

ENCLOSURE 3

SHINE MEDICAL TECHNOLOGIES, LLC

**SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION
SUPPLEMENT NO. 5 AND RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PUBLIC VERSION**

SHINE MEDICAL TECHNOLOGIES, LLC

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION PUBLIC VERSION

The U.S. Nuclear Regulatory Commission (NRC) staff determined that additional information was required (Reference 1) to enable the continued review of the SHINE Medical Technologies, LLC (SHINE) operating license application (Reference 2). The following information is provided by SHINE in response to the NRC staff's request.

Chapter 6 – Engineered Safety Features

RAI 6b.3-1

FSAR, Section 6b.3.1.4, states that an additional penalty of 0.01 was assigned to SHINE's proposed margin of subcriticality (0.05) to account for the limited number of experimental benchmarks available for uranyl sulfate systems in the validation of SHINE's computational method (MCNP5), resulting in a margin of subcriticality of 0.06. However, SHINE FSAR Section 4a2.6.2.6.1, "Uncertainties in K_{eff} Values Relying on MCNP Calculation," assigns a margin of subcriticality of 0.05 to the target solution vessel (TSV) dump tanks despite being subject to the same vulnerabilities as those necessitating an additional penalty of 0.01 and relying on the same computational method, cross-section library, and validation report.

- a. Justify the use of a unique margin of subcriticality (0.05) for the TSV dump tanks, noting that any additional data obtained through 1/M experiments does not contribute to the validation of SHINE's computational method (MCNP5) and its determination of bias and bias uncertainty.
- b. Describe how the use of a unique margin of subcriticality (0.05) for the TSV dump tanks is consistent with American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.24-2017, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," as committed to in FSAR, Section 6b.3.1.3, "Use of National Consensus Standards."

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE's conduct of operations will be based on nuclear criticality safety (NCS) technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

SHINE Response

SHINE applies a subcritical margin of 0.06 to target solution processes, with the exception of the target solution vessel (TSV). Sections 4a2.6 and 4a2.8 of the Final Safety Analysis Report (FSAR) have been revised to describe the application of this subcritical margin to the

TSV dump tank and the TSV off-gas system (TOGS). A mark-up of the FSAR incorporating these changes is provided as Attachment 1. SHINE has also revised Table 5.5.4 of the technical specifications to describe the application of this subcritical margin to the TSV dump tank and the TOGS. A mark-up of the technical specifications incorporating these changes is provided as Attachment 2. SHINE will provide a revision to the technical specifications incorporating the mark-up by February 28, 2021.

RAI 6b.3-2

FSAR, Section 6b.3.1.5, states that if the double contingency principle (DCP) cannot be employed, consideration is given to the use of ANSI/ANS-8.10-2015, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement," to allow single-contingency operations or mitigation of consequences. FSAR, Section 6b.3.1.3, states that SHINE is committed to following ANSI/ANS-8.10-2015, as endorsed by RG-3.71.

To understand if there are circumstances where the DCP cannot be employed, the applicant should identify any such cases in the license application in which the DCP is not practicable and should provide justification as to why the affected processes are acceptably safe.

- a. Describe how ANSI/ANS-8.10 would be implemented in the event that the DCP cannot be employed given that ANSI/ANS-8.10, Paragraph 4.1(a), states that the provisions of the standard may only be applied in facilities where all operations involving fissionable materials are conducted remotely by personnel located outside of the shielded area.
- b. RG-3.71 includes a clarification to ANSI/ANS-8.10 that the dose limits for an intermediate consequence event in 10 CFR Section 70.61, "Performance Requirements," may be used in lieu of the dose limits specified in Section 4.2.1 of the standard. State which dose limits would be applied if ANSI/ANS-8.10 were implemented.
- c. Discuss any instances in which the DCP is not practicable at the SHINE facility. Provide a justification for any such instance.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

SHINE Response

There are no instances in the SHINE facility where the double contingency principle is not practicable. Subsections 6b.3.1.3 and 6b.3.1.5 of the FSAR have been revised to remove the commitment to the use of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.10-2015, Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement (Reference 3). Subsection 6b.3.1.4 of the FSAR has been revised to state that, where the double contingency principle is employed, the nuclear criticality safety evaluation contains a description of its implementation. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 6b.3-3

FSAR, Section 6b.3.1.5, states that processes within the radioisotope production facility (RPF) generally comply with the DCP. FSAR Section 6b.3.1.3 states that SHINE commits to ANSI/ANS 8.1-2014, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," which includes adherence to the DCP. However, SHINE does not define the terms "unlikely" or "credible" as they apply to the DCP.

Define the terms "unlikely" and "credible" as they relate to the DCP.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

SHINE Response

While the term "credible" does not appear in the establishment of the double contingency principle, it is closely related as part of the process analysis requirement in ANSI/ANS-8.1-2014, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 4). As discussed in Appendix B of ANSI/ANS-8.1-2014, the term "credible" relies on the judgement of key professionals involved in the evaluation process. Consequently, SHINE employs a multi-disciplinary approach in the evaluation process which includes key stakeholders from operations, design engineering, and safety analysis to determine the set of credible process upsets for each evaluated process. The nuclear criticality safety evaluation process utilizes appropriate hazard evaluation techniques (what-if/checklist and event tree analysis) to aid in the determination of credible process upsets. Process upsets are considered credible unless they are not physically possible or are caused by a sequence of events involving many unlikely human actions or errors for which there is no reason or motive.

Because the nuclear criticality safety evaluation process does not rely on quantitative analysis to determine the likelihood of potential process upsets, the term "unlikely" is used in its plain meaning as interpreted by the key stakeholders involved in the evaluation process. For each identified process upset, the conditions of the system, its construction, and the event sequence are considered by the evaluators when determining whether the process upset can qualitatively be considered unlikely. The independence of potential simultaneous process upsets is similarly considered as part of the evaluation process to ensure that the process analysis requirement is satisfied and the application of the double contingency principle is adequate.

The application of this approach is addressed in Subsection 6b.3.1.4 of the FSAR, which states, in part, "identified scenarios are screened based on a qualitative determination of likelihood." Nuclear criticality safety evaluations are documented with sufficient detail, clarity, and lack of ambiguity to allow independent judgment of results by personnel familiar with the physics of nuclear criticality, the facility operations, and the associated nuclear criticality safety practices. To support this programmatic requirement, nuclear criticality safety evaluations contain explicit descriptions of the implementation of the double contingency principle, as described in Subsection 6b.3.1.5 of the FSAR.

