentration);
- Zone 2 ( $1.0 - 10$  derived air concentrations); and
- Zone 3 ( $> 10$  derived air concentrations).

Partitioning of airborne radioactive material concentrations associated with normal operations into noble gases, radioiodines, and particulates will be provided in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

**Figure 11.1-1-1: Limiting Radioactivity Areas of the SHINE Facility**

Security-Related Information

- b) The expected airborne radioactive material concentrations associated with facility accidents compared to their respective derived air concentrations (DACs) are shown in Table 11.1-1-1. The occupational airborne concentration for each radionuclide was compared to the DAC contained in Appendix B to 10 CFR 20. A sum of fractions method was used to calculate values for noble gases, radioiodines, and particulates for each accident scenario. The DAC values below are for areas that could be occupied by workers and are applicable throughout the RCA of the SHINE production facility building. Evacuation of the workers is to be completed within 10 minutes, as described in Tables 13a2.2.1-2 and 13b.2.1-2 of the PSAR.

**Table 11.1-1-1: Expected Airborne Radioactive Material Concentrations for Workers during Accidents**

Accident Scenario	Concentration/DAC		
	Noble Gases	Radioiodines	Particulates
Target Solution Release Into the IU Cell (Subsection 13a2.2.1)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
Mishandling or Malfunction of Equipment Affecting the PSB (Subsection 13a2.2.7)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
TPS Design Basis Accident (Subsection 13a2.2.12.3)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ] <sup>(a)</sup>
Release of Inventory Stored in NGRS Storage Tanks (Subsection 13b.2.1)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
Loss of Piping or Tank Integrity in the RPF (Subsection 13b.2.4)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
Inadvertent Release from NGRS (Subsection 13b.2.4)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]
RPF Fire (Subsection 13b.2.6)	[ Proprietary Information ]	[ Proprietary Information ]	[ Proprietary Information ]

(a) Tritium gas is included under particulates for the purpose of this table.

c) Basis for the Production Rate Data in Table 11.1-9 of the PSAR

The basis for the production rate data provided in Table 11.1-9 is a single TSV operating at the licensed power limit of [ Proprietary Information ]. Cumulative fission yields were used to account for both instantaneous fission products and decay chain products that would be produced over the course of the entire operational cycle. All fission was assumed to occur by thermal neutrons in U-235. Fission of transmuted Pu-239 and fast neutron fission of U-238 were assumed to be negligible in comparison to U-235 fission rates. A total energy yield for neutron-induced fission of 200 MeV/fission was used. Decay constants for individual isotopes were used to convert the generation rate (atoms/sec) to a fission product activity generation rate (Ci/sec) for each radionuclide. The annual releases provided in Table 11.1-9 assume eight TSVs are in operation.

#### Leakage Pathway to Workers (Normal Operation)

During normal operation, there are no significant anticipated leakage pathways for worker exposure in normally occupied areas to airborne radioactive material. Normally occupied areas within the RCA are serviced by RVZ2 and RVZ3. Positive pressure is maintained in normally occupied areas relative to RVZ1 areas, which potentially contain airborne radioactive materials. See Subsection 9a2.1.1 of the PSAR and the discussion of key parameters of the ventilation system below for additional information on the RV within the SHINE facility.

#### Leakage Pathway to Workers (Accident Analysis)

For the leakage pathway to workers during accident events, excluding the TPS DBA, SHINE assumed that a maximum of 10 percent of the airborne release would escape from the confinement area (e.g., TOGS shielded cell, IU cell, noble gas storage cell, hot cells) penetrations prior to the evacuation. Also, for the releases involving target solution and the release from the TOGS, SHINE assumed that only 25 percent of the available source term is released from the TSV, piping, or TSV dump tank prior to evacuation of the facility ( $0.1 \times 0.25 = 0.025$ ). This assumption is made based on the systems operating near atmospheric pressure, leading to non-energetic releases, the slow pumping rate of the solution from the TSV dump tanks, and the maximum evacuation time for the facility being 10 minutes. For the TPS DBA, no reduction was credited for confinement features for workers, resulting in a leak path factor of 1.0.

#### Key Parameters of the Ventilation System (Normal Operation)

Ventilation within the SHINE facility is a once-through system with cascading ventilation zones designed to protect workers from exposure to airborne radioactive material. RVZ2 areas receive outside air through the RVZ2SA air handling units and exhaust through the RVZ2 exhaust header, which is maintained at a negative pressure relative to outside air. RVZ2 areas also receive a small fraction of air from RVZ3, which also receives its supply air from the RVZ2SA air handling units. Areas serviced by RVZ1 (e.g., IU cells, hot cells) are maintained at a negative pressure in relation to RVZ2 to prevent leakage of airborne radioactive material into normally occupied areas. See Subsection 9a2.1.1 of the PSAR for additional information on the RV.

#### Key Parameters of the Ventilation System (Accident Analysis)

During an accident event in areas of the facility serviced by RVZ1, bubble-tight isolation dampers in the RVZ1 exhaust will close. No components of the RV (e.g., dampers or filters) are credited with reducing worker exposure during accident events.

(Applies to RAIs 11.1-3 through 4)

*10 CFR 20.1101, "Radiation protection programs," Item (b) requires licensees to "...use, to the extent practical, procedures and engineering controls...to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA)."*

*NUREG-1537, Part 2, Section 11.1.3, "ALARA Program" Acceptance Criteria, states, in part: "The highest levels of facility management should be committed to the ALARA program."*

#### **RAI 11.1-4**

*SHINE PSAR, Section 11.1.3, "ALARA Program," states, in part, that the "ALARA concept is also incorporated into the design of the facility. The plant is divided into radiation zones with radiation levels that are consistent with the access requirements for those areas. Areas where on-site personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable." Additional information is needed on the radiation zones for the NRC staff to determine their consistency with ALARA principles.*

*Provide the radiation zone designations based on a consideration of neutron and gamma dose rates for locations that could be occupied, as well as the unrestricted environment as referenced in SHINE PSAR, Section 4a2.5.4. Use definitions for radiation zones similar to the following: Zone 1 (background – 2 millirem/hour); Zone 2 (2 - 100 millirem/hour); and Zone 3 (>100 millirem/hour). Using the preceding radiation zone definitions (or equivalent), provide a tabulation of radiation zone designations that could be occupied by radiation workers, even on a transient basis. Also include doses resulting from anticipated operational occurrences and accidents.*

#### **SHINE Response**

SHINE will use the following radiation area designations, as defined in 10 CFR 20, including consideration for neutron and gamma dose rates:

- Unrestricted area means an area to which access is neither limited nor controlled by SHINE. This would be the area beyond the site boundary.
- Radiation areas (RAs) are those accessible areas where the dose rate is greater than 5 millirem per hour (mrem/hr) at 30 centimeters from the radiation source or from any surface that the radiation penetrates.
- High radiation areas (HRAs) are those accessible areas where the dose rate is greater than 100 mrem/hr at 30 centimeters from the radiation source or from any surface that the radiation penetrates.
- Very high radiation areas (VHRAs) are those accessible areas where the dose rate is greater than 500 rads per hour at 1 meter from the radiation source or 1 meter from any surface that the radiation penetrates.

The SHINE facility will be designed and constructed so that the measureable dose rate in the normally occupied and unrestricted area due to activities at the plant will be less than the limits of 10 CFR 20.1301(a)(2). As stated in Subsection 11.1.1 of the PSAR, the goal for the normal operations dose rate for normally occupied locations in the facility is 0.25 mrem/hr at the surface, and SHINE facility shielding is designed to meet this goal.

Procedures for transient access to the IU cells, hot cells, and other shielded vaults; cells; and rooms will ensure doses are maintained ALARA by addressing the following:

- job planning;
- radiation protection coverage;
- survey techniques and frequencies;
- training of workers;
- pre-work briefing;
- frequency for updating radiation work permits or their equivalent; and
- placement of measuring and alarming dosimeters.

The IU cells, hot cells, and other shielded vaults; cells; and rooms designated as HRAs or VHRAs will not normally be occupied when those conditions exist. Administrative procedures will address the management oversight and specific control measures needed for entry into HRAs and VHRAs, if it is ever necessary to do so. The procedures will include the process for gaining entry to these areas, such as the control and distribution of keys.

Typical transient access to the IU cells, hot cells, and other shielded vaults; cells; and rooms that are usually HRAs or VHRAs for maintenance or other necessary work will normally be performed after dose rates have been reduced to at least the level of a radiation area. This will be done by removing the radioactive materials or changing the conditions (such as shutting down the accelerator in an IU cell), and waiting some time period for decay.

A tabulation of normally and transient-occupied areas, dose rates, and designations is provided in Table 11.1-4-1.

**Table 11.1-4-1: Radiation Areas at the SHINE Facility**

<b>Area</b>	<b>Dose Rate</b>	<b>Designation</b>
Normally occupied areas within the RCA	$\leq 5$ mrem/hr	Normally occupied area
IU cells, hot cells, and other shielded vaults; cells; and rooms – material not present or accelerator not in operation, after sufficient decay period	$> 5$ mrem/hr but $\leq 100$ mrem/hr	RA (transient occupation)
IU cells, hot cells, and other shielded vaults; cells; and rooms – material present or accelerator in operation or shutdown without sufficient decay period	$> 100$ mrem/hr (HRA) $> 500$ rad/hr (VHRA)	HRA or VHRA (rarely occupied, per ALARA controls)

Figure 11.1-4-1 provides the probable radiation area designations, above grade, within the RCA at the SHINE facility.

**Figure 11.1-4-1: Probable Radiation Area Designations Within the SHINE RCA**

Security-Related Information

(Applies to RAIs 11.1-6 through 8)

*NUREG-1537, Part 2, Section 11.1.7, "Environmental Monitoring," Acceptance Criteria, states, in part, that "[t]he methods and techniques to sample and analyze the radiological effect of facility operation should be complete, applicable, and of sufficient validity that the environmental impact can be unambiguously assessed."*

*SHINE's proposed radiological environmental monitoring program for plant operation is provided in SHINE PSAR, Section 11.1.7, "Environmental Monitoring." Additional information is needed for the NRC staff to determine the adequacy of several elements of the proposed operational radiological environmental monitoring program.*

#### **RAI 11.1-6**

*SHINE PSAR, Section 11.1.7.2.2.1, "Air Sampling Locations," discusses the proposed air monitoring program. When discussing the equipment that will be used for air sampling, the applicant uses the term CAM (continuous air monitor). The conventional use of the term "continuous air monitor" denotes equipment that both samples and quantifies the activity on the sample media (i.e., real-time monitoring). Normally, CAMs are not used for such purposes and the NRC guidance document, NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," that the applicant cites, does not specify CAMs for environmental air sampling.*

*Clarify whether the term "air monitoring" is intended to refer to sample collection followed by laboratory analysis or real-time air monitoring.*

#### **SHINE Response**

SHINE inappropriately used the term "CAM" to refer to a continuous air sampler in Subsection 11.1.7.2.2.1 of the PSAR. SHINE will employ continuous air samplers at the stated locations, with the samples collected and analyzed in a laboratory. SHINE will correct the terminology in Subsection 11.1.7.2.2.1, Table 11.1-8, and Figure 11.1-3 of the PSAR in the FSAR. An IMR has been initiated to track the correction Subsection 11.1.7.2.2.1, Table 11.1-8, and Figure 11.1-3.



## Section 11.2 – Radioactive Waste Management

*(Applies to RAIs 11.2-1 through 2)*

*In accordance with 10 CFR Part 20, an as low as reasonably achievable (ALARA) program is required and 10 CFR 50.34(a)(3)(i) requires that preliminary design information provided for the facility include principal design criteria.*

*NUREG-1537, Part 2, Section 11.2.1, “Radioactive Waste Management Program,” Acceptance Criteria, states, in part, “the program should be designed to address all technical and administrative functions necessary to limit radiation hazards related to radioactive waste.”*

### **RAI 11.2-1**

*The Mo-99 extraction columns are a frequent (400 target solution volumes per year) and initially highly radioactive solid waste generated by the proposed SHINE facility. As a supplement to material presented in SHINE PSAR, Section 9b.7.2, “RCA Material Handling,” and Section 11.2, “Radioactive Waste Management,” additional information is needed on criteria for the handling of this waste stream and the handling of the extraction column in and from the supercells to the shielded vaults and further to packaging and shipping for disposal. This information is needed for the NRC staff to ascertain safety, as well as SHINE’s ability to meet the regulatory requirement regarding hazardous material identification in shipping papers (10 CFR 20, Subpart K, “Waste Disposal”), and conformance with ALARA goals.*

*Provide the following information so that staff may assess compliance with the ALARA requirement of 10 CFR Part 20:*

- a) Describe the inlet and outlet connections of the Mo-99 columns that permit frequent remote replacement while providing leak-tightness and preventing the spread of contamination during replacement. Provide the estimated dose rate from an extraction column at time of removal and after 2 weeks storage in the supercell.*
- b) Provide information on the material handling methods of moving shielded containers of an extraction column from the supercell to the shielded vaults at the other end of the facility from the supercells. If this material handling includes movement by crane, include a load drop in the accident analyses or justify why such an event need not be considered.*
- c) Clarify how long extraction columns are maintained in shielded vault storage. SHINE PSAR, Table 11.2-3, “Waste Methodology for Columns,” says approximately 400 days of decay are required to be Class A; PSAR, Section 9b.7.5.4.2, “Solid Radioactive Waste Handling Hot Cell,” says they are transferred to the storage vault for an additional 6 months.*
- d) Provide information on the transfer of an extraction column into one of the six separate shielded storage vaults shown on figures presented in SHINE PSAR, Chapter 1, “The Facility.”*
- e) Clarify, whether there are any differences between the handling of the Mo-99 columns.*

### **SHINE Response**

- a) The inlet and outlet connectors for the extraction columns and [ Proprietary Information ] are a quick-disconnect style specifically designed for use with remote manipulators in hot cell environments. Radiation and wear-resistant seals provide leak tightness. Automatic valves built into the connectors close when the connector is separated, which significantly reduces the leakage of contaminated fluids from the process lines and columns. As the target solution is washed from the extraction column (by an acid wash, water wash, and strip solution) prior to the column being removed, there will be significantly lower levels of contamination present in the column and lines during disconnection than during extraction steps. Means will be provided to capture the small leakage of liquid that will occur during the disconnection of the connectors.

The estimated dose rate for an extraction column at the time of removal is approximately 6,700 rem/hr at 3 feet unshielded. The peak dose rate drops to approximately 340 rem/hr at 3 feet unshielded after 2 weeks of storage in the supercell.

- b) The material handling methods involved in the transfer of an extraction column from the supercell to the shielded vaults are described in Part d of the SHINE Response to this RAI (Reference 4). SHINE will ensure safe load handling guidance is applied to the movement, as described in the SHINE Response to RAI 9b.7-1.
- c) Following storage in the supercell for no less than two weeks after use, the extraction columns will be maintained in shielded vault storage for approximately six months. The columns will then be transferred to the solid radioactive waste handling hot cell for packaging in approved containers. The containers will be transferred to the Waste Staging and Shipping Building for storage until they reach the requirements for Class A waste and can be shipped for disposal. Preliminary calculations indicate that a total decay time of approximately 400 days will be needed for the columns to reach the requirements for Class A waste. At least two weeks of this time will be spent in the supercell, approximately six months in shielded vaults, and the remaining time (approximately 200 days) will be spent in the Waste Staging and Shipping Building.
- d) See Reference 4 for Part d of the SHINE Response to RAI 11.2-1.
- e) See Reference 4 for Part e of the SHINE Response to RAI 11.2-1.

### **RAI 11.2-2**

*SHINE PSAR, Section 4b.4.1.1.4.1, "Uranyl Nitrate Preparation Process Sequence," explains part of the process for reusing target solution and states that the solid salts discharged from the centrifuge are moved to solid radioactive waste packaging in a 55-gallon drum. PSAR, Section 11.2.2.2.6, "Target Solution Clean-up," identifies that this waste stream is Class B. There is no discussion of the radiation levels emanating from these drums, no discussion of sealing the drums during handling to prevent spills, and no discussion of design features implemented to assure doses to workers are ALARA during these evolutions. PSAR, Table 11.2-6, "Waste Methodology for [ ]" (the rest of the table name is withheld as proprietary information) identifies that the waste stream must be sampled for waste characterization prior to solidification, but there is no discussion of how this is accomplished in an ALARA manner.*

*Provide discussion of the design features and design review procedures used to assure that the ALARA considerations committed to in SHINE PSAR, Section 11.1.3, "ALARA Program," are effectively implemented for each of the identified waste streams and the handling operations required during their processing.*

### **SHINE Response**

Subsection 11.1.2.1 of the PSAR discusses SHINE's commitment to the implementation of an ALARA program. The design features of the Uranyl Nitrate Preparation (UNP) waste streams, while preliminary, are provided below.

The UNP subsystem process is used to convert uranyl sulfate from the spent target solution into uranyl nitrate. A step in this process involves pumping the reaction product slurry from the uranyl nitrate conversion tank to the uranyl nitrate centrifuge. The centrifuge provides separation of precipitated sulfates from the uranyl nitrate solution. The sulfate sludge is transferred at the centrifuge discharge to the Solid Radioactive Waste Packaging (SRWP) system. The preliminary concept for transfer of the sulfate sludge to the SRWP system involves moving the sulfate sludge (sulfate solid concentration of 50 percent by weight) from the centrifuge discharge to shielded 55-gallon drums. The centrifuge discharge and the drum loading area will be inside a shielded area of the facility. The loaded drums will be sealed inside the shielded area in order to prevent spills during handling and remotely moved to the waste evaporation hot cell. Following evaporation, the waste will be moved to the liquid waste solidification hot cell for further processing, remote sampling for waste characterization, and surface decontamination activities and packaging. All process steps will be performed in accordance with written procedures which incorporate ALARA practices. The operators will be trained in the procedures and ALARA features. Storage of the processed waste for the final decay time and shipment consolidation will be in the Waste Staging and Shipping Building. SHINE will more fully specify waste handling for the UNP subsystem with respect to design features used to assure that ALARA considerations are effectively implemented during detailed design, and provide those details in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

While the SRWP system design has not been finalized, the design of the radiation shielding for tanks, walls, hot cells, and the drums will ensure the dose rates are within the regulatory limits of 10 CFR 20.1201 and 10 CFR 20.1301, and that the dose rates in normally occupied areas meet (to the extent practicable) the facility goal of 0.25 mrem/hr. These design features will also ensure that ALARA objectives to minimize exposure to facility workers during waste handling and transfer operations are met. Radiation monitors will be located at operations areas to detect and warn of changes in radiation levels.

A design review of the final SRWP system design will be performed to ensure that transfer and processing of the sulfate sludge will be done in an ALARA manner. The design review will be proceduralized, and these procedures will evaluate general design considerations and methods to maintain radiation exposures ALARA consistent with the recommendations of Regulatory Guide 8.8 (Reference 36). The ALARA design considerations have two objectives:

- Minimize the necessity for access to and personnel time spent in radiation areas; and
- Minimize radiation levels in routinely occupied areas in the vicinity of equipment expected to require personnel attention.

In order to meet these objectives, the ALARA design review of the SRWP system and associated interfaces will consider the following:

- Facility arrangement
  - Facility layout and access control
  - Equipment layout
  - Component layout
- Radiation shields and geometry
  - Bulk shielding
  - Penetration and partial shielding
  - Entryway shielding
- System design
  - Decontamination provisions
  - Redundancy, remote operation and instrumentation
  - Radiation monitoring system
- Pipe and valve design
  - Pipe routing and design
  - Valve location and design
- Materials and equipment design features
  - Specifications
  - Heat exchangers
  - Centrifuges
  - Tanks
  - Instruments
- Facility services
  - Lighting and communications
  - Contamination control and coatings
  - Access platforms
  - Airborne radioactivity and HVAC system
  - Sampling and monitoring
- Human factors
  - Visual aids
  - Auditory aids

#### **RAI 11.2-4**

*SHINE PSAR, Table 11.2-1 presents estimates of waste generation rates and waste classification without sufficient discussion or quantitative values to assess the reasonableness of the estimates presented. For example:*

- The total for all the liquid radioactive waste inputs is presented to five significant figures (59,708 gallons per year) but only one liquid waste stream has an estimated generation rate associated with it in the text (scrubber solution at 20,000 gallons per year).*
- Coolant cleanup system spent ion exchange resins are not included in PSAR Table 11.2-1. A commitment to include this value in the FSAR exists in the text.*
- There is insufficient chemical characterization data of the individual waste streams to allow assessment of the potential for unexpected chemical reactions or to estimate volumes of acids or bases that may be needed for pH adjustment.*
- There is no identification of any anticipated upset or accident condition that could cause an input to the liquid waste processing system.*

*Provide a comprehensive liquid waste process flow diagram showing expected liquid waste generation rates (with chemical and radiological properties) for all liquid waste streams, washes, rinses, and chemical additions that flow to the consolidated radioactive liquid waste tanks. The process flow diagram should also quantify tank capacities and processing flow rates that demonstrate the capability to process wastes from normal operations and anticipated upset conditions with margin, or identify locations for interfacing with temporary mobile systems, as needed. The process flow diagram should include an estimate of the area needed for decay in storage of packaged waste and the criteria used to determine shielding requirements.*

#### **SHINE Response**

##### **Bullet 1**

*The total for all the liquid radioactive waste inputs is presented to five significant figures (59,708 gallons per year) but only one liquid waste stream has an estimated generation rate associated with it in the text (scrubber solution at 20,000 gallons per year).*

#### **SHINE Response**

SHINE will revise Table 11.2-1 of the PSAR in the FSAR to show an influent volume of approximately 52,000 gallons per year of liquid waste. The waste volumes provided in the last six rows of Table 11.2-1 will be revised as follows:

Description	Matrix	Class as Generated	As Generated Amount	As Generated Units	As Shipped (cubic feet(ft <sup>3</sup> ))	Ship Type	Destination
Spent Washes	Liquid <sup>(a)</sup>	A	2100	gallons/yr	14,000 <sup>(b)</sup>	LSA	Energy Solutions
Rotary Evaporator Condensate	Liquid <sup>(a)</sup>	A	200				
UREX Raffinate	Liquid <sup>(a)</sup>	B	27,000				
NO <sub>x</sub> Scrubber Solution	Liquid <sup>(a)</sup>	A	20,000				
Decontamination Waste	Liquid <sup>(a)</sup>	A	400				
Spent Eluate Solution	Liquid <sup>(a)</sup>	A	2600				

- a) This liquid waste discharged from the various processes at the SHINE facility is either solidified and then shipped to a waste depository or reused.
- b) As shipped volume of waste is in the form of concrete. Total liquid volume of approximately 52,000 gallons/yr is reduced via evaporation to approximately 35,000 gal/yr (volume reduction factor of 1.5). The liquid waste is then solidified by adding Portland cement (or equivalent). Concrete volume is estimated using a conservative waste to cement ratio of 0.5.