RAI 6b.3-4

FSAR, Section 6b.3.1.8, states that facility procedures include provisions for rapid evaluation of the significance of NCS events, including immediate notifications of facility NCS staff and the assessment of events with respect to the loss or degradation of double contingency protection. The DCP is primarily a design principle as opposed to a representation of a state of existence. Therefore, the determination of whether a report to the NRC is required should be based on the likelihood of inadvertent criticality, not whether double contingency protection was maintained.

- a. Describe the method in which NCS events are evaluated for whether a report to the NRC is required. Discuss whether SHINE intends to commit to the reporting requirements of 10 CFR 70, Appendix A for NCS events.
- b. State whether SHINE commits to meeting the requirements of 10 CFR 70.50 and 70.52 with respect to reporting the occurrence of inadvertent criticality to the NRC.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

- a. SHINE administrative procedures describe the response to anomalous conditions involving criticality safety, including the evaluation of anomalous conditions by nuclear criticality safety (NCS) staff. SHINE administrative procedures also direct NCS staff to consult the applicable nuclear criticality safety evaluation(s) to determine the significance of the event, and to coordinate with Licensing and Operations personnel to ensure appropriate reporting requirements are met.

Subsection 6b.3.1.8 of the FSAR has been revised to include a commitment to the reporting requirements of 10 CFR Part 70, Appendix A, for NCS events. A mark-up of the FSAR incorporating these changes is provided as Attachment 1. SHINE has also revised Section 5.8 of the technical specifications to include the applicable reporting requirements of 10 CFR Part 70, Appendix A in the Administrative Controls section of the technical specifications. A mark-up of the technical specifications incorporating these changes is provided as Attachment 2. SHINE will provide a revision to the technical specifications incorporating the mark-up by February 28, 2021.

- b. Subsection 6b.3.1.8 of the FSAR has been revised to include a commitment to the reporting requirements of 10 CFR 70.50 and 10 CFR 70.52. A mark-up of the FSAR incorporating these changes is provided as Attachment 1. SHINE has also revised Section 5.8 of the technical specifications to include the reporting requirements of 10 CFR 70.50 and 10 CFR 70.52 in the Administrative Controls section of the technical specifications. A mark-up of the technical specifications incorporating these changes is provided as Attachment 2. SHINE will provide a revision to the technical specifications incorporating the mark-up by February 28, 2021.

RAI 6b.3-5

The SSA states that reliability management measures are considered to be programmatic administrative controls. FSAR, Section 12.5, provides actions regarding reporting safety limit violations and occurrences requiring special reports other than a safety limit violation to the NRC, including observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to operations. However, it is not clear whether failures and/or degradations to reliability management measures that negatively impact a control's ability to perform its intended safety function would be reported to the NRC.

Discuss how failures and degradations of reliability management measures are assessed with respect to reporting to the NRC.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

Reliability management measures, discussed as programmatic administrative controls in the SHINE safety analysis (SSA), are referenced within Section 5.0 of the technical specifications.

Reporting requirements for SHINE are described in both Section 12.5 of the FSAR and in Section 5.8 of the technical specifications. Section 5.8.2 of the technical specifications describes the special reports that are required to be submitted by SHINE to the NRC. Item h. of this section specifies that reports are required upon an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to operations.

This reporting requirement applies to the administrative controls described in the technical specifications, including the nuclear criticality safety program described in Section 5.5.7 of the technical specifications and the other programmatic administrative controls referenced throughout Section 5.0.

Therefore, failures or degradations to reliability management measures (i.e., inadequacies in the implementation of administrative controls described in Section 5.0 of the technical specifications) that negatively impact a control's ability to perform its intended safety function (i.e., that cause or could have caused the existence or development of an unsafe condition with regard to operations) would be reported to the NRC in accordance with Section 5.8.2 of the technical specifications.

RAI 6b.3-6

FSAR, Section 6b.3.1.4, states that criticality safety limits are derived based on assuming optimum or most-reactive *credible* parameter values unless specific controls are implemented to limit parameters to a particular range. The term "credible" is defined in the SHINE SSA, as it relates to abnormal conditions, changes in process conditions, and accident sequences.

However, the definition provided does not address the use of the term “credible” as it relates to criticality safety parameter values.

Describe how the credibility of a most-reactive credible parameter value would be identified and how its credibility would be assessed.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

As discussed in Appendix B of ANSI/ANS-8.1-2014 (Reference 4), the term “credible” relies on the judgement of key professionals involved in the evaluation process. Consequently, SHINE employs a multi-disciplinary approach in the evaluation process which includes key stakeholders from operations, design engineering, and safety analysis to determine the set of credible process conditions and range of credible parameter values.

The nuclear criticality safety evaluation process utilizes appropriate hazard evaluation techniques (what-if/checklist and event tree analysis) to aid in the determination of credible process conditions. The range of each evaluated parameter is not constrained unless some values are not physically possible or are caused by a sequence of events involving many unlikely human actions or errors for which there is no reason or motive.

The value of a parameter which provides the highest multiplication factor (k_{eff}) is the most reactive value of that parameter. In combination with the judgment of the key professionals performing a nuclear criticality safety evaluation regarding which values are “credible,” the “most-reactive credible” value of a parameter is the value which lies within the range of credible values and has the highest multiplication factor.

RAI 6b.3-7

FSAR, Section 6b.3.1.3, states that SHINE commits to following ANSI/ANS-8.7-1998, “Nuclear Criticality Safety in the Storage of Fissile Materials,” as endorsed by Regulatory Guide (RG)-3.71, “Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores.” The staff reviewed a select sample of nuclear criticality safety evaluations (NCSEs) to evaluate the adequacy of SHINE’s criticality safety program commitments and identified apparent inconsistencies with certain aspects of ANSI/ANS-8.7. ANSI/ANS-8.7, Paragraph 1, states that the standard cannot effectively cover all conditions of interest with respect to the storage of fissile material, and for this reason supplementary information is encouraged. ANSI/ANS-8.7, Paragraph 4.2.1, states that limits for the storage of fissile material shall be based on experimental data or on the results of calculations made through the use of validated computational techniques.

State how the design of the uranium receipt and storage system (URSS) storage rack meets ANSI/ANS-8.7-1998, as endorsed by RG-3.71. Explicitly discuss the experimental data or calculations performed using a validated computational technique used in its design.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

SHINE Response

ANSI/ANS-8.7-1998, Nuclear Criticality Safety in the Storage of Fissile Materials (Reference 5), provides mass limits for a variety of array configurations for the storage of uranium. The mass limits provided in ANSI/ANS-8.7-1998 are based on validated computational methods. Section 5 of ANSI/ANS-8.7-1998 states that the limits provided in the tables may be applied directly to the solution of practical storage problems. Section 5 of ANSI/ANS-8.7-1998 also describes the method by which the limits were derived.

SHINE directly applies the limits provided in ANSI/ANS-8.7-1998 to the design of the uranium receipt and storage system (URSS) storage rack, and observes the restrictions identified in the standard without exception.

RAI 6b.3-8

FSAR, Section 6b.3.1.3, states that SHINE commits to following ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors," and that SHINE does not use soluble neutron absorbers as a means of criticality control and therefore does not commit to ANSI/ANS 8.14 2004, "Use of Soluble Absorbers in Nuclear Facilities Outside Reactors." These statements suggest that SHINE may use fixed neutron absorbers but not soluble. However, FSAR, Section 6b.3.2, does not include the use of either type of neutron absorber (fixed or soluble) as a means of criticality control. Additionally, no commitments or discussion is provided regarding the use of neutron absorbers as a means of criticality control.

State whether SHINE uses, or plans to use, neutron absorbers as a means of criticality control. If either type of absorber (fixed or soluble) is used, state SHINE's specific commitments regarding their use as a means of criticality control.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely.

SHINE Response

Fixed or soluble neutron absorbers are not used or planned to be used as a means of criticality control at the SHINE facility. Subsection 6b.3.1.3 of the FSAR has been revised to remove the commitment to the use of ANSI/ANS-8.21-1995, Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors (Reference 6). A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 6b.3-9

FSAR, Section 6b.3.2, states that the correlation of process variables to an associated controlled parameter is established by experiment or plant-specific measurements. However, no details are provided as to how this commitment would be implemented.

Explain how this commitment would be implemented for correlations established by both experiment and plant-specific methods. Explain how it is assured that inappropriate reliance is not placed on as found conditions or on process assumptions and characteristics that are not controlled.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

Correlation of process variables to controlled parameters is accomplished by direct or indirect measurements of plant equipment, plant processes, or experiments used to determine such a correlation, including consideration of associated uncertainty of the measurement technique. Direct correlations apply when monitored process variables are directly measured. Indirect correlations may be established when direct measurements of controls are not possible and may be developed using known empirical or theoretical relationships or developed through plant-specific data collection. Correlations are documented in accordance with SHINE procedures.

Correlation inputs will be defined and documented for a specific controlled parameter, including process assumptions and characteristics. A discrepant as-found condition or discovery of process conditions beyond what is assumed or controlled in a nuclear criticality safety evaluation will be identified as a nonconformance, and evaluated as a change (modification, addition to, or removal from the documented correlation) under the 10 CFR 50.59 process. Appropriate immediate actions will be taken to prevent criticality until safety can be assured. Such conditions will be evaluated for reportability, and if required, appropriate reports to the NRC will be made.

RAI 6b.3-10

SHINE document NCSE-2018-0011, Section 4.1.1, "Subcritical Mass Limits," states that the subcritical mass limits for operations in the Quality Control and Analytical Testing Laboratories (LABS) were derived based on the single parameter limits (SPLs) from ANSI/ANS-8.1. However, the SPLs used appear to be that of a material composition inconsistent with, and potentially nonconservative of, the materials associated with LABS operations.

- a. Describe the methodology used to determine whether a composition-specific SPL may be used to establish NCS limits for another material composition.

Provide a justification for applying the SPLs of a material composition other than those associated with a specific process.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have the capability to perform adequate safety analyses of all production processes that will be conducted in the facility.

SHINE Response

The SHINE Response to RAI 6b.3-10 will be provided by January 31, 2021.

RAI 6b.3-11

FSAR, Section 6b.3.1.4, "Nuclear Criticality Safety Evaluations," states that NCSEs are conducted for each fissionable material operation (FMO) within the RPF to ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including the use of an approved margin of subcriticality for safety. However, in SHINE document NCSE-2018-0011, Section 5.1, "Normal Process Conditions," it is not clear how SHINE applies the definition of fissionable material operation as described in the FSAR.

Stating that an operation does not qualify as an FMO, while acknowledging that there are credible accident sequences associated with the operation requiring the implementation of controls to limit its likelihood of occurrence, is inherently contradictory. Additionally, stating that an operation does not qualify as an FMO effectively bypasses SHINE's commitment to perform an NCSE per FSAR, Section 6b.3.1.4.

Clarify the statements in the FSAR regarding what qualifies as an FMO and when an NCSE is required with information from SHINE document 1500-09-01, "Criticality Safety Program," as applicable.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have the capability to perform adequate safety analyses of all production processes that will be conducted in the facility.

SHINE Response

SHINE document NCSE-2018-0011 has been revised to remove its distinction as not a fissionable material operation (FMO), resolving the inherent contradiction. The identified credible process upset for the quality control and analytical testing laboratories (LABS) is that the established mass limit would be exceeded; therefore, operations within the LABS have the potential to exceed 250 grams total uranium, consistent with the definition of an FMO provided in Subsection 6b.3.1.4 of the FSAR.

Use of the definition of an FMO provided in Subsection 6b.3.1.4 of the FSAR provides a screening criterion by which NCS staff can effectively ascertain when to perform a nuclear criticality safety evaluation for specific processes.

RAI 6b.3-12

SHINE document NCSE-2018-0010, "Nuclear Criticality Safety Evaluation of the Radioactive Liquid Waste Immobilization System (RLWIS)," Section 4.1.1, "Subcritical Limits Uranyl Sulfate," provides a methodology to derive the limits for uranium concentration. However, the proposed methodology is imprecise and does not necessarily provide an adequate demonstration that RLWIS operations are below the appropriate upper subcritical limit.

Provide a justification for using the stated methodology to determine NCS limits, demonstrating assurance that the USL is not exceeded using information from CALC-2018-0009, "Single Parameter Limits for Fissile Material" (2018), and any other supporting analyses, as applicable. Update the FSAR with a justification for using the stated methodology.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have the capability to perform adequate safety analyses of all production processes that will be conducted in the facility.

SHINE Response

The SHINE Response to RAI 6b.3-12 will be provided by January 31, 2021.

RAI 6b.3-13

10 CFR Part 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility as a whole.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that for each methodology used to perform a nuclear criticality safety analysis, a validation report should be generated.