An IMR has been initiated to track the revision to Table 11.2-5.

## Bullet 2

*Coolant cleanup system spent ion exchange resins are not included in PSAR Table 11.2-1. A commitment to include this value in the FSAR exists in the text.*

## SHINE Response

SHINE will revise Table 11.2-1 of the PSAR in the FSAR to include an estimate of spent ion exchange resins from the coolant cleanup systems (LWPS and PCLS). A preliminary estimate for this value is 48 ft<sup>3</sup>/year of spent resin, Class A as generated, Shipment Type LSA. This estimate assumes that each demineralizer has a resin bed volume of three ft<sup>3</sup>, and that conservatively, one bed is exhausted per year per coolant cleanup system due to the very low expected amount of ionic impurities in a closed, clean water system.

A final waste estimate will be developed during detailed design. An IMR has been initiated to track the revision to Table 11.2-1, which will include a final estimate of spent ion exchange resins from the coolant cleanup systems.

### Bullet 3

*There is insufficient chemical characterization data of the individual waste streams to allow assessment of the potential for unexpected chemical reactions or to estimate volumes of acids or bases that may be needed for pH adjustment.*

### SHINE Response

Table 9b.7-7 of the PSAR provides the waste stream chemical characterizations. To be consistent with the revision to Table 11.2-1 described above, SHINE will revise the waste volumes provided in Table 9b.7-7 in the FSAR, as follows:

Description	Matrix	Class as Generated	Contents	Volume	Volume as Shipped (ft <sup>3</sup> )	55-gallon drum equivalent as shipped	Ship Type	Destination
Spent Washes	Liquid <sup>(a)</sup>	A	[ Proprietary Information ]	2100 gallons/yr				
Rotovar Condensate	Liquid <sup>(a)</sup>	A	[ Proprietary Information ]	200 gallons/yr				
UREX Raffinate	Liquid <sup>(a)</sup>	B	[ Proprietary Information ]	27,000 gallons/yr				
Decontamination Waste	Liquid <sup>(a)</sup>	A	Decon fluid unknown	400 gallons/yr	14,000 <sup>(b)</sup>	2100 <sup>(c)</sup>	LSA	Energy Solutions
Spent Eluate Solution	Liquid <sup>(a)</sup>	A	[ Proprietary Information ]	2600 gallons/yr				
NO <sub>x</sub> Scrubber Solution	Liquid <sup>(a)</sup>	A	[ Proprietary Information ]	20,000 gallons/yr				

- This liquid waste discharged from the various processes at the SHINE facility is either solidified and then shipped to a waste depository or reused.
- As shipped volume of waste is in the form of concrete. Total liquid volume of approximately 52,000 gallons/yr is reduced via evaporation to approximately 35,000 gal/yr (volume reduction factor of 1.5). The liquid waste is then solidified by adding Portland cement (or equivalent). Concrete volume is estimated using a conservative waste to cement ratio of 0.5.
- A 55-gallon drum, filled to 90 percent to account for minor voiding, has a volume of approximately 6.6 ft<sup>3</sup>.

An IMR has been initiated to track the revision to Table 9b.7-7.

### Bullet 4

*There is no identification of any anticipated upset or accident condition that could cause an input to the liquid waste processing system.*

### SHINE Response

Margin is included in the design of the liquid waste processing equipment to account for anticipated upset or accident conditions by the inclusion of two liquid waste storage tanks. The liquid waste processing equipment will be sized to allow processing of one tank while the other is filling in order to maintain buffer storage capacity. Anticipated upset conditions may include

the production of extra decontamination waste used to clean up small spills, extra waste from UNCS if a portion of it needs to be repeated for a specific batch (i.e., contactor failure leads to inadequate separation), extra column wash solution due to a failure of the wash delivery system, or contaminated fire protection water caused by a sprinkler failure. Accident conditions that may produce additional liquid waste are described in Subsections 13a2.1.4 and 13b.2.4 of the PSAR.

Paragraph (Bullet 5)

*Provide a comprehensive liquid waste process flow diagram showing expected liquid waste generation rates (with chemical and radiological properties) for all liquid waste streams, washes, rinses, and chemical additions that flow to the consolidated radioactive liquid waste tanks. The process flow diagram should also quantify tank capacities and processing flow rates that demonstrate the capability to process wastes from normal operations and anticipated upset conditions with margin, or identify locations for interfacing with temporary mobile systems, as needed. The process flow diagram should include an estimate of the area needed for decay in storage of packaged waste and the criteria used to determine shielding requirements.*

SHINE Response

A liquid process flow diagram is provided as Figure 9b.7-5 of the PSAR. Figure 11.2-4-1, below, provides a modified version of Figure 9b.7-5 showing expected liquid waste generation rates and chemical and radiological properties for the liquid waste streams, washes, rinses, and chemical additions that flow to the liquid waste storage tanks (1-RLWS-01TA/B). Figure 11.2-4-1 also shows tank capacities, processing flow rates, a location for interfacing with temporary mobile systems if needed, and an estimate of the area needed for decay in storage. SHINE will update Figure 9b.7-5 of the PSAR in the FSAR to incorporate the process details described above. An IMR has been initiated to track the update to Figure 9b.7-5 in the FSAR.

Table 9b.7-7 of the PSAR provides the waste stream chemical characterizations and the expected liquid waste generation rates. As described above, Table 9b.7-7 will be revised in the FSAR to be consistent with the waste generation rates in the corrected Table 11.2-1. An IMR has been initiated to track the revision to Table 9b.7-7.

The radiological properties of the streams are described qualitatively as follows, and summarized in Table 11.2-4-1 below.

Irradiated target solution is passed through the Molybdenum Extraction and Purification System (MEPS) extraction column and then processed for re-irradiation. The large majority of fission products stay with this stream, to be removed as waste into the UREX raffinate stream after [ Proprietary Information ] cycles of irradiation. Therefore, the UREX raffinate stream is the largest contributor to the radiological inventory of the consolidated liquid waste, comparable to the radiological properties of irradiated target solution after the removal of uranium, molybdenum, and small fractions of other elements that are expected to be retained on the extraction column. The estimated radioactive inventory from one TSV batch is [ Proprietary Information ], as stated in Subsection 4b.4.1.2.1.2 of the PSAR.

The spent washes consist of potassium permanganate and water washes of the extraction column after the target solution has been passed through the column. Between 0 percent and 30 percent of each element in the target solution stream is expected to be partitioned to the wash streams, with the partitioning of most elements being less than approximately 5 percent.



Therefore, the estimated radioactive inventory is expected to be less than 30 percent of the radioactive inventory of one TSV.

Spent eluate solution from the [ Proprietary Information ] and rotary evaporate condensate are both waste streams created in process steps downstream of the extraction column. As described above, the majority of fission products are partitioned into earlier process waste streams, and as the volume of these two streams is much lower than other streams (see corrected Table 9b.7-7 above), the radiological inventory is also expected to be low in comparison to the waste generated in upstream processes.

Decontamination waste will have variable radiological properties, but is expected to have a low radiological inventory based on the low volume of generated waste and the low amount of leakage or spillage that will require cleanup and decontamination.

NO<sub>x</sub> scrubber solution is also expected to have a low radiological inventory. The PVVS uses a caustic solution to remove acid gases and byproduct material. The byproduct material consists of vent gases from process tanks and from the NGRS after at least 40 days of decay, where the majority of this byproduct material, specifically short-lived isotopes of noble gases, has decayed to stable isotopes.

Volumes for Cintichem liquid waste and NGRS condensate are both expected to be less than one gallon per TSV batch, and are therefore not significant inputs and are not included in Table 9b.7-7 of the PSAR or Table 11.2-4-1, below. The generation rates, chemical properties, and radiological properties of these two streams will be determined during detailed design and provided in the FSAR. An IMR has been initiated to track the inclusion of this information in the FSAR.

**Table 11.2-4-1: Radiological Properties of Liquid Waste Streams, Washes, Rinses, and Chemical Additions**

<b>Description</b>	<b>Radiological Inventory (Relative to Other Liquid Waste Streams)</b>	<b>Qualitative Radiological Properties</b>
Spent Washes	Medium	Most fission products pass through the extraction column, though some are expected to be retained on the column and then removed with column washes.
Rotary Evaporate Condensate	Low	Fission products remaining after majority removed in prior MEPS processing steps.
UREX Raffinate	High	The majority of non-volatile fission products are removed from target solution during UREX processing. The estimated radioactive inventory from one TSV batch is [ Proprietary Information ].
NOx Scrubber Solution	Low	Byproduct material from process tank vents and NGRS gases after 40 days of decay.
Decontamination Waste	Low	Dependent on decontamination needs
Spent Eluate Solution	Low	Fission products remaining after majority removed in prior MEPS processing steps.
Chemical Additions	None	Non-radioactive chemicals may be added as needed for pH adjustment.

Subsection 9b.7.4.2.3 of the PSAR contains a description of the radioactive liquid waste storage system. This section describes the two liquid waste storage tanks, configured in parallel, each with a working volume of approximately [ Security-Related Information ]. This provides around 40 days of buffer storage capacity of liquid wastes produced by normal facility operations. The buffer storage capacity will provide margin for anticipated upset conditions. Anticipated upset conditions may include the production of extra decontamination waste used to clean up small spills, extra waste from UNCS if a portion of it needs to be repeated for a specific batch (i.e., contactor failure leads to inadequate separation), extra column wash solution due to a failure of the wash delivery system, or contaminated fire protection water caused by a sprinkler head failure. Accident conditions that may produce additional liquid waste are described in Subsections 13a2.1.4 and 13b.2.4 of the PSAR.

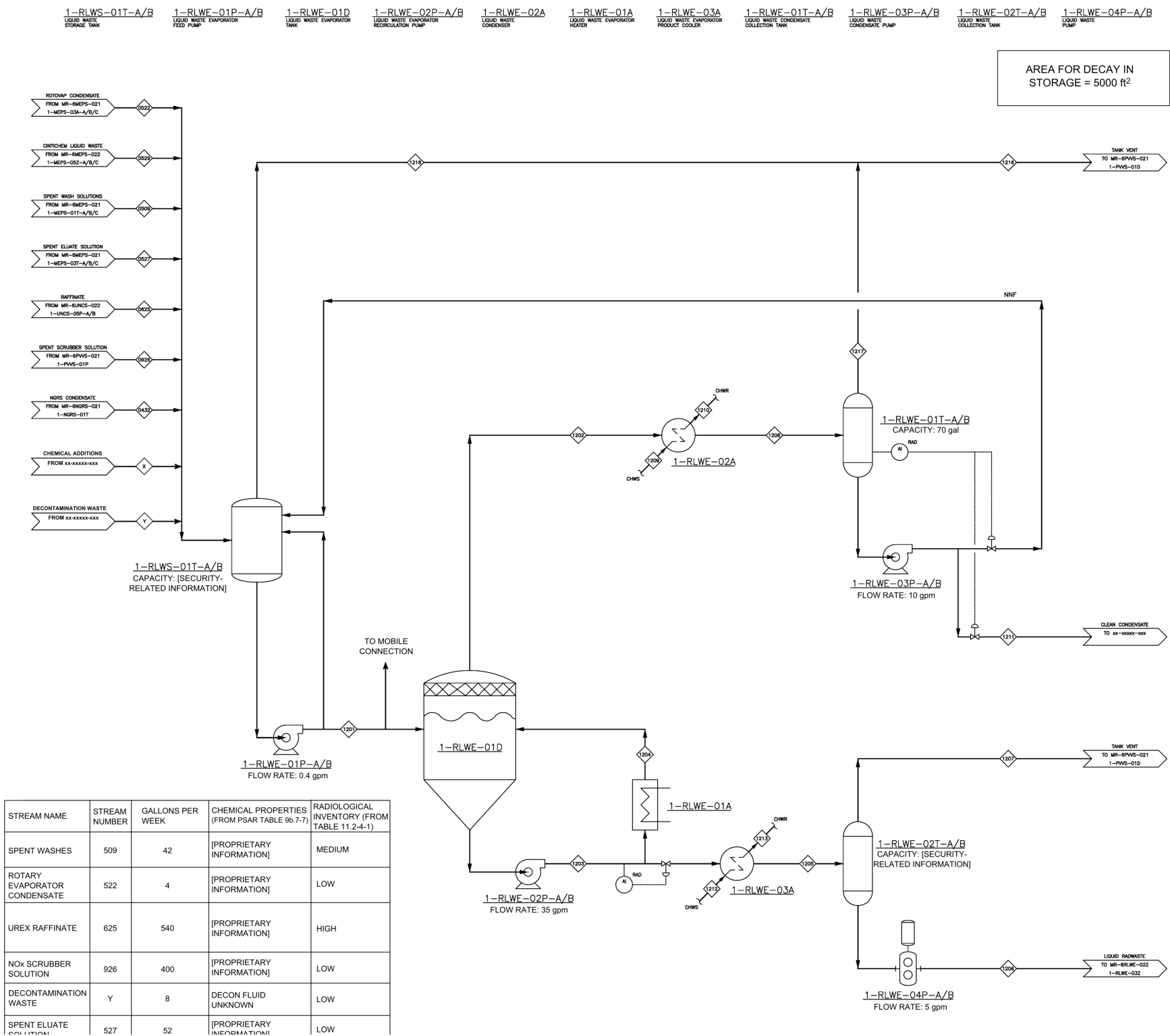
Processing flow rate into the liquid waste storage tanks is expected to be approximately 1040 gallons per week, based on a 50-week year (see the SHINE response to RAI 11.2-3 (Reference 4)). The equipment downstream of the liquid waste storage tanks will be specified in detailed design, and will be sized to allow processing of one tank while the other is filling in order to maintain the buffer storage capacity. Preliminary equipment specifications are provided in Table 9b.7-3 of the PSAR, and included in Figure 11.2-4-1, below. Liquid waste is expected to be processed in batches, so the flow rates provided in Figure 11.2-4-1 will only be applicable when the equipment is in operation. An IMR has been initiated to ensure the equipment downstream of the liquid waste storage tanks is specified in detailed design, and is sized to allow processing of one tank while the other is filling in order to maintain the buffer storage capacity.

The area used for decay in storage of packaged waste is the Waste Staging and Shipping Building. This building is identified in Figure 1.3-4 of the PSAR, where it has approximate dimensions of 50 ft by 100 ft. Therefore, the estimated total area needed for decay in storage of packaged waste is approximately 5000 ft<sup>2</sup>. The exact dimensions of the Waste Staging and Shipping Building will be determined during detailed design and provided in the FSAR. An IMR has been initiated to ensure the exact dimensions of the Waste Staging and Shipping Building are determined during detailed design and provided in the FSAR.

The criteria for shielding requirements of packaged grouted waste in the Waste Staging and Shipping Building will be selected to ensure the site dose limits of 10 CFR 20 are met. An IMR has been initiated to ensure the final design of the Waste Staging and Shipping Building incorporates criteria for shielding requirements such that the site dose limits of 10 CFR 20 are met.

The criteria will follow ALARA principles to manage radiation exposure in the Waste Staging and Shipping Building to ensure that occupational and off-site doses are maintained ALARA.

Figure 11.2-4-1: RLWE Liquid Waste Evaporator and Storage System Process Flow Diagram



### **RAI 11.2-5**

*NUREG-1537, Part 2, Section 11.2.2, "Radioactive Waste Controls," Evaluation Findings, states, in part: "The descriptions of the plans and procedures provide reasonable assurance that radioactive wastes will be controlled at all times in a manner that protects the environment and the health and safety of the facility staff and public."*

*Disposal sites have established waste acceptance criteria, as identified in PSAR, Chapter 11.*

*The inputs to the consolidated radioactive liquid waste tanks are a mixture of strong acids and bases, chemicals in solution, and water, all containing fission products. This chemical mixture is then concentrated through evaporation to reduce the volume of waste to be solidified for packaging and disposal.*

*Provide references that support the validity of the assumption that the evaporator concentrates of the consolidated liquid waste stream can be solidified on Portland cement to meet the waste acceptance criteria of the potential disposal sites. Alternatively, commit to conducting a solidification testing program during construction of the facility to be able to define the requirements of the solidification Process Control Program in the PSAR.*

### **SHINE Response**

SHINE plans to adjust the pH of the consolidated radioactive liquid waste prior to evaporation and solidification using Portland cement as the solidification agent.

The Electric Power Research Institute (EPRI) has addressed the issue of cement solidification in Topical Report NP-2900 (Reference 37). Table F-1 of EPRI NP-2900 presents the type and amount of solidification agent to be used with various types of waste streams found in the nuclear power industry. The table specifically addresses the use of Portland cement as a solidification agent and addresses solidification of sodium sulfate waste streams which typically are generated by the neutralization of sodium hydroxide with sulfuric acid prior to solidification.

Appendix D of EPRI NP-2900 discusses leach rate tests of solidified waste. Appendix D provides significant data on leach rate tests of cement solidified sodium sulfate waste streams. Figure 1 of Appendix D indicates that if the weight percent of sodium sulfate in the solidified waste is eight weight percent or less, then leaching is in compliance with federal regulations, even for samples immersed in water.

The SHINE facility design is for waste streams with low concentrations of solid species that are below eight weight percent, as shown in the modified Table 9b.7-7 provided in the SHINE Response to RAI 11.2-4. Even after a volume reduction factor of 1.5, the contents of species other than water are not expected to exceed eight weight percent. Therefore, SHINE anticipates that the solidification process with cement will be successful and meet the acceptance criteria of potential waste disposal sites. The guidance included in EPRI NP-2900 will be employed in detailed design and in developing facility operating procedures for pre-treatment and cement solidification of liquid radwaste.

The operating procedures will implement the requirements of the facility solidification Process Control Program (PCP). The PCP will ensure that the solidified product will have no detectable free liquid, consistent with nuclear power industry guidance contained in Branch Technical Position 11-3 (Reference 38). During the SHINE facility commissioning, solidification demonstration test runs will be performed using non-licensed materials.

## CHAPTER 12 – CONDUCT OF OPERATIONS

### **Section 12.7 – Emergency Planning**

#### **RAI 12.7-12**

*NUREG-0849, Section 6.0, “Emergency Planning Zones” (EPZs), Evaluation Items 1. and 2., states the emergency plan should identify the EPZ and, “if the EPZ is not consistent with Appendix II, the plan shall include an acceptable basis for the EPZ.”*

*The SHINE Preliminary Emergency Plan, Revision 0, Section 6.0, “Emergency Planning Zones,” addresses SHINE’s implementation of ANSI/ANS-15.16-2008 and NUREG-0849 related to the identification of an EPZ at the SHINE facility. ANSI/ANS-15.16 and NUREG-0849 support an EPZ size of the “operations boundary,” “100 meters,” “400 meters,” “800 meters,” or a size “determined on a case-by-case basis.” SHINE’s proposed EPZ is not consistent with ANSI/ANS-15.16 or NUREG-0849.*

*Identify in the emergency plan, the EPZ size for the SHINE facility. If the EPZ is not consistent with ANSI/ANS-15.16-2008 or NUREG-0849, include an acceptable basis for the EPZ size selected or explain why an EPZ is not necessary.*

#### **SHINE Response**

Section 6.0 of NUREG-0849 (Reference 39) states the following:

*“As part of emergency planning, the reactor owner/operator of a facility that identifies radiological emergencies which result in offsite plume exposures exceeding 1 rem whole body or 5 rem thyroid should identify an emergency planning zone (EPZ). The postulated radioactive release from credible accidents provides the basis for determining the need for an EPZ. The size of the EPZ should be established so that the dose to individuals beyond the EPZ is not projected to exceed the PAG. As an alternative to performing such calculations, the EPZ sizes in Appendix II may be adopted according to the power level.”*

The Foreword of ANSI/ANS-15.16-2008 (Reference 40) states the following, in parts:

*“In addition, the regulations require the determination of the need for establishing an off-site emergency planning zone (EPZ) on a case-by-case basis and require the identification of predetermined actions for protecting individuals within this zone.”*

and

*“The risk from credible radiological emergency situations at many research reactor facilities is usually minimal and may not require the application of all the emergency plan elements in this standard or the establishment of an EPZ.”*

Section 3.6 of ANSI/ANS-15.16-2008 states the following:

*“As part of emergency planning, the reactor licensee or owner/operator of a facility that identifies radiological emergencies that result in off-site plume exposures exceeding 10 mSv deep dose (1 rem whole body) or 50 mSv (5 rem thyroid) shall identify an EPZ.”*

Section 3.6.1 of ANSI/ANS-15.16-2008 states the following:

*“The postulated radioactive releases from credible accidents provide the basis for determining the need for an EPZ. The EPZ size depends on the distance at which the protective actions are calculated to be warranted. As an alternative to performing such calculations, the EPZ sizes in Table 2 may be adopted according to the power level. Table 2 is based upon highly conservative dose calculations that are generically applicable to research reactors.”*

SHINE has performed calculations as part of the accident analysis, and the calculation results are described in Chapter 13 of the PSAR. SHINE has concluded that the radiological releases and consequences to workers and the public are maintained within 10 CFR 20 limits (i.e., 100 millirem to the general public and five rem to the worker). For this reason, Section 6.0 of the SHINE Preliminary Emergency Plan (Reference 20) states:

*“As described in Chapter 13 of the SHINE PSAR (Reference 11.8), no credible accident scenarios at the SHINE facility result in radiological emergencies that involve an off-site plume exposure exceeding 1 rem whole body or 5 rem thyroid. Therefore, in accordance with ANSI/ANS-15.16-2008 (Reference 11.1) and NUREG-0849 (Reference 11.3), no EPZs have been identified for the SHINE facility.”*

Based on the above information, SHINE concludes that an emergency planning zone (EPZ) is not required at the SHINE facility, and the guidance and information contained in ANSI/ANS-15.16-2008 and NUREG-0849 align with this position. There is no radiological emergency which would result in off-site plume exposures exceeding one rem whole body or five rem thyroid.