FSAR, Section 6b.3.1.3, states that SHINE commits to ANSI/ANS-8.24-2017, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations;" however, this is not sufficient to satisfy the ISG augmenting NUREG-1537 acceptance criteria related to validation. SHINE provided document CALC-2018-0012, "MCNP5 Validation for Reactivity in Solution Systems for the SHINE Facility," Revision 0, which contains information related to the ISG augmenting NUREG-1537 acceptance criteria; however, this is not equivalent to providing explicit commitments in the FSAR.

Provide information that addresses the ISG augmenting NUREG-1537 acceptance criteria related to validation. Specifically address the following criteria:

- a. a summary of the validation methodology, including the method used to select benchmark experiments, determine bias and bias uncertainty, and determine the upper subcritical limit;

- b. a summary of the physical systems and area(s) of applicability covered by the validation report;
- c. a description of the methods used to justify applying the methodology outside the area(s) of applicability;
- d. a summary of the plant-specific benchmark experiments used to validate the methodology;
- e. a justification of the proposed margin of subcriticality; and
- f. a description of the controlled software and hardware.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have the capability to perform adequate safety analyses of all production processes that will be conducted in the facility.

SHINE Response

- a. The SHINE validation report uses benchmarks from the Handbook of the International Criticality Safety Benchmark Evaluation Project (ICBEP) (Reference 7). Benchmarks were selected for evaluation based on their similarity to the SHINE solution system, as no plant-specific benchmark experiments are available. The fissile material, enrichment, chemical form, range of concentration, and reflector materials were considered in the selection of benchmarks. A total of 128 cases from 14 separate benchmark series were selected. The selected benchmarks series, number of cases selected from each benchmark series, and a description of each physical system is provided in Table 6b.3-13-1.

Table 6b.3-13-1: Summary of Benchmarks Selected for the SHINE Validation Report

Benchmark Series	Cases	Description of Physical Systems
LEU-SOL-THERM-003	9	10.06% enriched uranyl nitrate, un-reflected
IEU-SOL-THERM-002	13	30.45% enriched uranyl fluoride, water-reflected and un-reflected
IEU-SOL-THERM-003	46	30.3% uranyl fluoride, water-reflected and un-reflected
IEU-SOL-THERM-004	1	14.7% uranyl sulfate, reflected by beryllium oxide
LEU-SOL-THERM-004	7	9.97% enriched uranyl nitrate, water-reflected
LEU-SOL-THERM-007	5	9.97% enriched uranyl nitrate, un-reflected
LEU-SOL-THERM-008	4	9.97% enriched uranyl nitrate, concrete-reflected
LEU-SOL-THERM-016	7	9.97% enriched uranyl nitrate, water-reflected
LEU-SOL-THERM-017	6	9.97% enriched uranyl nitrate, un-reflected
LEU-SOL-THERM-018	6	9.97% enriched uranyl nitrate, concrete-reflected
LEU-SOL-THERM-020	4	9.97% enriched uranyl nitrate, water-reflected
LEU-SOL-THERM-021	4	9.97% enriched uranyl nitrate, un-reflected
LEU-SOL-THERM-023	9	9.97% enriched uranyl nitrate, un-reflected
LEU-SOL-THERM-025	7	9.97% enriched uranyl nitrate, concrete-reflected

The bias and bias uncertainty were calculated using the methodology described in NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (Reference 8). The benchmark data were tested using a modified Shapiro-Wilk test for normality and were determined to be normally distributed. A single-sided tolerance limit approach was used to determine the bias uncertainty. The upper subcritical limit is the

difference between unity and the sum of the bias (zero, because a positive bias was determined), the bias uncertainty, and the subcritical margin.

- b. A description of each physical system covered by the SHINE validation report is provided in Table 6b.3-13-1.

A summary of the area of applicability covered by the SHINE validation report is provided in Table 6b.3-13-2.

Table 6b.3-13-2: Area of Applicability Summary

Parameter	Area of Applicability
Fissile Material and Composition	Uranyl Sulfate Uranyl Nitrate Uranyl Fluoride
Chemical Form	Solution
Average Neutron Energy Causing Fission (ANECF) (MeV)	0.004-0.064
Enrichment (wt. %)	10-30.5
Reflector Materials	None Water Graphite Beryllium Oxide Concrete
Uranium Concentration (g-U/L)	52.8-960
H/ ²³⁵ U Ratio	75-1610

- c. As described in Subsection 6b.3.1.5 of the FSAR, the validation range may be extended beyond the range of the benchmark data using additional subcritical margin or bias trending analysis to ensure that the existing subcritical margin is appropriate. Where extrapolation or wide interpolations are used to extend the validation range, the recommendations of NUREG/CR-6698 (Reference 8) are used.
- d. No plant-specific benchmark experiments are available.
- e. The margin of subcriticality used for SHINE solution processes is 0.06. A subcritical margin of 0.05 was conservatively selected based on the quantity and quality of the selected benchmarks. An additional subcritical margin of 0.01 is applied to provide additional conservatism to account for the limited number of experimental benchmarks specific to uranyl sulfate systems.
- f. The SHINE validation process was performed using Monte Carlo n-Particle (MCNP) software, version MCNP5-1.60. ENDF/B-VII.1 standard cross-section libraries and ENDF/B-VII.0 thermal scattering cross-sections were used for the validation process. The software is installed on a group of Linux computers using CentOS release 7.3.1611. Each computer uses either AMD Opteron 6344 12-core or AMD Opteron 6100 16 core processors. Verification of the MCNP software installation was performed using developer-supplied verification tools, and re-verification of the computational system is conducted following any changes to the hardware or operating system.

Subsection 6b.3.1.4 and 6b.3.1.5 of the FSAR have been revised to further summarize the SHINE validation report, as described above. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 6b.3-14

10 CFR 50.68(a) states that the applicant shall comply with the requirements of 10 CFR 70.24, "Criticality accident requirements," or meet certain alternative requirements, as described in 10 CFR 50.68(b), in lieu of maintaining a criticality accident alarm system as described in 10 CFR 70.24.

10 CFR 70.24(a) requires, in part, that each licensee authorized to possess special nuclear material (SNM) in a quantity exceeding 700 grams of contained uranium-235 (U-235), 520 grams of U-233, 450 grams of plutonium, 1.5 kilograms of contained U-235 if no uranium enriched to more than 4 wt.% U-235 is present, or 450 grams of any combination thereof, maintain in each area in which such licensed SNM is handled, used, or stored, a criticality accident alarm system.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b3.2, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant should state clearly how the design of the facility or process provides for criticality control and should identify how the requirements of 10 CFR 70.24 were considered.

FSAR, Section 6b.3.3, "Criticality Accident Alarm System," states that the SHINE facility provides a criticality accident alarm system (CAAS) to detect a criticality event in the areas *in which non-exempt quantities of fissile material* greater than the limits identified in 10 CFR 70.24(a) are used, handled, or stored outside the irradiation units, where "exempt fissile material" is defined as SNM that meets the requirements from classification as fissile material as specified in 10 CFR 71.15. However, the requirements of 10 CFR 70.24 regarding whether a CAAS is required are based on specific, objective criteria of SNM mass quantities by isotope (or combinations thereof). It does not provide any distinctions as to whether such SNM quantities are, or should be considered, fissile or fissile-exempt, nor does it provide any exceptions for SNM quantities in excess of those limits. As such, SNM quantities greater than the limits established by 10 CFR 70.24 require CAAS coverage regardless of whether they meet the requirements from classification as fissile material as specified in 10 CFR 71.15.

Revise the FSAR to be consistent with the requirements of 10 CFR 70.24, or justify why those requirements do not need to be met for certain areas of the facility.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will develop, implement, and maintain a criticality accident alarm system that meets the acceptance criteria in Section 6b.3 of the ISG; and will have in place an NCS program.

SHINE Response

The SHINE Response to RAI 6b.3-14 will be provided by January 31, 2021.

RAI 6b.3-15

10 CFR 50.34(b)(6) states that the FSAR shall include information concerning facility operation, including the applicant's organizational structure, allocations or responsibilities, and personnel qualification requirements.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant's surveillance requirements should be considered acceptable if the applicant has met certain acceptance criteria or has identified and justified an alternative, including meeting the intent of ANSI/ANS-8.19, "Administrative Practices for Nuclear Criticality Safety," and ANSI/ANS-8.20, "Nuclear Criticality Safety Training," as they relate to training.

FSAR, Section 6b.3.1.2, discusses the minimum qualification entry requirements for criticality safety staff, including fissile material handlers. This suggests that fissile material handlers are considered part of the criticality safety staff. However, Section 6b.3.1.2 further states that there are three qualification levels for criticality safety staff and specific functional area qualifications for fissile material handlers. This suggests that fissile material handlers have separate qualification requirements from the criticality safety staff.

FSAR, Section 6b.3.1.2, states that SHINE's NCS training program consists of two tiers, with Tier 1 being directed toward personnel who manage, work in, or work near areas where a potential for criticality exists and Tier 2 being specific to NCS staff. However, it is not clear whether NCS staff are required to receive both tiers, or simply Tier 2, of training.

- a. Clarify whether fissile material handlers are considered part of the criticality safety staff. State the difference in qualification requirements for fissile material handlers and criticality safety staff.
- b. Clarify which training requirements apply to NCS staff.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

- a. Fissile material handlers are not part of the NCS staff. Fissile material handlers are operational staff that will receive training on nuclear criticality safety. In addition to receiving training on the tier one training requirements, fissile material handlers must demonstrate proficiency in the knowledge areas through the administration of examinations. The qualification and training for fissile material handlers at SHINE includes:
 - Tier one training requirements;
 - A working level of knowledge of criticality accident alarm system response and criticality safety and facility procedures for identifying, responding to, and reporting criticality safety non-conformances; and

- A working level of knowledge of facility processes as it applies to specific processes for which qualification as a fissile material handler is applicable.

The training and qualification requirements for NCS staff are described in Subsection 6b.3.1.2 of the FSAR.

Subsection 6b.3.1.2 of the FSAR has been revised to remove fissile material handlers from the listing of NCS staff. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

- b. The three qualification levels for NCS staff are NCS analyst, NCS engineer, and senior NCS engineer. NCS analysts are required to demonstrate knowledge and proficiency in areas appropriate to the ability to effectively conduct calculations related to nuclear criticality safety. NCS engineers are required to demonstrate knowledge of tier one and tier two training requirements. Senior NCS engineers are required to be qualified as NCS engineers and demonstrate higher level of proficiency for selected subjects. Training requirements for NCS staff are consistent with the training requirements specified in ANS/ANS-8.26-2007 (R2016), Criticality Safety Engineer Training and Qualification Program (Reference 9).

RAI 6b.3-16

10 CFR 50.34(b)(6)(i) states that the FSAR shall include information concerning facility operation, including the applicant's organizational structure, allocations or responsibilities, and personnel qualifications requirements.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant's NCS organization and administration should be acceptable if certain acceptance criteria are met, including a commitment to provide distinctive NCS postings in areas, operations, work stations, and storage locations relying on administrative controls for NCS; and a commitment to require personnel to perform activities in accordance with written and approved procedures. Unless a specific procedure deals with the given situation, personnel shall take no action until the NCS staff has evaluated the situation and provided recovery procedures.

FSAR, Section 6b.3.1, "Nuclear Criticality Safety Program," states that the criticality safety program (CSP) is executed by qualified NCS staff using written procedures, SHINE facility management has a responsibility to require that activities involving fissile material be conducted using written procedures, and personnel are required to take no action until the NCS staff has evaluated the situation and provided recovery instructions for situations in which existing procedures are inadequate or do not exist. However, these statements only provide commitments in terms of the responsibilities of facility management and NCS staff and are not equivalent to a commitment to require personnel to perform activities in accordance with written and approved procedures. Additionally, the FSAR does not address the use of NCS postings in areas, operations, workstations, and storage locations relying on administrative controls.

- a. State whether SHINE commits to conduct activities that affect NCS in accordance with written and approved procedures. For situations in which a specific procedure is inadequate or does not exist, state whether personnel are required to take no action until the NCS staff has evaluated the situation and provided recovery procedures.

- b. State whether SHINE commits to using NCS postings in areas, operations, workstations, and storage locations relying on administrative controls.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

- a. Activities involving fissile material are conducted using written and approved procedures. For situations in which approved procedures are inadequate or do not exist, personnel are required to take no action until the NCS staff has evaluated the situation and provided recovery instructions. Subsection 6b.3.1.7 of the FSAR has been revised to clarify to use of approved procedures for activities involving fissile material. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.
- b. Procedures are supplemented by appropriate material labeling and postings, specifying material identification and limits on parameters, in areas, operations, workstations, and storage locations subject to procedural controls. Subsection 6b.3.1.7 of the FSAR has been revised to clarify to use of material labeling and postings. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 6b.3-17

10 CFR 50.34(b)(6)(iv) states that the FSAR shall include information concerning facility operation, including the applicant's plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant's surveillance requirements should be considered acceptable if the applicant has met certain acceptance criteria or has identified and justified an alternative, including a commitment to conduct and document periodic NCS audits such that all NCS aspects of surveillance requirements will be audited at least every two years.