## CHAPTER 13 – ACCIDENT ANALYSIS

### **General Information Request**

#### **RAI 13a2-G**

*As required by 10 CFR 50.34(a)(4), a “preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.”*

*Many of the accident analyses in Chapter 13 make assumptions about the source term and release fractions through barriers, based on the design characteristics of the various systems, structures and components in the system.*

*For example, SHINE PSAR, Section 13a2.2.1.4, “Quantitative Evaluation of Accident Evolution,” states, in part: “The total release to the RCA through the IU cell penetrations during the accident is assumed to be no more than 10 percent of the airborne activity in the IU cell based on design characteristics of the penetrations.”*

- a) Discuss whether these release fractions are design specifications that the facility is being designed to.*
- b) Provide information stating whether all of these assumptions are being tracked so that the design will account for all of the assumptions.*
- c) Discuss how these release fractions will be verified in the as constructed facility.*
- d) Describe whether there will be periodic testing over the facility’s lifetime to ensure that the assumptions are still valid, as is done with periodic containment leakage testing in operating reactors.*
- e) Provide a discussion of how the design of the facility provides assurance that the assumed release fractions are bounding values, as compared to actual releases that would result in an accident scenario.*

#### **SHINE Response**

- a) The SHINE facility will be designed and built to meet or be more conservative than the release fractions assumed in the accident analysis.*
- b) Accident analysis assumptions will be accounted for in the design of the SHINE facility. An IMR has been initiated to track that accident analysis assumptions are accounted for in the final design of the SHINE facility.*

- c) As described in Section 3.5 of the PSAR, SSCs that are determined to have safety significance are tested commensurate with the criteria set forth in ANSI/ANS-15.8-1995 (R2013) (Reference 41) as implemented by the SHINE Quality Assurance Program Description (QAPD). This testing will include vendor testing, receipt inspection and testing, and as-constructed local and overall leak rate testing to verify the design release fractions in the as-constructed SHINE facility. An IMR has been issued to ensure release fractions are verified in the as constructed facility.
- d) As described in Section 3.5 of the PSAR, periodic testing of SSCs that are determined to have safety significance is performed commensurate with the criteria set forth in ANSI/ANS-15.8-1995 (R2013), as implemented by the SHINE QAPD. Specific surveillance testing requirements for periodic testing to ensure the assumptions are still valid will be placed in the SHINE TS, which will be submitted as part of the SHINE OL Application.
- e) As required by NUREG-1537 (References 27 and 42) and the Interim Staff Guidance (ISG) augmenting NUREG-1537 (References 43 and 44), SHINE postulated and evaluated an MHA. The MHA includes a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible. Because the MHA includes a non-mechanistic failure assumed to establish an outer limit consequence, the scenario is considered not credible. The SHINE accident analysis includes two MHAs, one for the IF and one for the RPF.

The SHINE evaluations of the consequences of the MHAs are described in Subsections 13a2.2.1 and 13b.2.1 of the PSAR. In these evaluations, the consequences of the events were shown to meet the SHINE radiological release limits for accidents. Because the consequences of these non-credible events were shown to meet the radiological release limits through the use of safety-related engineering controls, it is assumed that potential actual credible accidents will also meet the radiological consequences. This is consistent with the accident analysis philosophy of NUREG-1537.

The SHINE facility will be designed and built to meet or be more conservative than the release fractions assumed in the accident analysis. The assumptions in the accident analysis form part of the design basis for the SHINE facility. This information will be used for the final design of the SHINE facility, and will ensure that the assumptions are accounted for in the design. As described in Section 3.5 of the PSAR, the initial as-constructed and periodic testing of SSCs that are determined to have safety significance is performed in accordance with the criteria set forth in ANSI/ANS-15.8-1995 (R2013), as implemented by the SHINE QAPD. Specific surveillance testing requirements for periodic testing to ensure the assumptions are still valid will be placed in the SHINE TS, which will be submitted as part of the SHINE OL Application. These measures provide assurance that the facility design release fractions are built and tested to bound the actual releases that would result from an accident scenario.

## Section 13a2.1 – Accident-Initiating Events and Scenarios

(Applies to RAIs 13a2.1-2 through 8 and RAIs 13a2.2-1 through 3)

*The ISG Augmenting NUREG-1537, Part 2, Section 13a2, “Aqueous Homogeneous Reactor Accident Analyses,” states that the applicant should include a systematic analysis and discussion of credible accidents for determining the limiting event in each category and that the mathematical models and analytical methods employed, including assumptions, approximations, validation, and uncertainties, should be clearly stated.*

### **RAI 13a2.1-3**

*SHINE PSAR, Section 13a2.1.2.1, “Identification of Causes, Initial Conditions, and Assumptions,” discusses the insertion of excess reactivity. Since the system is over-moderated, decreasing the density of the coolant or introducing voids in the primary closed loop cooling system (PCLS) would result in a positive reactivity insertion.*

*Provide additional information discussing whether the situation of decreasing the density of the coolant or introducing voids in the PCLS has been analyzed as a possible accident scenario. Provide the reactivity worth of changing the density of the coolant from nominal operating conditions to fully voided conditions. Compare that reactivity worth to the margin of criticality in the system.*

### **SHINE Response**

SHINE performed a calculation of the expected reactivity changes due to voiding out the PCLS from nominal coolant temperature and density to a fully voided cooling system. Voids were assumed to occur in the cooling channels around the TSV [ Proprietary Information ]. The calculation was performed at cold (Mode 1) startup conditions and hot (Mode 2) irradiation conditions. Results of the calculation are provided in Table 13a2.1-3-1

**Table 13a2.1-3-1: Reactivity Changes Due to PCLS Voids**

Percent Void	Delta Reactivity (pcm)	
	Reactivity Change for Mode 1 Startup Conditions	Reactivity Change for Mode 2 Irradiation Conditions
20	[ Proprietary Information ]	[ Proprietary Information ]
40	[ Proprietary Information ]	[ Proprietary Information ]
60	[ Proprietary Information ]	[ Proprietary Information ]
80	[ Proprietary Information ]	[ Proprietary Information ]
100	[ Proprietary Information ]	[ Proprietary Information ]

The results provided in Table 13a2.1-3-1 show that for the PCLS, there is a negative insertion of reactivity as the percent void increases. The change in reactivity was calculated by subtracting the reactivity of PCLS void case from the reactivity of the base case (without PCLS voids).

The highest  $k_{\text{eff}}$  for both Mode 1 and Mode 2 conditions occurred when there were no voids present in the PCLS. While the target solution itself is over-moderated, the cooling system acts as a reflector in some locations and enhances the overall neutron economy.

Since the reactivity change is negative, the reactivity changes provided in Table 13a2.1-3-1 will increase the margin to criticality in the system.

An analysis of voiding in the LWPS cooling loop is provided in the SHINE Response to RAI 4a2.6-4.

#### **RAI 13a2.1-4**

*SHINE PSAR, Section 13a2.1.2.2.3, “Moderator Addition Due to Cooling System Malfunction,” discusses the addition of moderator due to a cooling system malfunction.*

*Provide additional information discussing whether a TOGS condenser heat exchanger (HX) failure or recombiner HX failure and water ingress has been considered as a possible accident scenario.*

#### **SHINE Response**

The TOGS condenser heat exchanger failure and recombiner heat exchanger failure and water ingress has been considered as a possible accident scenario. Like the cooling system malfunction described in Subsection 13a2.1.2.2.3 of the PSAR, a heat exchanger failure will result in a dilution of the TSV solution and a rise the TSV solution level. A TSV dilution event due to water ingress, regardless of the source, results in a significant negative reactivity insertion because of the corresponding decrease in uranyl sulfate density with an increase in the TSV water volume.

Calculations performed with a two cm and four cm rise in the TSV solution level due to water ingress showed a reactivity decrease of approximately [ Proprietary Information ], respectively. Eventually, additional water ingress to the TSV results in the excess solution flowing into the overflow lines that drain to the TSV dump tank.

### **RAI 13a2.1-5**

*SHINE PSAR, Section 13a2.1.2.2.1, “Increase in the Target Solution Density During Operations,” discusses increases in the target solution density during operations and concludes that “this event causes a positive reactivity addition, but not large enough to reach a critical condition...” However, additional information is needed to demonstrate that the system will not become critical.*

*Provide the expected reactivity insertion, following the maximum credible deflagration. The void fraction due to radiolytic decomposition will seldom, if ever, be zero, so it seems possible that the over-pressurization resulting from a deflagration could result in a  $k_{eff}$  greater than that occurring during cold startup, since the concentration of the solution is greater than what it is during startup.*

*Describe the approach used to determine this maximum  $k_{eff}$  value.*

### **SHINE Response**

SHINE calculated the expected reactivity insertion from the maximum credible deflagration to be approximately [ Proprietary Information ] using the calculation approach outlined below. The increase in reactivity due to water loss was found to be approximately [ Proprietary Information ]. The initial reactivity at hot operating conditions (i.e., prior to water loss or deflagration) was calculated to be approximately [ Proprietary Information ]. The increase in solution concentration due to water loss combined with the maximum deflagration event was not found to result in  $k_{eff}$  greater than that occurring during cold startup. The final  $k_{eff}$  was found to be approximately equal to [ Proprietary Information ].

The calculation was performed using MCNP5, version 1.60, and the SHINE best-estimate neutronics model.

Increased pressure in the TSV due to a maximum deflagration event in the TOGS would cause the target solution to be compressed as void space decreases. This would cause an increase in reactivity in the SHINE system. To determine the increase in reactivity prior to the maximum deflagration event, water hold-up from the TSV into the TOGS system is evaluated.

The following is a list of assumptions that were used to analyze the multiplication factor resulting from deflagration:

1. The maximum water hold-up in TOGS is one percent of the total TSV volume.
2. The bulk void fraction of the TSV during irradiation was assumed to be three percent. As described below, this assumption affects  $k_{eff}$  during normal irradiation, but does not significantly affect the post-deflagration  $k_{eff}$ .
3. The normal operating pressure within the TSV is 14.7 psia.
4. This calculation conservatively assumed that the increase in pressure due to a maximum deflagration event was 200 psi. This is expected to bound any credible deflagration.
5. The target solution in the TSV and in the dump lines has the same uranium concentration.
6. Gases within the target solution behave as ideal gases.
7. The number of moles of gas in the target solution is a constant just before and just after the deflagration.
8. The temperature of the target solution is a constant just before and just after the deflagration event.

The change in the concentration of uranium in the target solution due to the removal of one percent of water was calculated.

New void fractions following deflagration were calculated using the ideal gas law, assuming constant mass and temperature.

The results show that in the event of a maximum deflagration, with water holdup in TOGS accounted for, the  $k_{\text{eff}}$  values do not exceed those at cold zero power (CZP).

The analysis was also repeated at one percent initial bulk void fraction. The lower initial bulk void fraction results in higher reactivity during irradiation, but less increase due to deflagration. These effects compensate and the final condition (collapsed void with water loss) is nearly independent of the amount of initial bulk void fraction. The final  $k_{\text{eff}}$  after the maximum credible deflagration with one percent initial bulk void fraction was also found to be approximately [ Proprietary Information ].

### **RAI 13a2.1-6**

*“Scenario C- Loss of or Reduced PCLS and LWPS [light water pool system] Flow” is the most limiting of the reduction of cooling events, as described in SHINE PSAR, Section 13a2.1.3.1, “Identification of Causes, Initial Conditions, and Assumptions.” It is described as a low probability event not expected to occur during the facility lifetime.*

*Provide the technical basis for this claim.*

### **SHINE Response**

The loss of or reduced PCLS and LWPS flow coupled with continued operation of the neutron driver is a low probability event not expected to occur during the facility lifetime because it involves multiple unlikely simultaneous failures. If a loss of off-site power (LOOP) occurs, the neutron driver will be de-energized, as it is not powered by emergency power supplies. Therefore, the loss or reduced flow of both cooling systems with continued operation of the neutron driver would have to be caused by an event other than a LOOP. As described in Subsection 7a2.4.1 of the PSAR, a loss or reduction of PCLS flow below the trip setpoint initiates a TRPS trip, which opens the TSV dump valves and de-energizes the neutron driver. The TRPS trip is single-failure proof and would require both redundant safety-related channels to fail simultaneously for the neutron driver to continue operating upon PCLS flow below the trip setpoint.

SHINE has concluded that this low probability event is not expected to occur during the facility lifetime.

### **RAI 13a2.1-7**

*SHINE PSAR, Section 13a2.1.8.2, “General Scenario Description,” provides a general scenario description of potential power oscillations. However additional information is needed for the staff to verify that the system will not become critical.*

*Provide the expected magnitude of potential power oscillations, and a description of the mechanisms that are in place to ensure that they are “self-limiting.”*

### **SHINE Response**

Oscillations in TSV fission power are expected to be the result of coupled system oscillations, principally due to potential pressure variations in the TOGS and source oscillations from the neutron driver. Pressure variations in the TOGS at the TSV are expected to be less than 0.5 psi, and this pressure change has been estimated to affect TSV reactivity during irradiation by less than [ Proprietary Information ]. This change in reactivity is estimated to result in approximately a [ Proprietary Information ] change in TSV power.

Oscillations in the neutron driver neutron output are expected to be less than three percent from the target neutron output. Oscillations in neutron driver output do not result in direct changes to  $k_{eff}$ . With a subcritical assembly and no feedback effects, a change of three percent in the neutron source term would result in a three percent change in the output of the assembly. Due to negative temperature and void reactivity coefficients, the oscillations from this source variation will be less than three percent. An increase in accelerator output of three percent was estimated to result in a reactivity change of approximately [ Proprietary Information ] due to void and temperature feedback.

Total oscillations in TSV fission power are expected to be less than [ Proprietary Information ] due to potential superposition of these sources of oscillation. The system will remain subcritical as the reactivity changes from these oscillations are significantly less than the subcritical reactivity margin in Mode 2 conditions. As described in Subsection 4a2.6.1.2 of the PSAR, results of transient modeling of power oscillations using a dynamic model and reactivity feedback effects will be provided with the FSAR.

The negative temperature and void coefficients of the target solution in the subcritical assembly are the mechanisms that ensure that power oscillations are self-limiting. As described in the linear stability analysis of subcritical accelerator-driven fissile solution systems, provided as Attachment 8, these systems were found to be unconditionally stable in the linear approximation.

### **RAI 13a2.1-8**

*SHINE PSAR, Section 13a2.1.1.1, “Initial Conditions and Assumptions,” states that “the postulated [maximum hypothetical accident] MHA in the [irradiation facility] IF is a large rupture of the TSV dump tank resulting in a complete release of the target solution and fission product inventory into one IU cell.”*

*SHINE PSAR, Section 13a2.1.1.2, “General Scenario Description,” states that “the IF postulated MHA general scenario is a release of irradiated target solution to the IU cell as a result of a loss of TSV integrity.”*

*SHINE PSAR, Section 13a2.2.1.1, "Initiating Event," states "the target solution release in the IF is postulated to be a large rupture of the TSV and SASS resulting in a complete release of the target solution and fission product inventory into one IU cell."*

*Based on the statements above, additional information is needed for NRC staff to determine the adequacy of SHINE's evaluation of the maximum hypothetical accident in the irradiation facility.*

*Provide information indicating whether the maximum hypothetical accident in the irradiation facility is a result of a large rupture of the TSV or TSV dump tank.*

### **SHINE Response**

The IF postulated MHA is a release of irradiated target solution to the IU cell as a result of a loss of TSV integrity. It is assumed that the PSB and the Subcritical Assembly Support Structure (SASS) have breached. The presence of the light water pool is ignored for the purposes of the IF postulated MHA. The loss of TSV integrity encompasses either a TSV or TSV dump tank rupture.

SHINE will revise Subsection 13a2.1.1.1 of the PSAR in the FSAR to state that the postulated MHA in the IF is the loss of TSV integrity. An IMR has been initiated to track the revision to Subsection 13a2.1.1.1 in the FSAR.

## **Section 13a2.2 – Accident Analysis and Determination of Consequences**

### **RAI 13a2.2-1**

*SHINE PSAR, Section 13a2.2.1.5, "Radiation Source Term Analysis," lists the factors used to calculate the airborne and respirable source terms. The values used in this analysis for these factors are listed in Tables 13a2.2.1-2, 13a2.2.1-3, and 13a2.2.1-4.*

*Provide the technical basis for the quantities provided in Tables 13a2.2.1-2, 13a2.2.1-3, and 13a2.2.1-4 for each specific scenario considered.*

### **SHINE Response**

#### **Technical Basis for the Quantities in Table 13a2.2.2-2 of the PSAR**

The damage ratio was assumed to be 1.0, with all material at risk assumed involved in the event. This is a conservative assumption.

Release Fraction from IU cell (public/worker) is discussed below under the technical basis for the quantities in Table 13a2.2.1-3.

HEPA filters are credited with a particulate removal efficiency of 99 percent. Carbon adsorbers are credited with an iodine removal efficiency of 95 percent. These values will be confirmed through testing during operation of the facility and are consistent with Regulatory Guide 1.52 (Reference 45).

Airborne Release Fractions (ARF) and Respirable Fractions (RF) are discussed below under the technical basis for the quantities in Table 13a2.2.1-4.



Two sets of dose conversion factors (DCFs) were used in the dose consequence assessment. DCFs from ICRP Publication 30 (Reference 46) were used to convert inhaled activity to internal dose, assuming an aerodynamic mean of one  $\mu\text{m}$ . DCFs from Federal Guidance Report No. 12 (Reference 47) were used to convert exposure to an external amount of activity to an external dose.

The breathing rate for workers and off-site individuals of  $3.5\text{E-}04 \text{ m}^3/\text{s}$  was taken from Regulatory Guide 1.195 (Reference 48), which is used for accident analysis for light water power reactors for Control Room Operators and off-site individuals for the first eight hours of an event.

Dispersion values for the facility were calculated using PAVAN, which implements the guidance provided in Regulatory Guide 1.145 (Reference 49). The wind data consisted of joint frequency distribution of speed, direction, and Pasquill stability class obtained from the anemometer at the Southern Wisconsin Regional Airport in Janesville, Wisconsin for the period 2005 to 2010. The release point to the receptors was calculated from a circle that envelopes the corners of the isotope production facility, which is conservative because it minimizes the distance between the release point and any receptor.

The bounding 50<sup>th</sup> percentile dispersion value that was calculated for the site boundary was  $3.88\text{E-}04 \text{ s/m}^3$  based on a two hour time period and the highest X/Q value calculated in any of the 16 direction sectors. The 50<sup>th</sup> percentile dispersion value for the nearest resident was calculated to be  $5.43\text{E-}05 \text{ s/m}^3$ .

The internal gross volume of the RCA of the production facility building was calculated as  $47,061 \text{ m}^3$ . The volume of below grade areas (e.g., tank vaults and trenches) was not included in the volume estimate. It was assumed that 25 percent of the facility volume is occupied by equipment and structures, leaving  $35,296 \text{ m}^3$  as the free volume in the facility.

Ten minutes was used as a conservative assumption for the worker evacuation time. Workers will be trained to immediately evacuate the area in response to a high radiation alarm or Criticality Accident and Alarm System (CAAS) alarm.

#### Technical Basis for the Quantities in Table 13a2.2.2-3 of the PSAR

The following discussion provides the technical basis for the public and worker leak path factors (LPFs) during the following events:

- Target solution release into the IU cell (IF postulated MHA)
- Mishandling and malfunction of equipment affecting the PSB
- Mishandling or malfunction of target solution
- TPS DBA

#### LPF to the Public

For the LPF to the public, the total release to RVZ1 through the bubble-tight isolation dampers is assumed to be no more than one percent of the airborne activity prior to closing. The particulate LPFs to the public are further reduced by the HEPA filters, which are assumed to have an efficiency of 99 percent ( $0.01 \times 0.01 = 0.0001$ ). The halogen LPFs to the public are further reduced by the carbon adsorbers, which are assumed to have an efficiency of 95 percent ( $0.01 \times 0.05 = 0.0005$ ). There are no additional reductions for noble gases or tritium gas released.

### LPF to the Worker

For events other than the TPS DBA, with releases occurring inside of confinement area (i.e., IU cell or TOGS cell), SHINE assumed that prior to evacuation of the facility, a maximum of 10 percent of the airborne release would escape through the cell penetrations and that only 25 percent of the source term is released from the TSV, piping, or TSV dump tank ( $0.1 \times 0.25 = 0.025$ ). This assumption is made based on the systems operating near atmospheric pressure, leading to non-energetic releases, the slow pumping rate of the solution from the dump tanks, and the maximum evacuation time for the facility being 10 minutes.

The SHINE Response to RAI 6a2.2-11 provides a discussion of the changes to the PSAR for the TPS DBA. For the TPS DBA, the LPF for workers is 1.0. No dose reductions due to the LPF were credited for workers.

### Technical Basis for the Quantities in Table 13a2.2.1-4 of the PSAR

The ARF and RF values for particulates were taken from Table 3-1 of NUREG/CR-6410 (Reference 50). For the target solution release from the IU cell, values for a free-fall spill of aqueous liquids (Event 3.3.4.2 of NUREG/CR-6410) were used. For the mishandling and malfunction of target solution, values for liquid confined in a vessel or container with a rapid buildup of pressure, vented below the surface level of liquid (Event 3.3.1.9(b) of NUREG/CR-6410) were used. The ARF for halogens in all scenarios was taken from Table 3-18 of NUREG/CR-6410, and an RF value for halogens of 1.0 was chosen as a conservative value.

Conservative ARF and RF values of 1.0 were used for noble gases in the target solution release into the IU cell and mishandling and malfunction of equipment scenarios. For the mishandling and malfunction of target solution scenario, an estimate was made of the solubility of noble gases within the target solution, which indicated that normal operations gas release fraction for the dominant fission product noble gases would be below 10 percent. An ARF of 0.1 was selected as a bounding value. For the TPS DBA, conservative ARF and RF values of 1.0 were used. SHINE has added the ARF and RF values for the TPS DBA to Table 13a2.2.1-4 (see the SHINE Response to RAI 13b.1-1).

### **RAI 13a2.2-3**

*SHINE PSAR, Figure 9a2.1-1, “RVZ1 Ventilation Flow Diagram,” shows a HEPA filter located in the outlet duct of the irradiation unit ventilation system. While a description of this filter is also provided in SHINE PSAR, Section 9a2.1.1, there is no mention of this filter in PSAR Chapter 13, “Accident Analysis.”*

*Additional information is needed on this filter for NRC staff to determine the adequacy of SHINE’s accident analysis.*

*Provide information indicating whether decontamination credit has been given to the filter identified above in the accident analysis.*

### **SHINE Response**

The HEPA filter located in the IU cell outlet filter housing, shown in Figure 9a2.1-1 of the PSAR, is not credited for providing any particulate filtration capability in the SHINE accident analysis. Any airborne activity that may occur within RVZ1 is filtered through the two trains of HEPA filters and carbon adsorber beds located in the RVZ1 exhaust final outlet filter housings, as described in Subsection 13a2.1.1.1 of the PSAR.

### **RAI 13a2.2-4**

*SHINE PSAR, Section 13a2.2.1.6, “Radiological Consequence Analysis,” provides a high-level description of the SHINE radiological dose consequence analysis.*

*Additional information is needed for NRC staff to determine the adequacy of SHINE’s radiological dose consequence analysis as part of its accident analysis.*

*Provide information on dose calculations, including a description of the methods and codes used, important input parameter values, and calculated values of dose components (e.g., inhalation dose, immersion dose, ground contamination dose, etc.) Also, provide information describing any important assumptions made while performing dose calculations.*

### **SHINE Response**

A bounding source term was created for the irradiation of the TSV uranyl sulfate solution to determine the material at risk (MAR) term for normal operations and fault events. MCNP5, version 1.60, the Los Alamos National Laboratory (LANL) MCNP radiation transport code was used to calculate fluxes and cross-sections specific to various TSV configurations. The COUPLE and ORIGEN-S codes, both modules of the larger SCALE-6.1 (Standardized Computer Analyses for Licensing Evaluations) computational system from Oak Ridge National Laboratory (ORNL), were used to generate radionuclide inventory source term data.

Bounding parameter values, as described in Table 11.1-1 of the PSAR, were used to determine the material at risk for the radiological consequence analysis. These values included the nominal U-235 enrichment of 19.75 weight percent, maximum concentration of U-234 and U-236, a bounding TSV power of [ Proprietary Information ] ([ Proprietary Information ] license limit plus 10 percent), a 5.5 day irradiation period, [ Proprietary Information ] decay in dump tank following irradiation, [ Proprietary Information ], 80 percent extraction of [ Proprietary Information ], an extraction time in the supercell of [ Proprietary Information ], and

[ Proprietary Information ] of decay prior to the next cycle. Using the minimum anticipated extraction efficiencies for the [ Proprietary Information ] isotopes maximizes their carryover to subsequent cycles. The source term values are taken at shutdown, following the [ Proprietary Information ] cycle.

The analyses resulted in radionuclide inventories or source term(s) for various stages of target solution irradiation that were then applied to subsequent analysis for analyzing DBAs, evaluating the adequacy of radiation protection shielding, and for generating the associated waste streams.

#### Important Assumptions for Radiological Dose Calculations

1. The SHINE facility is designed and constructed to withstand any and all external events that were evaluated.
2. Bubble-type isolation dampers close upon receipt of a high radiation signal from the radiation monitoring system to minimize releases.
3. TSV dump tank transfer pump and system isolation valves trip or close upon receipt of a high radiation signal from the radiation monitoring system to minimize releases.
4. Only one mechanical failure is considered per design basis event.
5. Zeolite holdup of iodine in the TOGS is 95 percent.
6. HEPA filter efficiency for removing particulates is 99 percent.
7. Iodine removal efficiency of carbon adsorber is 95 percent.
8. Members of the public are exposed for the duration of the release.
9. Under normal operating conditions no workers are in the IU cell.
10. Radionuclides are distributed instantaneously and homogeneously throughout the location of the accident.
11. Potential doses to workers because of decontamination activities were not analyzed.
12. One fourth of the building volume is occupied by equipment.
13. Workers require 10 minutes to identify the accident and evacuate the area.
14. Due to short duration of worker exposure, ventilation does not affect the radionuclide concentration in the RCA.
15. Due to the short duration of the exposure, radioactive decay following the event is negligible to the calculated dose.
16. The combustible loading is limited, such that a fire would only affect a single tank.

#### Dose Calculation Methodology

The total effective dose equivalent (TEDE) to workers and members of the public is calculated by determining the radiological dose due to internal and external radiation. The radionuclides deposited in the body produce an internal dose known as the committed effective dose equivalent (CEDE). The external dose equivalent (EDE) is due to radionuclides in the atmosphere that irradiate individuals, which consists of immersion and exposure from a contaminated surface. These are summed for the TEDE as shown in Equation 13a2.2-4-1.

$$H_k[rem] = C_k[rem] + E_{Ik}[rem] + E_{Sk}[rem] \quad \text{Equation 13a2.2-4-1}$$

where:

$H_k$	$\equiv$	TEDE of group k
$C_k$	$\equiv$	CEDE of group k
$E_{Ik}$	$\equiv$	EDE due to immersion of group k
$E_{Sk}$	$\equiv$	EDE due to surface contamination of group k

## Internal Dose

The internal dose is due to radionuclides that are inhaled in the body. Equation 13a2.2-4-2 and Equation 13a2.2-4-3 show this mathematically for workers and members of the public, respectively.

$$C_O[rem] = \frac{\sum ST_R(i)[Ci] \times DCF(i) \left[ \frac{rem}{Ci} \right] \times BR \left[ \frac{m^3}{s} \right] \times t[s]}{V[m^3]} \quad \text{Equation 13a2.2-4-2}$$

where:

$C_O$	$\equiv$	Occupational CEDE
$ST_R(i)$	$\equiv$	Respirable source term for radionuclide i
$DCF(i)$	$\equiv$	Internal dose conversion factor for radionuclide i
$BR$	$\equiv$	Breathing rate
$V$	$\equiv$	Dispersion volume
$t$	$\equiv$	Exposure time

$$C_P[rem] = \sum \left( ST_R(i)[Ci] \times DCF(i) \left[ \frac{rem}{Ci} \right] \right) \times BR \left[ \frac{m^3}{s} \right] \times \frac{\chi}{Q} \left[ \frac{s}{m^3} \right] \quad \text{Equation 13a2.2-4-3}$$

where:

$C_P$	$\equiv$	Public CEDE
$\chi/Q$	$\equiv$	Dispersion value

The respirable source term for radionuclide i [ $ST_R(i)$ ] is the amount of a radionuclide i at the origin of the accident, which becomes airborne respirable due to the physical stress acting on the material. It is calculated using Equation 13a2.2-4-4.

$$ST_R(i)[Ci] = MAR(i)[Ci] \times DR \times LPF \times ARF(i) \times RF(i) \quad \text{Equation 13a2.2-4-4}$$

where:

$MAR(i)$	$\equiv$	Material at risk for radionuclide i
$DR$	$\equiv$	Damage ratio
$LPF$	$\equiv$	Leakpath factor for radionuclide i
$ARF$	$\equiv$	Airborne release fraction for radionuclide i
$RF$	$\equiv$	Respirable fraction for radionuclide i

The factors used to calculate the source term are the MAR, the DR, the LPF, the ARF, and the RF. The DR is the fraction of the MAR actually impacted by the stress. The LPF is the fraction of the radionuclide made airborne that challenges the interface of the facility and ambient environment. The ARF is the coefficient used to estimate the amount of radioactive material that can be suspended in the atmosphere as an aerosol and made available for airborne transport under the physical stresses from a specific accident. The RF is the fraction of airborne radionuclide particles/aerosols that can be transported through air and inhaled into the human respiratory system.

The MAR is the amount of a radionuclide acted upon by a given physical stress and is calculated using Equation 13a2.2-4-5 for events that involve radioactive decay. For events that do not include decay, the MAR is equal to the activity, A(i), of the radionuclide i in the process stream.

$$MAR(i)[Ci] = A(i) \times \exp(-\lambda(i)\tau) \quad \text{Equation 13a2.2-4-5}$$

where:

$$\begin{aligned} A(i) &\equiv \text{Activity of radionuclide } i \\ \lambda(i) &\equiv \text{Decay constant of radionuclide } i \\ \tau &\equiv \text{Decay time} \end{aligned}$$

### External Dose Equivalent

The EDE is the sum of the dose due to immersion in radionuclides and the dose due to exposure of areal contamination. The dose due to contamination is not evaluated for workers since releases occur in locations that are not normally occupied by workers and the exposure is strictly due to airborne contaminants. The dose due to contamination is not evaluated for the public since releases are filtered through HEPA and charcoal filters and dose from contamination is not expected to be significant. Equation 13a2.2-4-6 shows the calculation for the immersion pathway to the workers and Equation 13a2.2-4-7 for members of the public.

$$E_{IO}[\text{rem}] = \frac{\sum ST_A(i)[Ci] \times DCF(i)_I \left[ \frac{\text{rem} \cdot \text{m}^3}{\text{Ci} \cdot \text{s}} \right] \times t[\text{s}]}{V[\text{m}^3]} \quad \text{Equation 13a2.2-4-6}$$

where:

$$\begin{aligned} E_{IO} &\equiv \text{Occupational EDE due to immersion} \\ ST_A(i) &\equiv \text{Airborne source term for radionuclide } i \\ DCF(i)_I &\equiv \text{Immersion dose conversion factor for radionuclide } i \end{aligned}$$

$$E_{IP}[\text{rem}] = \sum ST_A(i)[Ci] \times DCF(i)_I \left[ \frac{\text{rem} \cdot \text{m}^3}{\text{Ci} \cdot \text{s}} \right] \times \frac{X}{Q} \left[ \frac{\text{s}}{\text{m}^3} \right] \quad \text{Equation 13a2.2-4-7}$$

where:

$$E_{IP} \equiv \text{Public EDE due to immersion}$$

The airborne source term for radionuclide i [ST<sub>A</sub>(i)] is the amount of a radionuclide i at the accident that becomes airborne due to the physical stress acting on the material and is calculated using Equation 13a2.2-4-8, which is similar to Equation 13a2.2-4-4, but the respiratory fraction is not applied.

$$ST_A(i)[Ci] = MAR(i)[Ci] \times DR \times LPF \times ARF(i) \quad \text{Equation 13a2.2-4-8}$$

### Organ Dose

In addition to the TEDE, the thyroid dose for workers is calculated. The equation uses the respirable source term as shown in Equation 13a2.2-4-4 and the DCF for the thyroid.

$$H_o(T)[rem] = \frac{\sum ST_R(i)[Ci] \times DCF_T(i) \left[ \frac{rem}{Ci} \right] \times BR \left[ \frac{m^3}{s} \right] \times t[s]}{V[m^3]} \quad \text{Equation 13a2.2-4-9}$$

where:

$H_o(T)$      $\equiv$     Occupational thyroid dose  
 $DCF_T(i)$     $\equiv$     Thyroid dose conversion factor for radionuclide i

### Off-Site Dose

Since the TEDE is required at both the site boundary and the nearest resident, the off-site doses are calculated slightly different than the methodology described above. Instead of calculating the CEDE and EDE and then summing them for all radionuclides, the  $\chi/Q$  factor is applied after all radionuclide “doses” are calculated. This has no affect in the overall dose, but eliminates repetition of calculations.

### Input Parameter Values

Important input parameter values can be found in Tables 13a2.2.1-1, 13a2.2.1-2, 13a2.2.1-3, 13a2.2.1-4, and 13a2.2.7-1 of the PSAR for potential accidents occurring inside the IF. Important input parameter values can be found in Tables 13b.2.1-1, 13b.2.1-2, 13b.2.1-3, 13b.2.1-4, 13b.2.4-1, and 13b.2.6-1 for potential accidents occurring inside the RPF.

### Dose Calculation Results

Dose calculation results for postulated accident scenarios occurring within the IF and RPF are presented in Table 13a2.2-4-1. The results are presented by the event which corresponds to the associated Subsection of the PSAR. The results in Table 13a2.2-4-1 are broken down into immersion and inhalation doses where appropriate.

**Table 13a2.2-4-1: Potential Consequences of Postulated Accidents Within the IF and RPF**

Event	50 <sup>th</sup> Percentile Dispersion Value				Workers (rem)	
	Site Boundary (rem)		Nearest Resident (rem)			
	Immersion	Inhalation	Immersion	Inhalation	Immersion	Inhalation
Target Solution Release Into the IU Cell (Subsection 13a2.2.1)	1.60E-02	5.05E-04	2.23E-03	7.07E-05	1.95	1.11
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	1.68E-03	5.05E-04	2.35E-04	7.07E-05	3.82E-01	1.12
Mishandling or Malfunction of Equipment Affecting the PSB (Subsection 13a2.2.7)	1.59E-02	---	2.23E-03	---	1.74	4.82E-02
TPS Design Basis Accident (Subsection 13a2.2.12.3)	3.04E-08	5.6E-04	4.26E-09	8E-05	1.33E-04	2.4
MHA in the RPF (Subsection 13b.2.1)	8.20E-02	---	1.15E-02	---	3.59	---
Loss of Piping or Tank Integrity (Subsection 13b.2.4)	1.68E-03	5.05E-04	2.35E-04	7.07E-05	3.82E-01	1.12
Inadvertent Release from NGRS (Subsection 13b.2.4)	8.17E-02	---	1.14E-02	---	3.58	---
RPF Fire (Subsection 13b.2.6)	3.54E-04	5.23E-04	4.95E-05	7.32E-05	5.09E-02	5.28E-01

### **Section 13b.1 – Radioisotope Production Facility Accident Analysis Methodology**

*(Applies to RAIs 13b.1-1 through 2 and 13b.2)*

*As required by 10 CFR 50.34(a)(4), “[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility..., and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.”*

*As set forth in ISG Augmenting NUREG-1537, Part 1, Section 13b, “Radioisotope Production Facility Accident Analyses,” the NRC staff has determined that the “use of ISA methodologies, as described in 10 CFR Part 70 and NUREG-1520, application of the radiological and chemical consequence and likelihood criteria contained in the performance requirements of 10 CFR 70.61, designation of IROFS, and establishment of management measures are acceptable ways of demonstrating an adequate margin of safety for the medical isotopes production facility. Applicants may propose alternate accident analysis methodologies, alternate radiological and chemical consequence and likelihood criteria, alternate safety features, and alternate methods of assuring the availability and reliability of the safety features. As used in the ISG, the term “performance requirements”, when referencing 10 CFR Part 70, subpart H, is not intended to mean that the performance requirements of subpart H are required for a radioisotope production facility license, only that their use as accident consequence and likelihood criteria may be found acceptable by NRC staff.*



### **RAI 13b.1-1**

*The ISG Augmenting NUREG-1537, Part 2, Section 13b.1, "Radioisotope Production Facility Accident Analysis Methodology," states that an "integrated safety analysis should be performed for each process or process segment" in the radioisotope production facility.*

*The cover letter to part two of the application for a construction permit, dated May 31, 2013, (ADAMS Accession No. ML13172A361), states that the ISA Summary will be provided in the Operating License Application. Based on the process descriptions and hazards identified in the PSAR, certain engineered safety features should be identified and described in the PSAR if they will be constructed or procured and installed under the construction permit.*

*Address the following:*

- a) Potential accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena;*
- b) The consequence and the likelihood of occurrence of each potential accident sequence identified, and the methods used to determine the consequences and likelihoods; and*
- c) Each passive engineered or active engineered IROFS, the characteristics of the IROFS' preventive, mitigative, or other safety function; and the assumptions and conditions under which the IROFS is relied upon to support compliance with the performance requirements of 10 CFR 70.61.*

### **SHINE Response**

- a) Potential accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena, are identified in the Initiating Event Scenario column of Table 13b.1-1-4.
- b) The consequences and likelihood of occurrence of each potential accident sequence are included in Table 13b.1-1-4. The methods used to determine the consequences and likelihoods are as follows:

The Consequence Estimate column in Table 13b.1-1-4 provides a description of the consequences of the scenario. The consequence severity categories are based on 10 CFR 70.61, and are provided in Table 13b.1-1-1. The severity categories are assigned per the following designations:

HC: High Consequences

IC: Intermediate Consequences

LT IC: Less than Intermediate Consequences (i.e., Low Consequences)

**Table 13b.1-1-1: Consequence Severity Categories Based on NUREG-1520**

	<b>Workers</b>	<b>Offsite Public</b>	<b>Environment</b>
Category 3 High Consequence (HC)	RD > 1 sievert (Sv) (100 rem)  CD > AEGL-3, ERPG-3	RD > 0.25 Sv (25 rem)  30 milligrams (mg) sol U intake  CD > AEGL-2, ERPG-2	
Category 2 Intermediate Consequence (IC)	0.25 Sv (25 rem) < RD ≤ 1 Sv  AEGL-2, ERPG-2 < CD < AEGL-3, ERPG-3	0.05 Sv (5 rem) < RD ≤ 0.25 Sv  AEGL-1, ERPG-1 < CD ≤ AEGL-2, ERPG-2	Radioactive release > 5,000 times Table 2 of Appendix B to 10 CFR 20
Category 1 Low Consequence (LT IC)	Accidents with lower radiological and chemical exposures than those above	Accidents with lower radiological and chemical exposures than those above	Radioactive releases producing lower effects than those above

**Notes:**

RD = Radiological Dose

CD = Chemical Dose

Definitions of the Acute Exposure Guideline Level (AEGL) and Emergency Response Planning Guideline (ERPG) levels are provided below.

- AEGL-1: The airborne concentration, expressed as parts per million (ppm) or milligrams per cubic meter (mg/m<sup>3</sup>) of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation, or certain asymptomatic non-sensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.
- AEGL-2: The airborne concentration (expressed as ppm or mg/m<sup>3</sup>) of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects or an impaired ability to escape.
- AEGL-3: The airborne concentration (expressed as ppm or mg/m<sup>3</sup>) of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.
- ERPG-1: The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient health effects or perceiving a clearly defined, objectionable odor.
- ERPG-2: The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action.
- ERPG-3: The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Consequences estimates in the Consequence Estimate column in Table 13b.1-1-4 are provided for situations with no mitigation and with mitigation. In some unmitigated cases, credit may be taken for robust passive controls that maintain their effectiveness throughout the environment of the scenario. The description of unmitigated consequences will include a reference to any passive controls if they are being credited in the consequence estimation

for the unmitigated case. The consequence estimate field is usually presented in two paragraphs. The first paragraph describes potential consequences to the public and the second paragraph describes potential consequences to workers. A single paragraph is used if the same overall logic and consequence level applies to both the public and workers.

Frequency definitions, taken from NUREG-1520 (Reference 24), are provided in Table 13b.1-1-2.

**Table 13b.1-1-2: Likelihood Index Limit Guidelines**

	<b>Likelihood Category</b>	<b>Event Frequency Limit</b>	<b>Risk Index Limits</b>
Not Likely	3	More than $10^{-4}$ per event, per year	> -4
Unlikely	2	Between $10^{-4}$ and $10^{-5}$ per event, per year	-4 to -5
Highly Unlikely	1	Less than $10^{-5}$ per event, per year	$\leq$ -5

The likelihood of the event is described in the Unmitigated Likelihood column of Table 13b.1-1-4. This column contains the qualitative estimate of the accident scenario's uncontrolled frequency (likelihood), as determined in the SHINE preliminary hazard analysis. These estimates are general, order of magnitude, estimates using categories (or "bins").

The mitigated likelihood is determined by the number and type of preventor IROFS applied to each accident scenario, and is presented in the Mitigated Likelihood column of Table 13b.1-1-4. Table 13b.1-2-1, provided in the SHINE Response to RAI 13b.1-2, contains a list of IROFS that have been applied to each accident scenario, and whether they provide a preventive or mitigative function. The values in the Mitigated Likelihood column of Table 13b.1-1-4 reflect the reduction of frequency values of Table 13b.1-1-3. To calculate the mitigated likelihoods, it is assumed that the base rate for all unmitigated accident scenarios is 1 per year or 100, which is equivalent to a frequency index of zero. The mitigated likelihood has a maximum value of -6, as higher values are considered unrealistic. If the sum of preventor IROFS values exceeded -6, then the value -6 is underlined (i.e., -6).

**Table 13b.1-1-3: Failure Probability Index Numbers (from NUREG-1520)**

Probability Index No.	Probability of Failure on Demand	Based on Type of IROFS	Comments
-6*	$10^{-6}$		If initiating event, no IROFS needed
-4 or -5*	$10^{-4} - 10^{-5}$	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two redundant IROFS more robust than simple admin. IROFS (AEC, PEC, or enhanced admin.)	Can rarely be justified by evidence. Most types of single IROFS have been observed to fail.
-3 or -4*	$10^{-3} - 10^{-4}$	A single passive engineered IROFS (PEC) or an active engineered IROFS (AEC) with high availability	
-2 or -3*	$10^{-2} - 10^{-3}$	A single active engineered IROFS, or an enhanced admin. IROFS, or an admin. IROFS for routine planned operations	
-1 or -2*	$10^{-1} - 10^{-2}$	An admin. IROFS that must be performed in response to a rare unplanned demand	
* Indices less than (more negative than) -1 should not be assigned to IROFS unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the IROFS may be changed or not maintained.			

- c) The IROFS, as identified in the SHINE ISA, are presented in Table 13b.1-1-5. This table also includes each IROFS' safety function. Table 13b.1-2-1, provided in the SHINE Response to RAI 13b.1-2, provides the accident categories (conditions) where each IROFS is relied on to support compliance with 10 CFR 70.61. Table 13b.1-1-4 identifies that after mitigation, the consequences of all accident sequences are in compliance with 10 CFR 70.61.

Part a of the SHINE Response to RAI 3.5-1 (Reference 4) contains SHINE's process to designate SSCs important to safety that can be efficiently managed in the future licensed plant by integrating the ISA analysis results into the PSAR (FSAR). This process eliminates the IROFS designation from the PSAR and SHINE QAPD. Thus, Table 13b.1-1-5 also contains the new classification of each IROFS and the basis for that classification. IROFS are classified as safety-related if they are an SSC and meet some part of the SHINE revised definition for safety-related. SSCs that do not meet this definition are classified as nonsafety-related. IROFS that are not SSCs are classified as licensing basis administrative items (i.e., technical specifications, programs contained in the Administrative Controls section of the Technical Specifications, descriptions in the FSAR, or NRC Regulatory Commitments implemented in plant procedures) if they would otherwise meet the definition of safety-related.