FSAR, Section 6b.3.1.7, suggests that an audit of the overall effectiveness of the CSP is performed at least every three years. Additionally, reviews of NCSEs and calculations are performed such that each evaluation and calculation is reviewed at least once every three years. These commitments appear to be inconsistent with the ISG augmenting NUREG-1537 acceptance criterion that requires such audit aspects be performed at least every two years.

Provide a justification for performing NCS audits such that all NCS aspects will be audited every three years, as opposed to at least every two years.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who

are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

Performance of an audit of the overall effectiveness of the nuclear criticality safety program at least every three years is consistent with the guidance in Section 4.7 of ANSI/ANS-8.19-2014, Administrative Practices for Nuclear Criticality Safety (Reference 10), which states, "Management shall participate in auditing the overall effectiveness of the nuclear criticality safety program at least once every 3 years." The NRC endorsed the use of ANSI/ANS-8.19-2014 without exception via Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores (Reference 11).

RAI 6b.3-18

10 CFR 50.34(b)(6)(iv) states that the FSAR shall include information concerning facility operation, including the applicant's plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant's surveillance requirements should be considered acceptable if the applicant has met certain acceptance criteria or has identified and justified an alternative, including a commitment to include NCS audit requirements in the Administrative Controls section of the facility technical specifications.

Audit requirements are included in the Administrative Controls section of the facility technical specifications (Appendix A to the FSAR, "Technical Specifications and Bases"). However, this is not equivalent to a commitment to include audit requirements in the Administrative Controls section of the facility technical specifications.

State whether SHINE commits to include NCS audit requirements in the Administrative Controls section of the facility technical specifications.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that credible postulated criticality accident scenarios can be performed and adequate preventive and mitigative controls and measures will be included in the production facility technical specifications as required by 10 CFR 50.36.

SHINE Response

SHINE has revised Section 5.2.4 of the technical specifications to include the nuclear criticality safety program audit requirements in the Administrative Controls section of the technical specifications. A mark-up of the technical specifications incorporating these changes is provided as Attachment 2. SHINE will provide a revision to the technical specifications incorporating the mark-up by February 28, 2021.

RAI 6b.3-19

10 CFR 50.34(b)(6)(iv) states that the FSAR shall include information concerning facility operation, including the applicant's plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

The ISG augmenting NUREG-1537, Part 2, Chapter 6b.3, "Nuclear Criticality Safety in the Radioisotope Production Facility," states, in part, that the applicant's surveillance requirements should be considered acceptable if the applicant has met certain acceptance criteria or has identified and justified an alternative, including a commitment to conduct and document walkthroughs of all operating SNM process areas such that all areas will be reviewed at some specified frequency. The reviewer should consider the complexity of the process, the degree of process monitoring, and the degree of reliance on administrative controls in assessing the acceptability of the specified frequency.

FSAR, Section 6b.3.1.7, states that operations are reviewed at least annually to verify that procedures are being followed and that process conditions have not been altered to affect the NCSE. NCS staff conduct and participate in routine audits of NCS practices, including compliance with procedures. However, it is not clear whether these oversight activities include a physical walkthrough of operating process areas, and a justification for the frequency in which these oversight activities are performed is not provided.

State whether SHINE commits to have NCS staff conduct and document walkthroughs of all operating SNM process areas such that all areas will be reviewed at some frequency. Provide a justification for the specified frequency.

This information is necessary for the NRC staff to make the necessary evaluation findings described in the ISG augmenting NUREG-1537, Part 2, Section 6b.3. Specifically, the requested information will support the NRC staff in concluding that SHINE will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.

SHINE Response

SHINE will conduct walkthroughs of facility processes and procedures as part of the annual operational review described in Subsection 6b.3.1.7 of the FSAR. The annual frequency of the operational review, including the review of process conditions, is consistent with the guidance in Section 8.6 of ANSI/ANS-8.19-2014 (Reference 10). Subsection 6b.3.1.7 of the FSAR has been revised to clarify that the annual operational review includes walkthroughs of facility processes and procedures. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 6b.3-20

10 CFR 50.59 states, in part, that licensees may make changes in the facility as described in the FSAR, make changes in the procedures as described in the FSAR, and conduct tests or experiments not described in the FSAR without obtaining a license amendment if certain criteria are met. 10 CFR 50.59(a)(1) defines "change" as a modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended function will be accomplished.

FSAR, Section 6b.3.1.5, states that process or design changes that could affect NCS limits or controls are evaluated using the facility change process requirements of 10 CFR 50.59. Such changes include new design, operation, or modification to existing SSCs; computer programs; processes; operating procedures; or administrative controls. This appears to be inconsistent with the definition of “change” in 10 CFR 50.59, and thus does not provide a satisfactory commitment to evaluate all appropriate changes against 10 CFR 50.59 criteria. Specifically, changes to methodologies, such as a change to computational code validation methodology that could impact code bias, bias uncertainty, or the approved margin of subcriticality, would not be subject to SHINE’s commitment to evaluate the change against 10 CFR 50.59 criteria, despite 10 CFR 50.59(c) requiring an evaluation of such a change.

Clarify how SHINE intends to meet the requirements of 10 CFR 50.59, including a discussion of which changes SHINE will evaluate and the method in which such changes will be evaluated.

SHINE Response

The referenced statement in Subsection 6b.3.1.5 of the FSAR was not intended to be an all-inclusive listing of changes to be evaluated under the SHINE configuration management program, described in Section 5.5.4 of the technical specifications. SHINE will evaluate changes in accordance with 10 CFR 50.59, including the 10 CFR 50.59(a)(1) definition of “change,” without modification or exception. Subsection 6b.3.1.5 of the FSAR has been revised to remove the referenced statement. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

Chapter 13 – Accident Analysis

RAI 13-1

When licensing non-power production or utilization facilities, there have been questions as to what standards and criteria should be used in evaluating design basis accidents to evaluate the design basis of systems, structures and components that mitigate radiological releases to the environment (exposure to any individual in the unrestricted area). Presently, no radiological accident dose criterion is set forth in regulation and subsequent guidance to assess the risk to public health and safety resulting from the operation of non-power production or utilization facilities. Instead, the standards of 10 CFR Part 20, “Standards for Protection Against Radiation,” have been applied for evaluating the effects of a postulated accident, for instance:

- Before January 1, 1994, the accident dose criteria used to license a research reactor were generally compared to the public dose limits of 10 CFR 20.1 through 20.602 and Appendices. Therefore, the accident criteria the staff generally found acceptable for accident analyses were less than the public dose limits of 0.5 rem whole body and 3 rem thyroid for members of the public.
- On January 1, 1994, the NRC amended 10 CFR Part 20 to reduce the dose limit to a member of the public to 0.1 rem total effective dose equivalent (TEDE) with an implementation date of January 1, 1994. In lieu of an accident dose criterion, under 10 CFR 20.1301(d), a licensee or license application may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem. The 0.5 rem refers to the TEDE, defined in 10 CFR 20.1003, as the sum of the effective dose equivalent and the committed effective dose equivalent.