IROFS that were credited in the ISA with preventing or mitigating accidents with potential consequences described in the SHINE definition of safety-related are now classified as safety-related or as licensing basis administrative items. SHINE has revised the PSAR to accomplish the following:

- 1) Former IROFS that are now considered safety-related SSCs or licensing basis administrative items that were not previously included in the PSAR have been added to the PSAR to ensure that the results of the ISA and the PSAR are aligned.
- 2) Items previously classified in the PSAR as Defense-in-Depth (DID), or classified as IROFS but were not credited with preventing or mitigating accidents in the ISA, have been removed from sections of the PSAR listing safety-related SSCs and Technical Specification Administrative Controls. Changes have also been made to the PSAR to indicate that these items are nonsafety-related.
- 3) Accidents and associated controls related to normal chemical hazards, and not to hazardous chemicals associated with, produced from, or affecting the safety of licensed materials, have been removed from Chapter 13 of the PSAR. Additionally, Tables 13b.3-1 and 13b.3-2 of the PSAR have been revised to apply only to hazardous chemicals produced from licensed materials in cases where using a bounding on-site inventory of a chemical (typically in storage) was not representative of the chemical hazard when it was used in a process with licensed materials. An error in reporting the distance of the site boundary (or the maximally exposed individual (MEI)), has also been corrected from 402 m to 249 m to more accurately reflect the SHINE site and the calculations that have been performed. An IMR has been generated to ensure assumptions made in the calculation of chemical source terms and concentrations are verified during detailed design and that the final results are incorporated into the FSAR.
- 4) The consequences and analyses of an inadvertent criticality accident have been removed from the SHINE PSAR since this accident is highly unlikely, and is prevented at the SHINE facility through the use of multiple preventative controls. Subsection 13b.2.5 of the PSAR provides a description of the safety controls, administrative controls, and safety-related SSCs that maintain subcriticality and prevent an inadvertent criticality under all normal and abnormal conditions in the RPF. The accident scenarios listed in Subsection 13b.2.5 that could cause an unintended criticality are highly unlikely due to the numerous safety controls listed in Table 13b.2.5-1 of the PSAR. Additionally, double-contingency protection will ensure that unintended criticality is highly unlikely.
- 5) Fire detection and suppression systems, including both the Facility Fire Protection System (FFPS) and the Hot Cell Fire Detection and Suppression System (HCFD), are classified as nonsafety-related. No fire detection or fire suppression systems are credited in the ISA. Fire-related accidents are prevented and/or mitigated by other controls, as described in Table 13b.1-2-1, provided in the SHINE Response to RAI 13b.1-2.
- 6) Changes have also been made to sections of PSAR Chapter 13 describing the TPS DBA (Subsections 13a2.1.12.3 and 13a2.2.12.3 of the PSAR). The basis for these changes is described in the SHINE Response to RAI 6a2.2-11.

**Table 13b.1-1-4: Potential Accident Sequences**

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
<b>Aircraft Accident and Seismic Event</b>					
1. Aircraft impact into IF area causes multiple adverse conditions (e.g., fire, deflagration, confinement failure), resulting in widespread dispersal of radioactive and hazardous materials	If no credit is taken for the presence of the facility structure and its internal structures, this event presents a HC to the public and the workers based upon analysis of an airplane crash as a maximum hypothetical accident. Taking credit for the facility structure and its internal structures, which provide passive protection, prevents penetration of all aircraft of credible size and type for accidental impact at this site. No accident-related release of significance is anticipated for such events.	LT IC Pub	2	-6	yes
		LT IC Wrk			yes
2. Aircraft impact into RPF causes multiple adverse conditions (e.g., fire, deflagration, confinement failure), resulting in widespread dispersal of radioactive and hazardous materials	If no credit is taken for the presence of the facility structure and its internal structures, this event presents a HC to the public and the workers based upon analysis of an airplane crash as a maximum hypothetical accident. Taking credit for the facility structure and its internal structures, which provide passive protection, prevents damage to all significant inventories of radioactive materials. Potential consequences of exposure to nuclear materials are thus LT IC for the public and workers.	LT IC Pub	2	-6	yes
		LT IC Wrk			yes
3. Seismic event causes multiple adverse conditions (e.g., fire, deflagration), resulting in widespread dispersal of radioactive and hazardous materials	Unmitigated HC to the public based upon analysis of an airplane crash as a maximum hypothetical accident. Mitigated consequences are LT IC due to mitigators effectively preventing release of radioactive material.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers based upon analysis of an airplane crash as a maximum hypothetical accident. Mitigated consequences are LT IC due to mitigators effectively preventing release of radioactive material.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
<b>Hazardous Chemicals</b>					
1. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* inside vaults or cells (includes associated valves, piping, and overflow lines) due to operational mechanical failures, human errors, or natural phenomena events (e.g., leaks, rupture)	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.	N/A	3	N/A	N/A
* With the potential for exceeding ERPG-2 limits at site boundary	An unmitigated HC to workers due to potential direct contact with hazardous chemicals. In the mitigated case, the workers are not significantly exposed to the spill as it is in an area that is isolated from them. Their unmitigated consequence is LT IC.  Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 7, 9, 12, 13, 15, 16, and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Number 2, 3, and 4.	N/A			N/A
2. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* outside vaults or cells (includes associated valves, piping, and overflow lines) due to operational mechanical failures, human errors, or natural phenomena events (e.g., leaks, rupture)	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.	N/A	3	N/A	N/A
* With the potential for exceeding ERPG-2 limits at site boundary	An unmitigated HC to workers due to potential direct contact with hazardous chemicals. Mitigative measures including protective equipment and maintenance and other measure reduce potential consequences to IC.  No consequence involving licensed material	N/A			N/A

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
3. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* inside vaults or cells (includes associated valves, piping, and overflow lines) due to fires.	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.	N/A	3	N/A	N/A
* With the potential for exceeding ERPG-2 limits at site boundary	An unmitigated HC to workers due to potential direct contact with hazardous chemicals. In the mitigated case, the workers are not significantly exposed to the spill as it is in an area that is isolated from them. Their unmitigated consequence is LT IC.  Potential consequences involving licensed material are described in scenarios for fires in IF and RPF.	N/A			N/A
4. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* outside vaults or cells (includes associated valves, piping, and overflow lines) due to fires.	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.	N/A	3	N/A	N/A
* With the potential for exceeding ERPG-2 limits at site boundary	An unmitigated HC to workers due to potential direct contact with hazardous chemicals. Mitigative measures including protective equipment and maintenance and other measure reduce potential consequences to IC.  Potential consequences involving licensed material are described in Fires in IF and RPF scenarios 2A, 1B, 2B, 6B, 7B.	N/A			N/A



Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
5. Exothermic reactions between chemicals leading to damage to tanks or vessels containing significant quantities of hazardous toxic materials*in the RPF.	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.	N/A	3	N/A	N/A
* With the potential for exceeding ERPG-2 limits at site boundary	An unmitigated HC to workers due to potential direct contact with hazardous chemicals. Mitigative measures including protective equipment and maintenance and other measure reduce potential consequences to IC.  Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 7, 9, 12, 13, 15, 16, and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Number 2, 3, and 4.	N/A			N/A
6. Mishap during handling of chemicals leads to breach or spill of chemicals from tanks or vessels	An unmitigated HC to the public based on a loading dock spill calculation. A spill inside the building is highly diluted by exhaust from areas of the building not affected by the spill as well as the elevated release point. The mitigated consequence to the public is LT IC when the full scope of mitigative measures is considered.  An unmitigated HC to workers due to potential direct contact with hazardous chemicals. Mitigative measures including protective equipment and maintenance and other measure reduce potential consequences to IC.  Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 8, 11, and 14 and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Numbers 1 and 5.	N/A  N/A	3	N/A	N/A  N/A

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
7. Mishap during handling of chemicals leads to a minor spill of chemicals outside tanks or vessels	Minor limited quantity spill results in LT IC to the public and workers.	N/A	3	N/A	N/A
	Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 8, 11, and 14 and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Numbers 1 and 5.	N/A			N/A
8. Mishap (e.g., spill) during delivery of hazardous chemicals outside the facility. (Non-licensed material, assumes correct chemical is being delivered.)	Unmitigated IC to the public per calculation for HNO <sub>3</sub> . Emergency Preparedness Program reduces consequence to LT IC by combination of restricted access to affected areas and prompt treatment of spilled material so that the time for free evaporation from the pool is limited to one hour or less, thereby limiting the potential impact on the nearest residents. Calculation shows LT IC for one hour evaporation release.	N/A	3	N/A	N/A
	Unmitigated HC to worker per calculation. Assume protective equipment and spill procedures will reduce potential worker exposure to LT IC.	N/A			N/A
	No consequence involving licensed material.				
<b>Criticality</b>					
1. Seismic event causes addition of excess reactivity in TSV or process area equipment (e.g., change in geometry, addition of uranium), resulting in inadvertent criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
2. Overfill of TSV results in inadvertent criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
3. Uranium solution with elevated concentration or enrichment is introduced into TSV target solution, resulting in an inadvertent criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	-5	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
4. Uranium bearing particles transferred from the Uranyl Sulfate Preparation Tank (1-TSPS-01T) to the Target Solution Hold Tank (1-TSPS-03T) as a result of incomplete dissolution of UO <sub>3</sub> or U causes inadvertent criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	-6	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
5. Loss of uranium solution from TSV or processing equipment (e.g., vessel fails), resulting in the accumulation of target solution in an unfavorable geometry, causing an inadvertent criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
6. Positive displacement pump operates while downstream process line is blocked, causing over-pressurization of process components containing fissile solution, resulting in a criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
7. H <sub>2</sub> SO <sub>4</sub> supply tank accidentally charged with HNO <sub>3</sub> . HNO <sub>3</sub> then used as the solvent in U Metal Dissolution Tank, subsequently resulting in a criticality in the U Metal Dissolution Tank or other vessels (e.g., Target Solution Hold Tank, TSV).	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	-5	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
8. Water entering TSV, as a result of failure of the boundary between them, could have corrosion inhibitors or other components that could promote precipitation or reactivity issues, etc. resulting in a criticality.	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
9. Failure of boundary between TSV and light water jacket could result in pressure transients or oscillations.  Uranyl sulfate solution and fission products migrate into light water jacket and are transported to systems supporting the light water jacket. Water entering TSV could have corrosion inhibitors or other components that could promote precipitation or reactivity issues, etc. Hazards include power transient/oscillation and criticality (precipitation), transfer of uranium in light water jacket processing system, etc.	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	-6	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
10. Loss of power to contactors results in uranium and solvent carryover to raffinate tank, resulting in a criticality	Unmitigated IC to public due spilled solution and released fission gases. Mitigated LT IC due to mitigative measures.	LT IC Pub	3	-5	yes
	Unmitigated HC to workers due to exposure to excursion and radiation. Mitigated LT IC due to significant shielding and robust Irradiation Unit construction.	LT IC Wrk			yes
<b>H2 Deflagration or Detonation</b>					
1. Failure of the TSV Off-gas system due to loss of normal power	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-4	yes
	LT IC for workers due to isolation and robustness of irradiation cell.	LT IC Wrk			yes
2. Failure of the TSV Off-gas system due to loss recombiner capability	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-4	yes
	LT IC for workers due to isolation and robustness of irradiation cell.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
3. Failure of the TSV Off-gas system due to blockage (e.g., plugged zeolite beds, closed dampers or other failure) Hydrogen deflagration overpressurizes system causing release of radioactive solution.	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-4	yes
	LT IC for workers due to isolation and robustness of irradiation cell.	LT IC Wrk			yes
4. Accumulation of hydrogen and oxygen in the noble gas hold up waste tank(s). Ignition of the flammable mixture fails noble gas waste tank.	IC to public based on analysis of noble gas release associated with accumulated inventory from four cycles.	IC Pub	3	-4	yes
	HC to Worker based on analysis of noble gas release associated with accumulated inventory from four cycles. Mitigated consequence to workers is IC, due largely to administrative measures (e.g., radiation alarms, emergency procedures).	IC Wrk			yes
5. Failure to vent through the Process Vessel Vent System (PVVS in the RPF) results in release of radionuclides to RPF open areas.	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially zone 1, 2, and 3 ventilation.	LT IC Pub	3	-2	yes
	Unmitigated HC due to proximity of release. Mitigated case is LT IC due to mitigative measures including radiation alarms and administrative controls (e.g., procedures for evacuation).	LT IC Wrk			yes
6. Uranium metal dissolution tank sits idle for extended period (e.g., facility or module stand-down), causing accumulation of radiolysis products in the tank's head	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially zone 1, 2, and 3 ventilation.	LT IC Pub	3	-1	yes
	Unmitigated HC due to proximity of release. Mitigated case is LT IC due to mitigative measures including radiation alarms and administrative controls (e.g., procedures for evacuation).	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
7. Leak in accelerator deuterium supply results in gas ignition and deflagration, with dispersal of nearby radioactive material	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-5	yes
	LT IC for workers due to isolation and robustness of irradiation cell.	LT IC Wrk			yes
Deflagrations, Detonations, and Explosions for Materials Other Than Hydrogen					
1. Excessive time of process solution in the evaporator, creating increased concentrations and temperatures that promote formation of unstable compounds (e.g., reactions between nitric acid, Tri-Butyl Phosphate (TBP), and related decomposition products) that accumulate over time, resulting in an explosion.	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-2	yes
	HC for workers based on maximum hypothetical accident analysis. Mitigation measures for workers, especially chemical cells and vaults, reduces potential public consequences to LT IC.	LT IC Wrk			yes
2. Degradation products not removed from Strip Solution en route to Recycle UN Hold Tank. Solutions transferred for processing in the UN Evaporator and Thermal Denitrator resulting in a sudden reaction of unstable species giving rise to a chemical explosion in the UN Evaporator or Thermal Denitrator	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially filtered ventilation reduces potential public consequences to LT IC.	LT IC Pub	3	-4	yes
	Unmitigated HC for workers and LT IC when mitigated.	LT IC Wrk			yes
3. Leak in natural gas supply line outside facility results in gas ignition and deflagration external to the facility, with release of radioactive and hazardous materials.	HC for public based on maximum hypothetical accident analysis. Mitigation measures for public, especially facility structure which reduces potential impact to the facility internals and mitigates public consequences to LT IC.	LT IC Pub	3	0	yes
	LT IC for workers due to isolation and robustness of facility, and irradiation unit.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
<b>Fires in IF and RPF</b>					
<b>(A) Fire in Irradiation Facility (IF) area</b>					
1A. Operational fires where the fire is initiated inside the irradiation cell (irradiation cell is locked and closed).	LT IC for both workers and the public because the irradiation cell is impervious to fire, thus no significant release of radioactive material occurs.	LT IC Pub	3	-1	yes
		LT IC Wrk			yes
2A. Maintenance fires (access to irradiation cell is open). Fire may be initiated anywhere.	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially High Efficiency Particulate Air (HEPA) filtration.  Unmitigated HC due to open access to irradiation cell. Mitigated case is LT IC due to mitigative features, especially radiation alarms and administrative controls such as procedures for evacuation.	LT IC Pub	3	-2	yes
		LT IC Wrk			yes
3A. Fire propagation from RPF when the irradiation cell is locked and closed (i.e., Initiated in RFP and propagates to Irradiation Cell).	Not credible due to fire rating of irradiation cell	N/A	3	N/A	N/A
4A. Fire propagation from RPF (cell door and/or access port open)	(to be covered under RPF fires)	N/A	3	N/A	N/A
<b>(B) Fires initiated in RFP</b>					
1B. Operational fires initiated in RFP that affect RPF areas (inside or outside hot cells).	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially HEPA filtration.  Unmitigated HC to workers due to potential proximity of fire or fire damage to hot cell causing release transport pathways for radionuclides. Mitigated case is LT IC due to mitigative measures, including procedures for evacuation.	LT IC Pub	3	-6	yes
		LT IC Wrk			yes



Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
2B. Maintenance and other non-operations fires initiated in RPF (may initiate inside or outside hot cells but propagates to both areas)  Note: Deflagration or detonation not credible within the RPF	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially HEPA filtration.	LT IC Pub	3	-3	yes
	Unmitigated HC due to proximity of fire. Mitigated case is LT IC due to mitigative measures, especially administrative controls such as procedures for evacuation.	LT IC Wrk			yes
3B. Fire initiated in RPF (outside the irradiation cell) that propagates to the IF area (irradiation cell locked and closed).	Not credible due to fire rating of irradiation cell	N/A	3	N/A	N/A
4B. Fires initiated in vessels or tanks due to hydrogen accumulation and presence of radiolytic oxygen or air  Note: Potential deflagration for high inventory tanks or vessels with small head space if fails to purge hydrogen from the vessels or tanks.	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially HEPA filtration.	LT IC Pub	3	-5	yes
	Unmitigated HC to workers due to potential proximity of fire or fire damage to vessels or piping causing release transport pathways for radionuclides. Mitigated case is LT IC due to mitigative measures, including procedures for evacuation.	LT IC Wrk			yes
5B. Vent line to raffinate holding tank is plugged	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to mitigators, especially HEPA filtration.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers due to potential proximity of fire or fire damage to hot cell causing release transport pathways for radionuclides. Mitigated case is LT IC due to mitigative measures, including procedures for evacuation.	LT IC Wrk			yes
6B. Lightning strike to facility results in facility fire	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to facility structure.	LT IC Pub	3	0	yes
	Unmitigated HC to workers due to potential proximity of fire. Mitigated case is LT IC due to mitigative features including facility structure.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
7B. Leak in natural gas supply line outside facility results in gas ignition and fire that spreads into facility	Unmitigated HC to public based on maximum hypothetical accident analysis. Mitigated release is LT IC due to facility structure.	LT IC Pub	3	-6	yes
	Unmitigated HC to workers due to potential proximity of fire. Mitigated case is LT IC due to mitigative features including facility structure.	LT IC Wrk			yes
<b>Mishandling or Malfunction of Target Solution</b>					
1. TSV overfill	LT IC for public due to limited quantity of material released and the presence of material dissolved in solution.	LT IC Pub	3	-6	yes
	LT IC for workers due separation from the area of the spill.	LT IC Wrk			yes
2. TSV or Dump Tank Leak Into the Light Water Pool or primary cooling system	LT IC to public due to limited release quantity	LT IC Pub	3	-6	yes
	Unmitigated IC to worker due to lack of information regarding the potential for exposure. LT IC in mitigated case due to alarms, etc. indicating the failed condition.	LT IC Wrk			yes
3. Dump Tank piping Leak Into the Irradiation Cell  <i>Note: No sump in the IU, any leak from the dump tank will go to the cooling pool.</i>	LT IC for public due to limited quantity of material released.	LT IC Pub	3	-6	yes
	Unmitigated IC for worker due to lack of knowledge or warning of the released material, Mitigated exposure is LT IC due to information and warnings of the release of radioactive material, in addition to isolation to the irradiation cell.	LT IC Wrk			yes
4. Dump Tank piping Leak Into the Irradiation Facility Outside of the Irradiation Hot Cell	LT IC to public due to limited quantity released.	LT IC Pub	3	-4	yes
	Unmitigated HC to worker due to lack of knowledge of the release of radioactive material and their proximity to the material. Mitigated case is LT IC due to alarms and emergency procedures.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
5. Misdirection of Irradiated Target Solution Back To the Target Solution Hold Tank	LT IT to public due to limited quantity released from facility.	LT IC Pub	3	-6	yes
	LT IC to workers due to location of the vessel inside a hot cell.	LT IC Wrk			yes
<b>Mishandling or Malfunction of Equipment (Off-gas/NGRS)</b>					
1. Inadvertent venting of the Off-gas purge contents	Based on a large inventory noble gas release calculation, an unmitigated LT IC is assessed for the public due to limited inventory that could be released from a single run.	LT IC Pub	3	0	yes
	LT IC to workers as the release is through the vent system.	LT IC Wrk			yes
2. Release of noble gases from hold up tank due to mechanical failures (e.g., leaks, vessel failures)	Calculation shows an unmitigated IC to the public in the event of a large inventory noble gas release.	IC Pub	3	-6	yes
	An unmitigated HC to workers due to proximity to the release. This is mitigated to IC when the full scope of mitigative measures is applied to workers.	IC Wrk			yes
3. Release from the Off-gas system due to mechanical failures (e.g., leaks or bypass of the NGRS)	Calculation shows an unmitigated IC to the public in the event of a large inventory noble gas release.	IC Pub	3	-4	yes
	An unmitigated HC to workers due to proximity to the release. This is mitigated to IC when the full scope of mitigative measures is applied to workers.	IC Wrk			yes
<b>Loss of Confinement</b>					
1. Strong straight winds breach facility and uranium solution processing equipment, with release of radioactive and hazardous materials	HC unmitigated consequences to the public based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for the public.	LT IC Pub	3	<u>-6</u>	yes
	HC unmitigated consequences to workers is based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for workers.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
2. Tornado breaches facility and uranium solution processing equipment, with release of hazardous and radioactive materials	HC unmitigated consequences to the public based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for the public.	LT IC Pub	3	<u>-6</u>	yes
	HC unmitigated consequences to workers is based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for workers.	LT IC Wrk			yes
3. Heavy rain/flooding causes damage to facility and process equipment resulting in dispersal of radioactive and hazardous materials	HC unmitigated consequences to the public based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for the public.	LT IC Pub	3	<u>-6</u>	yes
	HC unmitigated consequences to workers is based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for workers.	LT IC Wrk			yes
4. Heavy snow causes damage to facility and process equipment resulting in dispersal of radioactive and hazardous materials	HC unmitigated consequences to the public based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for the public.	LT IC Pub	3	<u>-6</u>	yes
	HC unmitigated consequences to workers is based on maximum hypothetical accident analysis. Facility is designed to withstand natural phenomena stresses so potential consequences would be mitigated to LT IC for workers.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
5. Vessel (e.g., extraction column) rupture or leak, resulting in discharge of radioactive materials	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to separation from the location of the spill and adequate shielding and isolation.	LT IC Wrk			yes
6. Explosion at nearby facility generates a mechanical insult that compromises the integrity of the Off-gas or Noble Gas Removal System, leading to dispersal of radioactive gas	IC to public based on analysis of noble gas release associated with accumulated inventory from four cycles.	IC Pub	3	<u>-6</u>	yes
	HC to Worker based on analysis of noble gas release associated with accumulated inventory from four cycles. Mitigated consequence to workers is IC, due largely to administrative measures (e.g., radiation alarms, emergency procedures).	IC Wrk			yes
7. Leak in TSV cooling lines causes flow of TSV coolant into TSV vessel. TSV overfills, sending TSV solution into TSV vent line. TSV solution enters Off-gas system and eventually drains into recombiners, generating steam, which damages process system components.	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-2	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to separation from the location of the spill and adequate shielding and isolation.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
8. Process tank (e.g., Target Solution Hold Tank) overflows into vent system allowing a release pathway for fission products (recognizes a pathway for liquids that was only intended for handling tank vapors)	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to IC mitigative features including radiation alarms and administrative controls including protective actions.	LT IC Wrk			yes
9. Pressure increases to pump head pressure in Mo Extraction Column, resulting in a spill	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to hot cell that isolates workers from solutions.	LT IC Wrk			yes
10. No flow from TSV Off-gas System to Noble Gas System due to blockage in pathway, resulting in fission gases released through the PVVS	Unmitigated consequence of IC to public based on analysis of noble gas release associated with accumulated inventory from four cycles. With mitigation, such as radiation monitor alerting of the failure and termination of operations, the potential consequence is limited to LT IC.	LT IC Pub	3	-2	yes
	IC to Worker due to potential external exposure due to proximity of release pathway (i.e. PVVS lines, external line source). Mitigated consequence to workers is LT IC, due largely to administrative measures (e.g., radiation alarms, emergency procedures).	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
11. Unanticipated transfer into <sup>99</sup> Mo system (Line 0308) resulting in Mo Extraction System hot cell spill.	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-2	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to hot cell that isolates workers from solutions.	LT IC Wrk			yes
12. Positive displacement pump operates while downstream process line is blocked, causing overpressurization of process components containing fissile solution, resulting in a spill in the IF	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-2	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to hot cell that isolates workers from solutions.	LT IC Wrk			yes
13. Positive displacement pump operates while downstream process line is blocked, causing overpressurization of process components containing fissile solution, resulting in a spill in the radioisotope production facility (RPF)	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated consequences would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-2	yes
	IC to Worker due to potential external exposure due to proximity of release. Mitigated consequence to workers is LT IC, due largely to administrative measures (e.g., radiation alarms, emergency procedures).	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
14. Sulfuric acid source tank charged with nitric acid, causing uranium oxides to be dissolved in nitric acid, resulting in uranyl nitrate solution being sent to Target Solution Hold Tank with potential for release of NOx	Unmitigated consequences of LT IC to public due to limited release rate of NOx.	N/A	3	N/A	N/A
	Unmitigated consequences of LT IC to workers due to isolation from the released gas.	N/A			N/A
	No consequence involving Licensed Material				
15. Rotating equipment malfunctions, resulting in rupture of solution tank and release of uranium solution	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-2	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to hot cell that isolates workers from solutions.	LT IC Wrk			yes
16. Failure of liquid waste tank (e.g., corrosion) results in release of radioactive material.	Unmitigated IC to public based on release being less than 10 % of that of the maximum credible accident (e.g., per lower release fractions and less energy input). Mitigated would be LT IC due to mitigation including confinement functions.	LT IC Pub	3	-5	yes
	Unmitigated HC worker consequences due to proximity to high activity solutions. Mitigated worker consequences are reduced to LT IC due to hot cell that isolates workers from solutions.	LT IC Wrk			yes
<b>Excess Reactivity</b>					
1. Pressurization of target solution fluid in TSV	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution.	LT IC Pub	3	-2	yes
(Details of deflagration consequences to be addressed in the deflagration section)	LT IC for workers due separation from the area of the spill.	LT IC Wrk			yes



Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
2. Excessive cool down of target solution fluid in TSV	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution.	LT IC Pub	3	-2	yes
	LT IC for workers due separation from the area of the spill.	LT IC Wrk			yes
3. Moderator addition in TSV due to cooling system malfunction (e.g. cooling tube rupture)	The inherent response of the system would reduce its reactivity, but a release path could exist into the cooling system. Radionuclides are in dilute solution with only a potential indirect path to release. Consequences to the public would be LT IC.	LT IC Pub	3	0	yes
	Unmitigated consequences to the worker would be IC due to lack of knowledge of the event and the potential proximity of radionuclides. Mitigating features, including radiation alarms would alert workers to take protective actions per administrative controls. The mitigated worker consequence is LT IC.	LT IC Wrk			yes
4. Target solution injection leads to excessive level in TSV	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution.	LT IC Pub	3	-4	yes
	Unmitigated IC for workers due potential for target solution to enter PVVS and provide radiation fields in unexpected places. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
5. Adverse geometry changes in TSV lead to insertion of excess reactivity	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution. Potential for criticalities involving spilled solution would provide LT IC to the public with credit taken for passive building structure which is robust.	LT IC Pub	3	-3	yes
	Unmitigated IC for workers due potential for exposure to criticality emissions from spilled solutions as well as released radionuclides. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes
6. Moderator lumping effects in the target solution lead to reactivity insertion in TSV	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution. Potential for criticalities involving spilled solution would provide LT IC to the public with credit taken for passive building structure which is robust.	LT IC Pub	3	0	yes
	Unmitigated IC for workers due potential for exposure to criticality emissions from spilled solutions as well as released radionuclides. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
7. Inadvertent introduction of other materials into the target solution	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution. Potential for criticalities involving spilled solution would provide LT IC to the public with credit taken for passive building structure which is robust.	LT IC Pub	3	-4	yes
	Unmitigated IC for workers due potential for exposure to criticality emissions from spilled solutions as well as released radionuclides. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes
8. Bulk boiling of the target solution	Analysis shows that this scenario is not credible for this system.	N/A	1	N/A	N/A
9. Precipitation of uranium/fission products leads to insertion of excess reactivity in the TSV	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution. Potential for criticalities involving spilled solution would provide LT IC to the public with credit taken for passive building structure which is robust. Even if consequences are more severe, facility structure, ventilation systems, and other mitigative features will assure that the mitigated consequences are LT IC.	LT IC Pub	3	-3	yes
	Unmitigated IC for workers due potential for exposure to criticality emissions from spilled solutions as well as released radionuclides. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
10. Excess enrichment of U-235 (>20%) or excess concentration of solution in TSV leads to excess reactivity insertion	LT IC for public due to potential for limited quantity of material to be released and the presence of material dissolved in solution. Potential for criticalities involving spilled solution would provide LT IC to the public with credit taken for passive building structure which is robust. Even if consequences are more severe, facility structure, ventilation systems, and other mitigative features will assure that the mitigated consequences are LT IC.	LT IC Pub	3	-4	yes
	Unmitigated IC for workers due potential for exposure to criticality emissions from spilled solutions as well as released radionuclides. Mitigated consequences are LT IC due to radiation alarms and worker protective actions per administrative controls.	LT IC Wrk			yes
<b>Radiation Exposure</b>					
1. Inadvertent operation of accelerator, resulting in radiation exposure to workers	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-6	yes
	Unmitigated HC to workers. Mitigated IC to workers	IC Wrk			yes
2. Misalignment of accelerator beam, resulting in radiation exposure to workers	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	<u>-6</u>	yes
	Unmitigated HC to workers. Mitigated LT IC to workers by virtue of shielding design which considers this exposure pathway.	LT IC Wrk			yes
3. Inadvertent access to irradiation cell during irradiation, resulting in radiation exposure to workers	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-4	yes
	Unmitigated HC to workers. Mitigated LT IC to workers	IC Wrk			yes

Initiating Event Scenario	Consequence Estimate	Mitigated Conseq.	Unmitigated Likelihood	Mitigated Likelihood	10 CFR 70.61 Comply
4. Mishap during handling of waste container leads to container breach, resulting in worker contamination	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-4	yes
	Unmitigated HC to workers. Mitigated IC to workers	IC Wrk			yes
5. Loss of hot cell confinement during hot cell operations (e.g., seal failure) results in worker contamination	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-3	yes
	Unmitigated IC to workers. Mitigated LT IC to workers	LT IC Wrk			yes
6. Failure in the tritium purification system results in leak of tritium	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-4	yes
	Unmitigated IC to workers. Mitigated LT IC to workers	LT IC Wrk			yes
7. Inadvertent access to cell during operation, resulting in radiation exposure to workers	Unmitigated LT IC to the public due to the passive structure that separates them from the hazard.	LT IC Pub	3	-6	yes
	Unmitigated HC to workers. Mitigated IC to workers	IC Wrk			yes

**Table 13b.1-1-5: Items Relied on For Safety**

<b>Control</b>	<b>Safety Function</b>	<b>SHINE Classification</b>	<b>Classification Basis</b>
Use of Accelerator Audible/Visual Warnings	Provides audible and visual signals prior to and during the activation of the Accelerator. Incorporates a delay so that personnel can react to the warning signals.	TS AC	Not an SSC, but related to Part 3 of safety-related (SR) definition
Accelerator Interlocks	Control consists of the interlocks that prevent the activation of the accelerator while personnel are present.	SR	Part 3 of SR definition
Accelerator Key Switch	Control consists of the key switch that prevents the activation of the accelerator while personnel are present.	SR	Part 3 of SR definition
Accelerator Kill Switch	Manual switch located in the IF area that can be used to stop accelerator operation.	SR	Part 3 of SR definition
Accelerator Manual Shut-off	Manual switch used to shut down the accelerator in the event of any abnormalities.	SR	Part 3 of SR definition
Analysis of Feed Material	Facility analysis to ensure that uranium feed material received by the facility has the proper enrichment and other characteristics. This analysis is performed prior to usage of uranium feed material within the facility.	TS AC	Not an SSC, but related to Part 4 of SR definition
Chemical Analysis of Feed Solutions	Facility chemical analysis of feed solutions (e.g., acids) to ensure that they are the appropriate chemicals for the intended usage. This analysis is performed prior to usage with the facility.	TS AC	Not an SSC, but related to Part 4 of SR definition
Chemical Concentration Limits	Facility analysis to determine the concentrations of chemicals (e.g., acids) to ensure that these concentrations are appropriate for the intended usage. This analysis is performed prior to usage with the facility.	TS AC	Not an SSC, but related to Part 4 of SR definition
Chemistry Control for Irradiation Solution	Used to control hydrogen peroxide so as to prevent precipitation of uranium and fission products. Precipitation of uranium and fission products could lead to an inadvertent criticality.	TS AC	Not an SSC, but related to Part 4 of SR definition
Column Pressure Monitor	Provides pressure monitoring for Mo Extraction Column	SR	Part 3 of SR definition, Part 5 of SR definition
Combustible Loading Limits	Limits on transient combustible loading restricts the size of a fire and makes a fuel-limited fire. No combustibles allowed in berms, sumps, and pipe trenches.	TS AC	Not an SSC, but relied on to ensure SR SSCs are available and reliable
Conduct of Operations Program	This program provides workers and operations management a disciplined and formal method for safely performing work.	TS AC	Not an SSC, but required to ensure SR SSCs are available and reliable
Criticality Accident Alarm System	Provides monitoring to alert personnel to a criticality event.	SR	Part 3 of SR definition
Criticality Safety Program	This program is used to prevent inadvertent nuclear criticality. General limits and controls (engineered and administrative) are applied to fissionable material operations to ensure subcritical configurations under all normal and credible abnormal conditions.	TS AC	Not an SSC, but related to Part 4 of SR definition

Control	Safety Function	SHINE Classification	Classification Basis
Dampers (Irradiation Cell)	Redundant "bubble-tight" dampers isolate the irradiation cells and confine any potential release of radioactive material. The dampers are interlocked with radiation monitors and fire detection. Closure of dampers on fire detection causes the fire to become oxygen limited. However, the dampers do not function as fire dampers. Instead, the welded ventilation ductwork provides a fire barrier to the cell penetrations. Isolation function defeated when door or access port is open.	SR	Part 3 of SR definition
Dampers (RPF)	Redundant "bubble-tight" dampers isolate the hot cells and confine any potential release of radioactive material. The dampers are interlocked with radiation monitors and fire detection. Closure of dampers on fire detection causes the fire to become oxygen limited. However, the dampers do not function as fire dampers. Instead, the welded ventilation ductwork provides a fire barrier to the cell penetrations.	SR	Part 3 of SR definition
Deuterium Source Vessel Inspection Program	Ensures that deuterium is present during accelerator activation and prevents the release of tritium from the tritium target.	TS AC	Not an SSC, but related to Part 3 of SR definition
Electrical Equipment Design	All electrical equipment (except neutron driver) is designed to the National Electrical Code (NEC).	N/A	SHINE design standard as described in the PSAR, some SSCs are SR, some are NSR
Facility Shielding	The primary safety function of this control is to provide shielding for gamma radiation and neutrons.	SR	Part 3 of SR definition
Facility Siting	Reduces the threat of vehicles impacting the facility as the facility is located far from commonly traveled roads. This control also provides spacing between the facility structure and the electrical substation. In addition, this control establishes distance between members of the public and the facility, which can serve to reduce accident consequences to members of the public (e.g., from a criticality event).	PSAR/FSAR	Not an SSC. Facility site is described in Chapters 1 and 2 of the PSAR
Facility Structure	The facility's primary safety function is to provide confinement of radioactive materials and chemicals. The structure will also provide protection from external hazards such as fire and wind-driven missiles.	SR	Part 3 of SR definition
Filters (Downstream of 1-TSPS-01P and 1-TSPS-02P)	Prevents uranium metal particles or other fissile particles from entering further into the process system and introducing chemical or neutron reactivity issues.	SR	Part 4 of SR definition
Filters (Downstream of 1-TSPS-01P and 1-TSPS-02P) Differential Pressure Monitor	Provides indication of differential pressure across tank filters. The filters prevent uranium metal particles or other fissile particles from entering further into the process system and introducing chemical or neutron reactivity issues.	SR	Part 4 of SR definition
Fire Protection Program	This program develops and maintains effective fire protection and suppression measures for the protection of personnel and facilities. The Fire Protection Program is included in the Technical Specification Administrative Controls.	TS AC	Not an SSC, but related to Part 3 of SR definition

Control	Safety Function	SHINE Classification	Classification Basis
Fire Rating for Irradiation Cell	Prevents fire propagation between fire areas. Minimum fire rating is 2 hours (goal is 3 hours). Fire rating lost when access port is open. The Fire Protection Program is included in the Technical Specification Administrative Controls.	SR	Part 3 of SR definition
Fire Rating for RPF Areas	Fire rated areas prevent fire propagation between fire areas. Minimum fire rating of 2 hours for all RFP fire areas except vaults. Minimum fire rating of 1 hour between vaults and to the vaults. Fire rating lost when door or access port is open. The Fire Protection Program is included in the Technical Specification Administrative Controls.	SR	Part 3 of SR definition
Fire Watch During Maintenance Operations	Prevents fire initiation, provides early suppression, and compensates for loss of fire barrier (Tech Spec issue). The Fire Protection Program is included in the Technical Specification Administrative Controls.	TS AC	Not an SSC, but related to Part 3 of SR definition
Fissile Concentration Limits	Administrative limits for fissile material that address both enrichment and concentration of 235U. Measurements and verification is performed prior to transferring solution to the TSV.	TS	Not an SSC, but related to Part 4 of SR definition
Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment	Prevents criticality. In addition, this control serves to protect against criticality-induced vessel or pipe rupture.	SR	Part 4 of SR definition
High Level Waste Containers	Provides confinement for the high level (to be defined later) waste containers.	TS AC	Not an SSC, but related to Part 3 of SR definition
Use of Hot Cell Door Audible/Visual Warnings	Provides audible and visual signals prior to opening access pathways into the hot cell.	TS AC	Part 3 of SR definition
Hot Cells requiring periodic / routine entry door Interlock with Internal Radiation Field	Prevents cell door from being opened if there is an excessively high internal radiation field. The interlock can be bypassed, but only with special keys that are administratively controlled.	SR	Part 3 of SR definition
Hot Cell Door Key Switch	Consists of a key switch that allows opening of the cell door. The keys are administratively controlled.	SR	Part 3 of SR definition
Hot Cell and Irradiation Cell Integrity	Provides confinement. Hot cells are located within the RPF, while the irradiation cells are located in the IF Area.	SR	Part 3 of SR definition
Hot Cell Structure	The hot cell structure provides protection from external hazards, such as aircraft impact and seismic effects.	SR	Part 3 of SR definition
Hydrogen Monitor (NGRS)	Provides information on hydrogen concentration in the NGRS.	SR	Part 3 of SR definition
Hydrogen Monitor (TSV Off-gas)	Provides information on hydrogen concentration. Monitors hydrogen in TSV Off-gas system (downstream from the recombiners) preventing venting into suction side of the compressor system, thus preventing hydrogen from being present in the noble gas tank. Hydrogen and pressure accumulation trip signals are interlocked to the TRPS and dump valves	SR	Part 3 of SR definition



Control	Safety Function	SHINE Classification	Classification Basis
Irradiation Unit	Provides confinement function for fission product release. Houses TSV, TSV dump tank and associated piping. Designed to survive all environments created by postulated accidents. Designed to withstand overpressures created by deflagration or detonation. Seismically qualified. Designed to withstand overpressures created by deflagration or detonation. Includes IF Area penetration seals, which serve to contain target solution leakage within the IF Area and prevent it from flowing into the RPF or into the surrounding environment. Also includes the Irradiation Cell Biological Shielding (ICBS).	SR	Part 3 of SR definition
Irradiation Unit Isolation Valves	These valves provide a means to isolate the boundary between the IF Area and RPF. Piping that penetrates the IF Area boundary is isolatable by means of redundant valves in series.	SR	Part 3 of SR definition
Light Water Coolant Activity Monitoring Program	Provides information on radioactivity within light water coolant.	TS AC	Not an SSC, but related to Part 3 of SR definition
Light Water Pool System (LWPS)	A closed cooling loop that circulates the light water pool water through it to remove waste heat generated during normal irradiation and shut down operations. This system reduces the likelihood of an excess reactivity insertion by preventing excessive cooldown or pressurization of the TSV contents. The light water pool provides a barrier separation between the TSV and cooling water.	SR	Part 2 of SR definition
Low Level Waste Containers	Provides confinement for low level waste (to be defined later).	TS AC	Not an SSC, but related to Part 4 of SR definition
Manual TSV Trip Switch	Provides a means for operators to override all signals to shut down irradiation operations. Interlocked to the neutron driver and to the dump valves.	SR	Part 2 of SR definition
Material Receipt	Helps to ensure that the uranium received by the facility has the proper enrichment and other characteristics. This action involves inspection of paperwork provided by the supplier upon delivery of material at the facility.	TS	Not an SSC, but related to Part 4 of SR definition
Moisture-Leak Detection/Instrumentation and Alarm (Rad)	Alerts personnel to the situation that radiological materials that should be in tanks and piping have spilled. Provides a means to detect leaks of radioactive liquids in cells, pipe trenches, and sumps.	SR	Part 3 of SR definition
Natural Gas Low Flow Volume	Limits the amount of natural gas that could build up and cause a fire or strong deflagration outside the facility.	PSAR/FSAR	Not a SHINE SSC, natural gas pipelines described in Section 2.2.3.1.2.1 of PSAR
Noble Gas Removal System (NGRS)	Provides confinement for fission product gases and is interlocked to maintain a post-irradiation 40 day hold-up time.	SR	Part 3 of SR definition
Noble Gas Removal System (NGRS) Tank Location	The NGRS storage tank is isolated away from areas where deflagration potentially could occur.	SR	Part 3 of SR definition (tank and cell providing isolation are SR)

Control	Safety Function	SHINE Classification	Classification Basis
Positive Displacement Pump (PDP) Design / Relief System	Includes design features needed to ensure pressure relief to protect against overpressure from pump deadheading. If overpressure occurs, target solution or fissile solution could spill and result in a criticality.	SR	Part 4 of SR definition
Primary Closed Loop Cooling System (PCLS) (integrity)	A closed cooling loop that circulates cooling water through twelve vertical cooling tubes that penetrate the TSV to remove heat generated during normal irradiation and shut down operations. This system reduces the likelihood of an excess reactivity insertion by preventing excessive cooldown or pressurization of the TSV contents. This system provides a barrier separation between the TSV and cooling water.	NSR/SR	System external to TSV not relied on to perform a safety function per SR definition; PCLS tubes (part of TSV construction) are SR per Part 3 of SR definition
Process Tanks and Piping	Process tanks and piping provide a confinement function for process solutions (liquids), including high activity process solutions. Process piping also provides a confinement function for gases. Some tanks also provide a confinement function for gases (e.g., NGRS holdup tank), while other tanks that are continuously vented provide a confinement function for gases (e.g., Raffinate Hold Tank, Recycle UN Hold Tank).	SR	Part 3 of SR definition
Process Vessel Vent System (PVVS)	Provides confinement and venting of process gases. Prevents the accumulation of hydrogen and supports volatile fission product handling. Unless it can be demonstrated that the ductwork remains intact from thermal insult in all postulated fire scenarios, it will be necessary to provide for a filtered vent release to achieve adequate risk reduction.	SR	Part 3 of SR definition
Radiation Area Monitoring System	Provides area and process monitoring to alert personnel to high background radiation levels.	SR	Part 3 of SR definition
Radiation Monitoring System in Irradiation Facility	Provides area and process monitoring to alert personnel to leakage in cells and pipe trenches. Provides early detection and alarms for operator actions. The system also is interlocked to the bubble dampers.	SR	Part 3 of SR definition
Reverse Flow Indication and Alarm	Provides indication and alarm for reverse flow conditions.	SR	Part 3 of SR definition
Robust Tanks and Vessels (RPF)	Metal construction resistant to postulated fire, provides confinement and fire protection.	SR	Part 3 of SR definition
Safe Geometry Overflow System	This system ensures that vessels won't discharge excess solution into inappropriate areas or systems (e.g., into the PVVS). The overflow system is geometrically safe.	SR	Part 4 of SR definition
Safe Geometry Sumps and Berms	Provide a geometrically safe sump and berm configuration to prevent criticality in the event of a leak or rupture of process equipment. Seismically qualified.	SR	Part 4 of SR definition

Control	Safety Function	SHINE Classification	Classification Basis
Shielded Pipe Trenches	An important safety function of this control is to provide radiation shielding for pipes that contain irradiated target solution. This control includes the pipe trench covers and the floor seals around them. In the event of a pipe leak or pipe break, the pipe trench, together with the associated trench covers and seals, provides a confinement function for discharged liquids. The pipe trenches have intermediate barriers such that liquid from a pipe leak will not spread the entire length of the affected pipe trench. While the pipe trenches are not ventilated, they do have leakage pathways into the Zone 2 ventilation system. The pipe trenches also have leak detection instrumentation (separate Defense-in-Depth (DID) control).	SR	Part 3 of SR definition
Solvent Residence Time	The elapsed time that solvent may be used in contact with nitric acid or nitrates is limited, thereby helping to prevent the accumulation of unstable compounds.	TS AC	Not an SSC, but related to Part 4 of SR definition
Solvent Wash Solution Quality Control	Helps to assure that the Solvent Wash Solution is maintained sufficiently free of reactive species so that it effectively removes such contaminants from the loaded Strip Solution.	TS AC	Not an SSC, but related to Part 4 of SR definition
Subcritical Assembly System (SCAS)	Contains TSV, Neutron Multiplier, Subcritical Assembly Support Structure (SASS), and TSV Dump Tanks. Does not include Tritium TCAP System, nor does it include PCLS or TSV Pool and associated cooling loop. Tritium target chamber and Deuterium source bottle are part of Neutron Driver System. Solution leaves Subcritical Assembly when it passes isolation valve on dump tank for transfer to Recovery Operations. Sealed support structure isolates TSV pressure boundary from light water pool. Metal construction of components containing target solution resistant to postulated fire, provides confinement and fire protection.	SR	Parts 1 and 2 of SR definition
Target Solution Hold Tank Design	The target solution (TS) hold tank is located below the TSV. Metal construction of tank is resistant to postulated fire, provides confinement. Also, the hold tank has a limited volume (~110% of the TSV volume).	SR	Part 1 of SR definition
Thermal Denitrator Vent	Helps to protect against the possibility of a red oil explosion. The vent helps to protect the vessel from overpressure, and also provides evaporative cooling to keep red oil from reaching the runaway temperature.	SR	Part 3 of SR definition
Raffinate Hold Tank Instrumentation	Prevent or mitigate a raffinate spill due to overflow from 1-UNCS-05T (Raffinate Hold Tank). This control will prevent the transfer of fissile material to an unsafe geometry tank downstream.	SR	Part 4 of SR definition
Tritium Purification System	Represents integrity of the tritium purification system with regard to the confinement function.	SR	Part 3 of SR definition
Tritium Recovery Glovebox	Provides confinement for the tritium recovery system.	SR	Part 3 of SR definition

Control	Safety Function	SHINE Classification	Classification Basis
TSV Dump Tank Design	The TSV dump tank is located below the TSV. The dump tank only accepts fluids from the TSV during a trip or if an operator takes action to drain the TSV following an irradiation cycle. Metal construction of the tank provides confinement and is resistant to fire. Also, the dump tank is geometrically safe.	SR	Part 2 of SR definition
TSV Fill Valve Interlock	Ensures that material cannot be added to the TSV during operations. Isolates the TSV once it is filled and ready for irradiation operations. Interlocked with mode switch (prevents valve opening while neutron generator is on). Valves also will be de-energized and administratively locked shut.	SR	Part 2 of SR definition
TSV Fill Valves and Piping Sizing	Controls TSV fill rate which in turn provides a means of reactivity control and reduces the likelihood of an excess reactivity insertion.	SR	Part 4 of SR definition
TSV Instrumentation	The TSV instrumentation will provide neutron flux detection and other monitoring services that will lead to shut-off of the accelerator and activation of the TSV dump valve. This instrumentation also supports TSV reactivity protection system (TRPS) control.	SR	Part 2 of SR definition
TSV Integrity	Provides confinement. Metal construction is resistant to postulated fire. Designed to withstand overpressures created by deflagration or detonation.	SR	Part 1 of SR definition
TSV Off-gas System	Provides confinement for gases generated in the TSV. Also prevents hydrogen accumulation and deflagration, given the system's ability to recombine hydrogen and oxygen, and purge. The TOGS is interlocked with TRPS on loss of power, high pressure, or hydrogen accumulation. Redundant blowers. No isolation devices are present between the TSV and TOGS. The TOGS is designed to withstand overpressure conditions. Also, the TOGS is interlocked to maintain a specified post-irradiation hold-up time for gases generated in the TSV.	SR	Part 3 of SR definition
TSV Overflow Line	Limits fluid level in TSV which in turn provides a means of reactivity control and reduces the likelihood of an excess reactivity insertion. Potential overflow into the TSV is routed into the dump tank.	SR	Part 4 of SR definition
TSV Reactivity Protection System (TRPS)	Provides reactivity control for TSV by monitoring multiple parameters, for example n-flux and solution temperature. Off-gas system is interlocked with TRPS on loss of power, high pressure, or hydrogen accumulation. Also will trip on TSV level during startup/fill operations. TRPS interlocked with neutron driver and with the TSV dump valves.	SR	Part 2 of SR definition
Uranyl Nitrate Evaporator Vessel Vent	Helps to protect against the possibility of a red oil explosion. The vent helps to protect the vessel from overpressure, and also provides evaporative cooling to keep red oil from reaching the runaway temperature.	SR	Part 3 of SR definition
Underground Vaults with Covers (RPF)	Protects high inventory tanks from fires and provides spill confinement. Provides fire barriers of at least 2 hours for tanks and vessels.	SR	Part 3 of SR definition

Control	Safety Function	SHINE Classification	Classification Basis
Uninterruptable Power Supply (UPS)	Provides backup power to the TSV Off-gas System for a minimum of two (TBD) hours. Also provides backup power for other safety-related controls. UPS units are redundant. Also referred to as backup electrical power.	SR	Part 3 of SR definition
Ventilation System (Passive Function)	Integrity of ventilation system in fires maintains isolation provided by bubble tight isolation dampers.	SR	Part 3 of SR definition
Solvent Solution Sampling and Analysis	Helps to assure that potentially unstable degradation products are maintained at safe levels.	TS AC	Not an SSC, but related to Part 3 of SR definition
Waste Tank Integrity	Provides confinement of liquid waste.	SR	Part 3 of SR definition
Zone 1 Ventilation	Receives discharge from TOGS and PVVS and provides confinement within the hot cells in the event of a release of fission products outside primary confinement vessels, including TSV, various process vessels, and off-gas system. This control provides confinement for the fission product particulates and iodine that will be present in the irradiation cells and hot cells, as well as conditioning of these gases (particulates and halogens only) in the event of a potential release. Air changes prevent accumulation of hydrogen in irradiation cells. Also provides fire protection by establishing a fire barrier to the irradiation cell penetrations through the ductwork (welded). Includes ductwork and dampers.	SR	Part 3 of SR definition
Zone 2 Ventilation	Ventilation of potentially contaminated areas in the RCA and at the access faces of hot cell interior interfaces. The primary safety function of this control is to provide confinement for the fission product gases that pass out of the Zone 1 ventilation system. The Zone 2 ventilation system includes a series arrangement of two high efficiency particulate air (HEPA) filter banks with a carbon bed positioned between the filter banks.	SR	Part 3 of SR definition

## **RAI 13b.1-2**

*The ISG Augmenting NUREG-1537, Part 2, Section 13b.1.1, "Operations Conducted Outside of the Reactor," states that "[t]he 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."*

*Several sections of SHINE PSAR, specifically in Chapters 1, 3, 4, 6, 9, and 13, contain information regarding radiological hazards, chemical hazards, and facility hazards. Chapter 9 indicates that there are nearly 400 accident scenarios, but Sections 13b.1 and 13b.2 describes only 6 accident sequences in the Radioisotope Production Facility (RPF).*

*The accident analysis describes a few example accident scenarios that the SHINE PSAR states are bounding, but does not describe all accident scenarios that could result in high or intermediate consequences. In order for NRC staff to determine the adequacy of SHINE's accident analysis, additional information is needed on all accident scenarios that could result in high or intermediate consequences, including a designation of the IROFS that prevent or mitigate consequences. Similarly, information is also needed that describes all accident sequences that could result in high or intermediate chemical consequences, including an identification of the SSCs provided for their prevention and mitigation.*

*Additionally, SHINE PSAR, Section 13b.2, "Analyses of Accidents with Radiological Consequences," states that active engineered controls are fail-safe; however, the application does not describe the accident sequences or active engineered controls in sufficient detail for staff to confirm that actuation of the controls are not necessary for them to perform their safety function.*

- a) *Provide the consequence and likelihood of each potential accident sequence, and the methods used to determine the consequences and likelihoods.*

*Additionally, provide each of the IROFS in each accident scenario, and describe each engineered IROFS' safety function and its availability and reliability to perform that safety function when needed, including any engineered IROFS that will be the sole item preventing or mitigating a high or intermediate consequence accident. Describe all passive engineered and active engineered IROFS that prevent or mitigate the accident scenarios with high or intermediate consequences. For accident sequences involving chemical consequences, identify the SSCs provided for the prevention and mitigation of the accident sequence.*

- b) *Provide the basis for asserting that active engineered controls are fail-safe.*

## **SHINE Response**

- a) The consequence and likelihood of each potential accident sequence is provided in Table 13b.1-1-4 of the SHINE Response to RAI 13b.1-1. The methods used to determine the consequences and likelihoods are provided in Part b of the SHINE Response to RAI 13b.1-1.

The IROFS, including passive engineered and active engineered IROFS, identified in the SHINE ISA for each accident scenario are provided in Table 13b.1-2-1. Chemical consequences involving licensed material are prevented or mitigated by licensing basis

administrative items and SSCs as listed in Table 13b.1-2-1. There are no sole IROFS for the SHINE facility. The safety function for each IROFS is described in Table 13b.1-1-5, provided in the SHINE Response to RAI 13b.1-1.

Part a of the SHINE Response to RAI 3.5-1 (Reference 4) describes the SHINE process to designate SSCs important to safety that can be efficiently managed in the future licensed plant by integrating the ISA analysis results into the PSAR (FSAR). This process eliminates the IROFS designation from the PSAR and SHINE QAPD. Thus, Table 13b.1-1-5 also contains the new classification of each IROFS and the basis for that classification.

Safety-related SSCs at the SHINE facility are maintained available and reliable using TS limiting condition for operation (LCO) and surveillance requirements, programs contained in the Administrative Controls section of the TS, and the maintenance, conduct of operations, configuration management, and quality assurance programs. Implementing procedures ensure programmatic and TS surveillance requirements are met. Changes to programs and procedures are evaluated under 10 CFR 50.59 to determine if NRC approval is required prior to making a change. Prior NRC approval is required for changes to the TS.

**Table 13b.1-2-1: Accident Sequences and Associated IROFS**

Initiating Event Scenario	IROFS	Function
<b>Aircraft Accident and Seismic Event</b>		
1. Aircraft impact into IF area causes multiple adverse conditions (e.g., fire, deflagration, confinement failure), resulting in widespread dispersal of radioactive and hazardous materials	Facility Structure Irradiation Unit (integrity, confinement)	Preventor Preventor
2. Aircraft impact into RPF causes multiple adverse conditions (e.g., fire, deflagration, confinement failure), resulting in widespread dispersal of radioactive and hazardous materials	Facility Structure Hot Cell Structure Underground Vaults with Covers (RPF)	Preventor Preventor Preventor
3. Seismic event causes multiple adverse conditions (e.g., fire, deflagration), resulting in widespread dispersal of radioactive and hazardous materials	Facility Structure TSV Integrity (provides confinement) Irradiation Unit (integrity, confinement) Process Tanks and Piping (confinement) Robust Tanks and Vessels (RPF) Hot Cell Structure Underground Vaults with Covers (RPF) High Level Waste Containers Low Level Waste Containers Fire Rating for RPF Areas Fire Rating for Irradiation Cell Underground Vaults with Covers (RPF) Facility Structure Zone 1 Ventilation (fire barrier) Zone 2 Ventilation	Preventor Preventor Preventor Preventor Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
<b>Hazardous Chemicals</b>		
1. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* inside vaults or cells (includes associated valves, piping, and overflow lines) due to operational mechanical failures, human errors, or natural phenomena events (e.g., leaks, rupture)	Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 7, 9, 12, 13, 15, 16, and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Number 2, 3, and 4.	
* With the potential for exceeding ERPG-2 limits at site boundary		



Initiating Event Scenario	IROFS	Function
<p>2. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* outside vaults or cells (includes associated valves, piping, and overflow lines) due to operational mechanical failures, human errors, or natural phenomena events (e.g., leaks, rupture)</p> <p>* With the potential for exceeding ERPG-2 limits at site boundary</p>	No consequence involving Licensed Material	
<p>3. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* inside vaults or cells (includes associated valves, piping, and overflow lines) due to fires.</p> <p>* With the potential for exceeding ERPG-2 limits at site boundary</p>	Potential consequences involving licensed material are described in scenarios for fires in IF and RPF.	
<p>4. Failure of tanks and/or vessels with significant quantities of hazardous toxic chemicals* outside vaults or cells (includes associated valves, piping, and overflow lines) due to fires.</p> <p>* With the potential for exceeding ERPG-2 limits at site boundary</p>	Potential consequences involving licensed material are described in Fires in IF and RPF scenarios 2A, 1B, 2B, 6B, 7B	
<p>5. Exothermic reactions between chemicals leading to damage to tanks or vessels containing significant quantities of hazardous toxic materials in the RPF. (Note: chemicals are segregated from the IF. No chemical mixing is performed in the IF.)</p> <p>* With the potential for exceeding ERPG-2 limits at site boundary</p>	Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 7, 9, 12, 13, 15, 16, and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Number 2, 3, and 4.	
<p>6. Mishap during handling of chemicals leads to breach or spill of chemicals from tanks or vessels</p>	Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 8, 11, and 14 and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Numbers 1 and 5.	
<p>7. Mishap during handling of chemicals leads to a minor spill of chemicals outside tanks or vessels</p>	Potential consequences involving licensed material are described in Loss of Confinement Scenarios Number 8, 11, and 14 and in Mishandling or Malfunction of Target Solution or Confinement Scenarios Numbers 1 and 5.	
<p>8. Mishap (e.g., spill) during delivery of hazardous chemicals outside the facility. (Assumes correct chemical is being delivered)</p>	No consequence involving Licensed Material	

Initiating Event Scenario	IROFS	Function
<b>Criticality</b>		
1. Seismic event causes addition of excess reactivity in TSV or process area equipment (e.g., change in geometry, addition of uranium), resulting in inadvertent criticality	Facility Structure TSV Integrity Irradiation Unit (integrity) Process Tanks and Piping Robust Tanks and Vessels (RPF) Safe Geometry Sumps and Berms Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
2. Overfill of TSV results in inadvertent criticality	TSV Fill Valve Interlock TSV Fill Valves and Piping Sizing TSV Overflow Line TSV Reactivity Protection System (TRPS) TSV Instrumentation Irradiation Unit (shielding) Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
3. Uranium solution with elevated concentration or enrichment is introduced into TSV target solution, resulting in an inadvertent criticality	Fissile Concentration Limits Material Receipt Conduct of Operations Program Analysis of Feed Material TSV Reactivity Protection System (TRPS) Irradiation Unit (shielding) Facility Shielding Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
4. Uranium bearing particles transferred from the Uranyl Sulfate Preparation Tank (1-TSPS-01T) to the Target Solution Hold Tank (1-TSPS-03T) as a result of incomplete dissolution of UO <sub>3</sub> or U causes inadvertent criticality	Filters (Downstream of 1-TSPS-01P and 1-TSPS-02P) Conduct of Operations Program Filters (Downstream of 1-TSPS-01P and 1-TSPS-02P) Differential Pressure Monitors Irradiation Unit (shielding) Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
5. Loss of uranium solution from TSV or processing equipment (e.g., vessel fails), resulting in the accumulation of target solution in an unfavorable geometry, causing an inadvertent criticality	Moisture-Leak Detection /Instrumentation and Alarm (Rad) Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment Safe Geometry Overflow System Safe Geometry Sumps and Berms Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
6. Positive displacement pump operates while downstream process line is blocked, causing over-pressurization of process components containing fissile solution, resulting in a criticality	Positive Displacement Pump Design (overpressure protection) Moisture-Leak Detection/Instrumentation and Alarm (Rad) Safe Geometry Sumps and Berms Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
7. H <sub>2</sub> SO <sub>4</sub> supply tank accidentally charged with HNO <sub>3</sub> . HNO <sub>3</sub> then used as the solvent in U Metal Dissolution Tank, subsequently resulting in a criticality in the U Metal Dissolution Tank or other vessels (e.g., Target Solution Hold Tank, TSV).	Chemical Concentration Limits Chemical Analysis of Feed Solutions Conduct of Operations Program Criticality Safety Program TSV Reactivity Protection System (TRPS) Facility Shielding Criticality Accident Alarm System Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
8. Water entering TSV, as a result of failure of the boundary between them, could have corrosion inhibitors or other components that could promote precipitation or reactivity issues, etc. resulting in a criticality.	Chemical Concentration Limits Chemical Analysis of Feed Solutions Conduct of Operations Program Criticality Safety Program TSV Reactivity Protection System (TRPS) Process Tanks and Piping Facility Shielding Criticality Accident Alarm System Light Water Coolant Activity Monitoring Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
9. Failure of boundary between TSV and light water jacket could result in pressure transients or oscillations.  Uranyl sulfate solution and fission products migrate into light water jacket and are transported to systems supporting the light water jacket. Water entering TSV could have corrosion inhibitors or other components that could promote precipitation or reactivity issues, etc. Hazards include power transient/oscillation and criticality (precipitation), transfer of uranium in light water jacket processing system, etc.	Chemical Concentration Limits Chemical Analysis of Feed Solutions Conduct of Operations Program Criticality Safety Program Process Tanks and Piping Facility Shielding TSV Reactivity Protection System (TRPS) Light Water Coolant Activity Monitoring Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
10. Loss of power to contactors results in uranium and solvent carryover to raffinate tank, resulting in a criticality	Electrical Equipment Design Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment Criticality Accident Alarm System Facility Shielding Raffinate Hold Tank Instrumentation Criticality Safety Program Zone 1 Ventilation Zone 2 Ventilation Facility Siting	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
<b>H2 Deflagration or Detonation</b>		
1. Failure of the TSV Off-gas system due to loss of normal power	TSV Off-gas System (TOGS) Uninterruptable Power Supply (UPS) Irradiation Unit (structural integrity) TSV Off-gas System (TOGS) (integrity) Noble Gas Removal System (NGRS) Tank Location Dampers (Irradiation Cell) (Bubble type) TSV Integrity Radiation Area Monitoring System Conduct of Operations Program	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
2. Failure of the TSV Off-gas system due to loss recombiner capability	TSV Reactivity Protection System (TRPS) TSV Off-gas System (TOGS) Dampers (Irradiation Cell) (Bubble type) TSV Off-gas System (integrity) Hydrogen Monitor (TSV Off-gas) (H2 and pressure accumulation trip signal) Noble Gas Removal System (NGRS) Tank Location Irradiation Unit (structural integrity) TSV Integrity Radiation Area Monitoring System	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
3. Failure of the TSV Off-gas system due to blockage (e.g., plugged zeolite beds, closed dampers or other failure) Hydrogen deflagration overpressurizes system causing release of radioactive solution.	TSV Reactivity Protection System (TRPS) TSV Off-gas System (TOGS) Dampers (Irradiation Cell) (Bubble type) TSV Off-gas System (integrity) Hydrogen Monitor (TSV Off-gas) (H2 and pressure accumulation trip signal) Noble Gas Removal System (NGRS) Tank Location Irradiation Unit (structural integrity) TSV Integrity Radiation Area Monitoring System	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
4. Accumulation of hydrogen and oxygen in the noble gas hold up waste tank(s). Ignition of the flammable mixture fails noble gas waste tank.	Hydrogen Monitor (TSV Off-gas) Hydrogen Monitor (NGRS) Radiation Area Monitoring System Noble Gas Removal System (NGRS) Tank Location	Preventor Preventor Mitigator Mitigator
5. Failure to vent through the Process Vessel Vent System (PVVS in the RPF) results in release of radionuclides to RPF open areas.	TSV Off-gas System (TOGS) Zone 1 Ventilation Zone 2 Ventilation Irradiation Unit (cell confinement) Radiation Monitoring System in Irradiation Facility Radiation Area Monitoring System	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
6. Uranium metal dissolution tank sits idle for extended period (e.g., facility or module stand-down), causing accumulation of radiolysis products in the tank's head	Conduct of Operations Program Zone 1 Ventilation Zone 2 Ventilation	Preventor Mitigator Mitigator
7. Leak in accelerator deuterium supply results in gas ignition and deflagration, with dispersal of nearby radioactive material	Deuterium Source Vessel Inspection Program Process Tanks and Piping Conduct of Operations Program Irradiation Unit	Preventor Preventor Preventor Mitigator
<b>Deflagrations, Detonations, and Explosions for Materials Other Than Hydrogen</b>		
1. Excessive time of process solution in the evaporator, creating increased concentrations and temperatures that promote formation of unstable compounds (e.g., reactions between nitric acid, Tri-Butyl Phosphate (TBP), and related decomposition products) that accumulate over time, resulting in an explosion.	Solvent Residence Time Conduct of Operations Program Process Tanks and Piping Zone 1 Ventilation Zone 2 Ventilation Radiation Area Monitoring System	Preventor Preventor Mitigator Mitigator Mitigator Mitigator
2. Degradation products not removed from Strip Solution en route to Recycle UN Hold Tank. Solutions transferred for processing in the UN Evaporator and Thermal Denitrator resulting in a sudden reaction of unstable species giving rise to a chemical explosion in the UN Evaporator or Thermal Denitrator	Solvent Solution Sampling and Analysis Solvent Residence Time Solvent Wash Quality Control Conduct of Operations Program Thermal Denitrator Vent Uranyl Nitrate Evaporator Vent Hot Cell Integrity Zone 1 Ventilation Zone 2 Ventilation Radiation Area Monitoring System	Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
3. Leak in natural gas supply line outside facility results in gas ignition and deflagration external to the facility, with release of radioactive and hazardous materials.	Natural Gas Low Flow Volume Hot Cell Structure Facility Structure Irradiation Unit Radiation Area Monitoring System	Mitigator Mitigator Mitigator Mitigator Mitigator
<b>Fires in IF and RPF</b>		
<b>(A) Fire in Irradiation Facility (IF) area</b>		
1A. Operational fires where the fire is initiated inside the irradiation cell (irradiation cell is locked and closed).	Combustible Loading Limits Fire Protection Program Irradiation Unit (fire rated) Dampers (Irradiation Cell) (Bubble type) Zone 1 Ventilation Zone 2 Ventilation Subcritical Assembly System (robust, includes TSV, dump/hold up tanks) Process Vessel Vent System (filtered vent release)	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
2A. Maintenance fires (access to irradiation cell is open). Fire may be initiated anywhere.	Combustible Loading Limits Fire Watch During Maintenance Operations Fire Protection Program Radiation Monitoring System in Irradiation Facility Zone 1 Ventilation Zone 2 Ventilation Subcritical Assembly System (robust, includes TSV, dump/hold up tanks) Fire Rating for Irradiation Cell (prevents spread of fire into other cells/fire zones) Conduct of Operations Program	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
3A. Fire propagation from RPF when the irradiation cell is locked and closed (i.e., Initiated in RFP and propagates to Irradiation Cell).	Not credible due to fire rating of irradiation cell	
4A. Fire propagation from RPF (cell door and/or access port open)	(to be covered under RPF fires)	
<b>(B) Fires initiated in RFP</b>		
1B. Operational fires initiated in RPF that affect RPF areas (inside or outside hot cells).	Combustible Loading Limits Underground Vaults with Covers (RPF) Fire Protection Program Conduct of Operations Program Fire Rating for RPF Areas Dampers (RPF) (Bubble type) Robust Tanks and Vessels (RPF) Ventilation System (Passive Function) Zone 1 Ventilation Zone 2 Ventilation	Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
2B. Maintenance and other non-operations fires initiated in RPF (may initiate inside or outside hot cells but propagates to both areas)  Deflagration or detonation not credible within the RPF	Combustible Loading Limits Fire Watch During Maintenance Operations Fire Protection Program Radiation Area Monitoring System Fire Rating for RPF Areas Ventilation System (Passive Function) Robust Tanks and Vessels (RPF) Conduct of Operations Program	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
3B. Fire initiated in RPF (outside the irradiation cell) that propagates to the IF area (irradiation cell locked and closed).	Not credible due to fire rating of irradiation cell	