However, as discussed in NUREG-1537, there are several instances the staff has accepted very conservative accident analyses that exceed the 10 CFR Part 20 public dose limits discussed above.

In the FRN, the NRC proposed to amend its regulations that govern the license renewal process for non-power reactors, testing facilities, and other production or utilization facilities, licensed under the authority of Section 103, Section 104a, or Section 104c of the AEA, as amended, that are not nuclear power reactors.¹ In this rule, the NRC collectively refers to these facilities as non-power production or utilization facilities (NPUFs). The NRC has determined that the public dose limit of 0.1 rem (0.001 Sv) TEDE is unduly restrictive to be applied as accident dose criteria for NPUFs, other than those NPUFs subject to 10 CFR part 100.² However, the NRC considers the accident dose criteria in 10 CFR part 100 applicable to accident consequences for power reactors, which have greater potential consequences resulting from an accident, to be too high for NPUFs other than testing facilities. For these reasons, the NRC proposed to amend its regulations in 10 CFR 50.34 to add an accident dose criterion of 1 rem TEDE for NPUFs not subject to 10 CFR part 100.

This is consistent with the guidance found in NUREG-1537, Part 2, which provides discussion on a postulated accident scenario whose potential consequences are shown to exceed and bound all credible accidents. For nonpower facilities, this accident is called the maximum

¹ See 82 FR 15643, March 30, 2017.

² The NRC Atomic Safety and Licensing Appeal Board stated that the standards in 10 CFR part 20 are unduly restrictive as accident dose criteria for research reactors (Trustees of Columbia University in the City of New York, ALAB-50, 4 AEC 849, 854-855 (May 18, 1972)).

hypothetical accident. Since the consequences of the postulated maximum hypothetical accident should exceed those of any credible accident at the facility, the accident is not likely to occur during the life of the facility. The maximum hypothetical accident is used to demonstrate that the maximum consequences of operating the facility at a specific site are within acceptable limits.

The accident dose criterion of 1 rem TEDE in the proposed NPUF rule is based on the Environmental Protection Agency's (EPA) Protection Action Guides (PAGs), which were published in the EPA document, 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." In January 2017, the EPA published an update to its PAGs in EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents." This update to the EPA PAGs did not change the basis for the 1 rem TEDE early phase PAG published in 1992. The purpose of the EPA PAGs is to support decisions on protective actions to provide reasonable assurance of adequate protection of the public from unnecessary exposure to radiation.

The EPA PAGs are dose guidelines to support decisions that trigger protective actions such as staying indoors or evacuating to protect the public during a radiological incident. The PAG is defined as the projected dose to an individual from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended. Three principles considered in the development of the EPA PAGs include: (1) prevent acute effects; (2) balance protection with other important factors and ensure that actions result in more benefit than harm; and, (3) reduce risk of chronic effects. In the early phase of the nuclear incident, which may last hours to days, the EPA PAG recommends the protective actions of sheltering-in-place or evacuation of the public to avoid inhalation of gases or particulates in an atmospheric plume and to minimize external radiation exposures between 1 rem to 5 rem. So, if the projected dose to an individual from an incident is less than 1 rem, no protective action for the public is recommended.

In its operating license application, SHINE selected accident dose criteria (in lieu of a criterion stated in the regulation) for members of the public as follows:

- Radiological consequences to an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not exceed 500 mrem TEDE for the duration of the accident; and,

Radiological consequences to workers do not exceed 5 rem TEDE during the accident. [SHINE justifies applying this criterion to a worker within the facility as opposed to the "control room" since immediate operator action inside the facility is not required to stabilize accident conditions. The SHINE irradiation units do not share systems and components. Therefore, the design basis accidents assume no interconnective failures. As generally assumed in the sequence of events, SHINE states facility personnel evacuate the immediate area [the facility confinement] within 10 minutes upon actuation of the radiation area monitors.]

The SHINE FSAR Chapter 13, "Accident Analysis," provides the design basis accident analyses which are evaluated against the dose criterion.³ The intent of these analyses is to evaluate 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 including determination of the margins of safety during normal operations and transient

³ ADAMS Accession No. ML19211C323

conditions anticipated during the life 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.

The SHINE FSAR Chapter 14, "Technical Specifications," provides limiting conditions for operation of the production facility.⁴ The safety margins contained within the design basis accidents are products of specific values and limits contained in the facilities technical specifications and other values, such as assumed accident or transient initial conditions or assumed safety system response times.

For a Part 50 license, the following is considered:

- Accident dose criteria, when compared against the maximum hypothetical accident, is a helpful aid in evaluating a proposed site with the objective of assessing the risk to public health and safety resulting from operation of the facility.

As discussed in the ISG Augmenting NUREG-1537, the maximum hypothetical accident is used to demonstrate that the maximum consequences of operating the nuclear facility at a specific site is within the acceptable accident dose limits. The maximum hypothetical accident is an accident with radiological consequences that bound all other credible accidents likely to occur over the life of the nuclear facility. Therefore, the assumed fission product release from the maximum hypothetical accident should be based upon a major accident, hypothesized for purposes of siting analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible.

- There is no 10 CFR Part 50 regulatory requirement for a worker accident dose criteria, other than the requirements in 10 CFR Part 20, "Standards for Protection Against Radiation."⁵ However, the SHINE Design Criterion 6 – Control Room, states:

A control room is provided from which actions can be taken to operate the irradiation units safely under normal conditions and to perform required operator actions under postulated accident conditions.