Initiating Event Scenario	IROFS	Function
4B. Fires initiated in vessels or tanks due to hydrogen accumulation and presence of radiolytic oxygen or air	Process Vessel Vent System Underground Vaults with Covers (RPF) Zone 1 Ventilation Zone 2 Ventilation Radiation Monitoring System in Irradiation Facility Fire Rating for RPF Areas Dampers (RPF) (Bubble type) Ventilation System (Passive Function) Robust Tanks and Vessels (RPF)	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
5B. Vent line to raffinate holding tank is plugged resulting in overfilling the contactors and spill of flammable material resulting in a fire.	Combustible Loading Limits Underground Vaults with Covers (RPF) Fire Protection Program Conduct of Operations Program Raffinate Hold Tank Instrumentation Fire Rating for RPF Areas Dampers (RPF) (Bubble type) Robust Tanks and Vessels (RPF) Ventilation System (Passive Function) Hot Cell Integrity Process Vessel Vent System (PVVS) Safe Geometry Overflow System	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
6B. Lightning strike to facility results in facility fire	Combustible Loading Limits Fire Protection Program Underground Vaults with Covers (RPF) Hot Cell Structure Facility Structure Irradiation Unit Facility Siting Fire Rating for RPF Areas Irradiation Unit (fire rated)	Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator



Initiating Event Scenario	IROFS	Function
7B. Leak in natural gas supply line outside facility results in gas ignition and fire that spreads into facility	Combustible Loading Limits Underground Vaults with Covers (RPF) Fire Protection Program Conduct of Operations Program Fire Rating for RPF (prevents fire spreading into RPF from NSR areas) Irradiation Unit (fire rated) (prevents fire spreading into IF from NSR areas) Fire Rating for RPF Areas Dampers (RPF) (Bubble type) Robust Tanks and Vessels (RPF) Ventilation System (Passive Function) Hot Cell Structure Process Vessel Vent System (PVVS) Facility Structure Irradiation Unit Natural Gas Low Flow Volume	Preventor Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
<b>Mishandling or Malfunction of Target Solution</b>		
1. TSV overfill	Target Solution Hold Tank Design (limited volume) TSV Overflow Line TSV Fill Valve Interlock Radiation Monitoring System in Irradiation Facility (monitors cell) Shielded Pipe Trenches (includes sumps) TSV Dump Tank Design (integrity) TSV Off-gas System Noble Gas Removal System (NGRS) (includes hold up tank) Dampers (Irradiation Cell) (Bubble type) Irradiation Unit (cell confinement)	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
2. TSV or Dump Tank Leak Into the Light Water Pool or primary cooling system	Light Water Pool System (integrity) Primary Closed Loop Cooling System (integrity) TSV Integrity, TSV Dump Tank Design (includes pipes and valves) Radiation Monitoring System in Irradiation Facility (monitors cell and cooling water) Zone 1 Ventilation Dampers (Irradiation Cell) (Bubble type) Irradiation Unit (cell confinement)	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
3. Dump Tank piping Leak Into the Irradiation Cell  Note: No sump in the IU, any leak from the Dump Tank will go to the cooling pool.	TSV Integrity, TSV Dump Tank Design (includes pipes and valves) TSV Dump Tank Design (Dump Tank integrity, includes pipes and valves) Shielded Pipe Trenches (includes sumps) Dampers (Irradiation Cell) (Bubble type) Irradiation Unit (cell confinement) Radiation Monitoring System in Irradiation Facility (monitors cell and cooling water) Zone 1 Ventilation	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
4. Dump Tank piping Leak Into the Irradiation Facility Outside of the Irradiation Hot Cell	Irradiation Unit Isolation Valves TSV Dump Tank Design (Dump Tank integrity, includes pipes and valves) Irradiation Unit (cell confinement) Dampers (Irradiation Cell) (Bubble type) Shielded Pipe Trenches (includes RFP cells confinement) Radiation Monitoring System in Irradiation Facility (monitors cells and trenches) Zone 2 Ventilation	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
5. Misdirection of Irradiated Target Solution Back To the Target Solution Hold Tank	TSV Fill Valve Interlock Target Solution Hold Tank Design (integrity) Shielded Pipe Trenches (includes sumps) TSV Off-gas System Dampers (Irradiation Cell) (Bubble type) Irradiation Unit (cell confinement) Noble Gas Removal System (NGRS) (includes hold up tank) Radiation Monitoring System in Irradiation Facility (monitors cell)	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
<b>Mishandling or Malfunction of Equipment (Off-gas/NGRS)</b>		
1. Inadvertent venting of the Off-gas purge contents	No controls required; LT IC unmitigated consequences	
2. Release of noble gases from hold up tank due to mechanical failures (e.g., leaks, vessel failures)	Process Tanks and Piping (confinement) Noble Gas Removal System (NGRS) (includes hold up tank) Conduct of Operations Program Zone 1 Ventilation Radiation Area Monitoring System	Preventor Preventor Preventor Mitigator Mitigator
3. Release from the Off-gas system due to mechanical failures (e.g., leaks or bypass of the NGRS)	TSV Off-gas System (TOGS) (integrity) Conduct of Operations Program Dampers (Irradiation Cell) (Bubble type) Irradiation Unit (cell confinement) Zone 1 Ventilation Process Tanks and Piping (confinement) Conduct of Operations Program	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
<b>Loss of Confinement</b>		
1. Strong straight winds breach facility and uranium solution processing equipment, with release of radioactive and hazardous materials	Facility Structure TSV Integrity (provides confinement) Irradiation Unit (integrity, confinement) Process Tanks and Piping (confinement) Robust Tanks and Vessels (RPF) High Level Waste Containers Low Level Waste Containers Conduct of Operations Program	Preventor Preventor Preventor Preventor Preventor Preventor Preventor Mitigator
2. Tornado breaches facility and uranium solution processing equipment, with release of hazardous and radioactive materials	Facility Structure TSV Integrity (provides confinement) Irradiation Unit (integrity, confinement) Process Tanks and Piping (confinement) Robust Tanks and Vessels (RPF) High Level Waste Containers Low Level Waste Containers Conduct of Operations Program	Preventor Preventor Preventor Preventor Preventor Preventor Preventor Mitigator
3. Heavy rain/flooding causes damage to facility and process equipment resulting in dispersal of radioactive and hazardous materials	Facility Structure TSV Integrity (provides confinement) Irradiation Unit (integrity, confinement) Process Tanks and Piping (confinement) Robust Tanks and Vessels (RPF) High Level Waste Containers Low Level Waste Containers Conduct of Operations Program	Preventor Preventor Preventor Preventor Preventor Preventor Preventor Mitigator
4. Heavy snow causes damage to facility and process equipment resulting in dispersal of radioactive and hazardous materials	Facility Structure TSV Integrity (provides confinement) Irradiation Unit (integrity, confinement) Process Tanks and Piping (confinement) Robust Tanks and Vessels (RPF) High Level Waste Containers Low Level Waste Containers Conduct of Operations Program	Preventor Preventor Preventor Preventor Preventor Preventor Preventor Mitigator

Initiating Event Scenario	IROFS	Function
5. Vessel (e.g., extraction column) rupture or leak, resulting in discharge of radioactive materials	Process Tanks and Piping (integrity) Robust Tanks and Vessels (RPF) Conduct of Operations Program Facility Structure Hot Cell Integrity Zone 1 Ventilation Zone 2 Ventilation Radiation Monitoring System in Irradiation Facility (monitors cell)	Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
6. Explosion at nearby facility generates a mechanical insult that compromises the integrity of the Off-gas or Noble Gas Removal System, leading to dispersal of radioactive gas	Facility Structure TSV Off-gas System (integrity) Noble Gas Removal System (NGRS) (integrity) Irradiation Unit (integrity) Facility Siting Radiation Area Monitoring System Noble Gas Removal System (NGRS) Tank Location	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator
7. Leak in TSV cooling lines causes flow of TSV coolant into TSV vessel. TSV overfills, sending TSV solution into TSV vent line. TSV solution enters Off-gas system and eventually drains into recombiners, generating steam, which damages process system components.	TSV Overflow Line (level detection) TSV Integrity TSV Off-gas System (integrity) Irradiation Unit (isolation and shielding) Radiation Area Monitoring System Zone 1 Ventilation Zone 2 Ventilation	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
8. Process tank (e.g., Target Solution Hold Tank) overflows into vent system allowing a release pathway for fission products (recognizes a pathway for liquids that was only intended for handling tank vapors)	Safe Geometry Overflow System Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment Radiation Area Monitoring System Zone 1 Ventilation Zone 2 Ventilation Conduct of Operations Program Moisture-Leak Detection /Instrumentation and Alarm (Rad)	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
9. Pressure increases to pump head pressure in Mo Extraction Column, resulting in a spill	Positive Displacement Pump Relief System Column Pressure Monitor Process Tanks and Piping (integrity) Hot Cell Integrity Zone 1 Ventilation	Preventor Preventor Preventor Mitigator Mitigator
10. No flow from TSV Off-gas System to Noble Gas System due to blockage in pathway, resulting in fission gases released through the PVVS	Process Vessel Vent System Radiation Area Monitoring System Conduct of Operations Program	Preventor Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
11. Unanticipated transfer into <sup>99</sup> Mo system (Line 0308) resulting in Mo Extraction System hot cell spill.	Reverse Flow Indication and Alarm Moisture-Leak Detection /Instrumentation and Alarm (Rad) Hot Cell Integrity Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment Zone 1 Ventilation	Preventor Mitigator Mitigator Mitigator Mitigator
12. Positive displacement pump operates while downstream process line is blocked, causing overpressurization of process components containing fissile solution, resulting in a spill in the IF	Positive Displacement Pump Design (e.g., pressure relief) Moisture-Leak Detection/Instrumentation and Alarm (Rad) Irradiation Unit (cell confinement) Dampers (Irradiation Cell) (Bubble type) Radiation Monitoring System in Irradiation Facility (monitors cell) Zone 1 Ventilation Geometrically Safe Configuration for Process Tanks, Pipes, and Other Process Equipment	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
13. Positive displacement pump operates while downstream process line is blocked, causing overpressurization of process components containing fissile solution, resulting in a spill in the radioisotope production facility (RPF)	Positive Displacement Pump Design (e.g., pressure relief) Moisture-Leak Detection/Instrumentation and Alarm (Rad) Hot Cell Integrity Dampers (RPF) (Bubble type) Radiation Area Monitoring System Zone 1 Ventilation	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
14. Sulfuric acid source tank charged with nitric acid, causing uranium oxides to be dissolved in nitric acid, resulting in uranyl nitrate solution being sent to Target Solution Hold Tank with potential for release of NOx	No consequence involving Licensed Materials	
15. Rotating equipment malfunctions, resulting in rupture of solution tank and release of uranium solution	Process Tanks and Piping (integrity) Moisture-Leak Detection/Instrumentation and Alarm (Rad) Hot Cell Integrity Dampers (RPF) (Bubble type) Radiation Area Monitoring System Zone 1 Ventilation	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
16. Failure of liquid waste tank (e.g., corrosion) results in release of radioactive material.	Waste Tank (integrity) Conduct of Operations Program Zone 1 Ventilation Moisture-Leak Detection/Instrumentation and Alarm (Rad) Hot Cell Integrity Dampers (RPF) (Bubble type) Radiation Area Monitoring System	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
<b>Excess Reactivity</b>		
1. Pressurization of target solution fluid in TSV	TSV Off-gas System TSV Reactivity Protection System (TRPS) Zone 1 Ventilation	Preventor Mitigator Mitigator
2. Excessive cool down of target solution fluid in TSV	TSV Reactivity Protection System (TRPS) TSV Dump Tank Design	Preventor Mitigator
3. Moderator addition in TSV due to cooling system malfunction (e.g. cooling tube rupture)  Note: Cool down below 20°C a concern, but not credible. Potential for in-leakage into the TSV  Note: Inherent properties of the irradiation solution will lead to a reduction in reactivity	Zone 1 Ventilation Radiation Monitoring System in Irradiation Facility (monitors cell) Zone 2 Ventilation	Mitigator Mitigator Mitigator
4. Target solution injection leads to excessive level in TSV	TSV Fill Valve Interlock TSV Reactivity Protection System (TRPS) (trip on TSV level during startup/fill operations) Manual TSV Trip Switch Zone 1 Ventilation Zone 2 Ventilation Radiation Area Monitoring System Criticality Safety Program	Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator
5. Adverse geometry changes in TSV lead to insertion of excess reactivity	Subcritical Assembly System (design integrity to survive all accident environments) TSV Reactivity Protection System (TRPS) Radiation Area Monitoring System Criticality Safety Program	Preventor Mitigator Mitigator Mitigator
6. Moderator lumping effects in the target solution lead to reactivity insertion in TSV	TSV Reactivity Protection System (TRPS) Radiation Area Monitoring System Criticality Safety Program	Mitigator Mitigator Mitigator
7. Inadvertent introduction of other materials into the target solution	TSV Fill Valve Interlock TSV Dump Tank Design TSV Reactivity Protection System (TRPS) Radiation Area Monitoring System Criticality Safety Program	Preventor Preventor Mitigator Mitigator Mitigator
8. Bulk boiling of the target solution	Analysis shows that this scenario is not credible for this system.	

Initiating Event Scenario	IROFS	Function
9. Precipitation of uranium/fission products leads to insertion of excess reactivity in the TSV	Chemistry Control for Irradiation Solution (control hydrogen peroxide) Chemical Concentration Limits Chemical Analysis of Feed Solutions TSV Reactivity Protection System (TRPS) Radiation Area Monitoring System Criticality Safety Program Facility Shielding Irradiation Unit (shielding) Irradiation Unit (isolation) Zone 1 Ventilation Zone 2 Ventilation	Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
10. Excess enrichment of U-235 (>20%) or excess concentration of solution in TSV leads to excess reactivity insertion	Fissile Concentration Limits Material Receipt Conduct of Operations Program Analysis of Feed Material TSV Reactivity Protection System (TRPS) Radiation Area Monitoring System Criticality Safety Program Facility Shielding Irradiation Unit (shielding) Irradiation Unit (isolation) Zone 1 Ventilation Zone 2 Ventilation	Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator Mitigator
<b>Radiation Exposure</b>		
1. Inadvertent operation of accelerator, resulting in radiation exposure to workers	Accelerator Key Switch Accelerator Interlocks Use of Accelerator Audible/Visual Warnings (before activation) Conduct of Operations Program Accelerator Manual Shut-off Use of Accelerator Audible/Visual Warnings (during activation) Accelerator Kill Switch	Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator
2. Misalignment of accelerator beam, resulting in radiation exposure to workers	Accelerator Key Switch Accelerator Interlocks Use of Accelerator Audible/Visual Warnings (before activation) Conduct of Operations Program Facility Shielding Accelerator Manual Shut-off Use of Accelerator Audible/Visual Warnings (during activation) Accelerator Kill Switch	Preventor Preventor Preventor Preventor Preventor Mitigator Mitigator Mitigator

Initiating Event Scenario	IROFS	Function
3. Inadvertent access to irradiation cell during irradiation, resulting in radiation exposure to workers	Use of Accelerator Audible/Visual Warnings Accelerator interlocks Conduct of Operations Program Use of Accelerator Manual Shut-off	Preventor Preventor Preventor Mitigator
4. Mishap during handling of waste container leads to container breach, resulting in worker contamination	High Level Waste Containers Low Level Waste Containers Conduct of Operations Program Radiation Area Monitoring System Conduct of Operations Program	Preventor Preventor Preventor Mitigator Mitigator
5. Loss of hot cell confinement during hot cell operations (e.g., seal failure) results in worker contamination	Hot Cell Integrity Zone 1 Ventilation Zone 2 Ventilation Radiation Area Monitoring System Conduct of Operations Program	Preventor Mitigator Mitigator Mitigator Mitigator
6. Failure in the tritium purification system results in leak of tritium	Tritium Purification System (confinement) Tritium Recovery Glovebox (confin. for Tritium recovery system) Radiation Area Monitoring System Conduct of Operations Program	Preventor Preventor Mitigator Mitigator
7. Inadvertent access to cell during operation, resulting in radiation exposure to workers	Use of Hot Cell Door Audible/Visual Warnings Hot Cells requiring periodic/routine entry door interlock with Internal Radiation Field Hot Cell Door Key Switch Conduct of Operations Program Radiation Area Monitoring System Conduct of Operations Program	Preventor  Preventor Preventor Mitigator Mitigator



- b) Active engineered controls are components which require a motive force (e.g., pressurized air, electrical power) to reposition. Active engineered controls are made fail-safe by virtue of their design, such that upon removal of the motivate force the component immediately and without any additional external force, power, or actuating signal, moves to the position necessary to perform its safety function. For example, a normally-open valve that needs to be closed to perform its safety function can be designated as fail-safe if a motive force is used to keep the valve open and the valve will go closed upon the loss of this motive force. Active safety-related SSCs used by SHINE will follow this fail-safe principle, where the components will fail to their safe position. The specific accident sequences are not relevant because the components are selected to be fail-safe for each application as required.

### **RAI 13b.2**

*The ISG Augmenting NUREG-1537, Part 2, Section 13b.2, Chemical Process Safety for the Radioisotope Production Facility, states that the application should include a chemical process description, chemical accident description, chemical accident consequences, chemical process safety controls, and chemical process surveillance requirements.*

*Additionally, the ISG Augmenting NUREG-1537, Part 2, Section 6b, states that the chemical performance requirements in 10 CFR 70.61(b)(4) and (c)(4) have been found to be acceptable criteria for chemical-related accident sequences.*

*As used in the ISG, the term, “performance requirements” is not intended to mean that the performance requirements of 10 CFR 70.61 are required by regulation, only that their use as accident consequence and likelihood criteria would be found acceptable by NRC staff. Chemical exposure criteria different from those described in this ISG will be acceptable if an adequate basis is provided for the NRC staff to make the determination needed to issue or continue a license.*

*The PSAR application states that exothermic reactions between chemicals stored on site are prevented by segregation and isolation.*

*Identify the incompatible chemicals and identify their storage and use locations in the facility, to demonstrate adequate segregation and isolation.*

### **SHINE Response**

SHINE will use and store acids and bases in large quantities, which if mixed, could cause a highly exothermic reaction. Table 13b.2-1 below provides a list of incompatible chemicals and their use location. As shown in Figure 1.3-2 and Table 13b.3-1 of the PSAR, the bulk quantities of acids will be stored separately from the bases and hydrogen peroxide in separate storage rooms outside the RCA. These chemical storage rooms provide adequate segregation and isolation for the incompatible chemicals.

**Table 13b.2-1: Incompatible Chemicals and Use Location**

<b>Acids Use Location</b>	<b>Bases and Hydrogen Peroxide Use Location</b>
Sulfuric (H <sub>2</sub> SO <sub>4</sub> ) Target Solution prep area	Ammonium Hydroxide (NH <sub>4</sub> OH) Supercells
Nitric (HNO <sub>3</sub> ) Target Solution prep area Target Solution cleanup area	Sodium Hydroxide (NaOH) PVVS cell
Hydrochloric (HCl) Supercells	Calcium Hydroxide (Ca(OH) <sub>2</sub> ) Liquid waste solidification cell
	Hydrogen Peroxide (H <sub>2</sub> O <sub>2</sub> ) Supercells

Diluted forms and preparations for use in the processing facility will be used and controlled using engineered systems (e.g., separate storage tanks and piping) to prevent inadvertent mixing of incompatible chemicals. The chemical processing steps using these chemicals will be engineered with enthalpy changes specifically addressed. Procedures for use of chemicals will be written to prevent hazardous mixtures. Small volumes of chemicals to be used in laboratory settings and in processes in small amounts will be stored and labeled in accordance with their Safety Data Sheets.

The remaining process chemicals listed in Table 13b.3-1 of the PSAR represent small volumes of chemicals. These chemicals will be stored and labeled in accordance with their Safety Data Sheets.

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