This criterion is similar to 10 CFR 50, Appendix A, General Design Criterion-19, Control Room, which is not applicable to NPUFs such as the SHINE facility. It is required for light water reactor nuclear power plant control room design where the operator's necessity to appropriately respond during an accident is properly viewed as having a potential impact on the public health and safety. The purpose is to provide a control room from which actions can be taken to operate the facility safely under normal conditions and to maintain the facility in a safe condition under accident conditions.⁶

At the SHINE facility, in the event of a design basis accident or transient, the other irradiation units will presumably be operating, and control room operators would need to take actions to

⁴ ADAMS Accession No. ML19211C339

⁵ Note: 10 CFR Part 70 does contain a regulatory requirement for accident dose to workers because of lessons learned from fatal and near miss accidents at fuel cycle facilities involving chemicals commingled with special nuclear material.

⁶ It is generally understood that an objective of the criteria is to ensure that the design of the control room and its habitability systems is such that a "shirt-sleeved" environment is provided for the control room operators. Such an environment is perceived to be supportive of facilitating operator response to normal and accident conditions and would minimize errors of omission or commission. Another objective is to ensure that the radiation dose levels in the control room would make it the "safest" location on site, thereby allowing the operators to remain in the control room and not evacuate. Any reduction in the ability of the operators to respond appropriately during an accident is properly viewed as having a potential impact on the public health and safety.

continue to operate the facility safely and to maintain the facility in a safe condition. It therefore seems appropriate to assess the radiological consequences of the control room operator, given General Design Criteria 6, as their required operations are necessary to continue operation of the other irradiation units and maintain the facility in a safe condition under accident conditions.

It is noted that the NRC staff views the accident dose design criterion as a “figure of merit” and does not represent actual doses received due to a design-basis event or transient. The shielding design of the facility ensures the applicable limits in 10 Part CFR 20 are met and thus protecting the worker which is discussed in Chapter 11, “Radiation Protection Program and Waste Management,” of the SHINE FSAR. Lastly, as low as reasonably achievable (ALARA) program practices such as work planning and source term minimization, coupled with existing radiation exposure procedural controls ensure worker doses are not adversely impact the licensee’s ability to maintain doses resulting from plant operation within the applicable limits.

Therefore, the NRC staff requests that SHINE discuss the following, with the support of any relevant reference calculations or documents, related to information provided in its operating license application:

- a. Confirm the NRC staff’s understanding of the SHINE-proposed accident dose criteria of 500 mrem TEDE to members of the public to serve the purpose as the site evaluation factor, as discussed in the Federal Register (FR), Volume 82, Number 60, dated March 30, 2017. Given the NRC-proposed draft rule, discuss a technical justification for the SHINE-proposed accident dose criterion, as necessary for the licensing of the SHINE Medical Isotope Production Facility; this should include a comparison to the basis for the NRC-proposed accident dose criterion of 1 rem TEDE in the draft NPUF rule (see: 82 FR 15643).

OR

Discuss whether SHINE would adopt, with justification, the accident dose criterion proposed in the NRC rule described in Federal Register Notice (FRN) 82 FR 15643, which provides reasonable assurance of adequate protection of the public in the unlikely event of radiological incident.

- b. In light of the discussion provided above, provide a technical justification for why the worker dose criterion is assumed to be analyzed for facility personnel and not the operator(s) for the SHINE facility. Please provide accident analysis results for control room operators to be consistent with the SHINE Design Criteria, Criterion 6 – Control Room.
- c. Clarify what 10 CFR Part 50 regulatory requirement SHINE is demonstrating to meet with the proposed worker accident dose criteria, and the basis for the dose value of 5 rem TEDE. Also clarify the purpose of the proposed worker accident dose criteria as there appears to be no necessary actions by the worker to maintain the facility in a safe condition under accident conditions. If there are necessary actions to control or mitigate the accident, provide these procedures and programmatic controls which can be implemented in the Technical Specifications (Administrative Controls or otherwise).

OR

If there are no necessary actions by the worker outside the control room to maintain the facility in a safe condition under accident conditions, then discuss whether it would be

appropriate to remove from the SHINE FSAR, with justification, the proposed worker dose accident dose criteria.

SHINE Response

- a. SHINE has chosen to adopt the accident dose criterion of 1 rem total effective dose equivalent (TEDE) in the proposed rule described in 82 FR 15643 (Reference 12). As stated in the proposed rule, the NRC is proposing to amend its regulations in 10 CFR 50.34 to add an accident dose criterion of 1 rem (0.01 Sv) TEDE for non-power production or utilization facilities (NPUFs) not subject to 10 CFR Part 100.

As stated in the proposed rule, the accident dose criterion of 1 rem (0.01 Sv) TEDE is based on the Environmental Protection Agency's (EPA) Protection Action Guides (PAGs), which were published in EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (Reference 13). The EPA PAGs are dose guidelines to support decisions that trigger protective actions such as staying indoors or evacuating to protect the public during a radiological incident. In the early phase (i.e., the beginning of the nuclear incident, which may last hours to days), the EPA PAG that recommends the protective action of sheltering-in-place or evacuation of the public to avoid inhalation of gases or particulates in an atmospheric plume and to minimize external radiation exposures, is 1 rem (0.01 Sv) to 5 rem (0.05 Sv). If the projected dose to an individual from an incident is less than 1 rem (0.01 Sv), then no protective action for the public is recommended. In light of this understanding of the early phase EPA PAG, the NRC's proposed accident dose criterion of 1 rem (0.01 Sv) TEDE for NPUFs, other than testing facilities would provide reasonable assurance of adequate protection of the public from unnecessary exposure to radiation.

SHINE has revised Sections 3.1, 13a2.2, 13a3, and 13b.2 of the FSAR to incorporate the accident dose criterion of 1 rem TEDE. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

- b. SHINE did not distinguish between the general facility staff and the control room operators in the worker dose calculations. To demonstrate that the SHINE Design Criterion 6 is met, a consequence analysis for the control room operator under accident conditions was performed.

Previous worker dose calculations only considered evacuation of workers from the radiologically controlled area (RCA) within a ten-minute evacuation period and did not consider continued occupation of the control room. SHINE has performed a radiological and chemical consequence analysis for the control room operator. The presence of operators in the control room is to only monitor facility conditions since the operators are not required to perform actions in the RCA.

SHINE has revised Sections 6a2.1, 6b.1, 13a2.2, 13a3, and 13b.2 of the FSAR to replace the RCA worker with the control room operator as the worker receptor for the radiological consequence analysis and to provide the results of the revised radiological consequence analysis to the control room operator. SHINE has revised Section 13b.3 of the FSAR to add the control room operator, in addition to the RCA worker, as the worker receptors for the chemical consequence analysis and provide the results of the revised chemical consequence analysis to the control room operator and RCA worker. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

- c. 10 CFR Part 50 does not contain an explicit requirement that an accident dose criterion be established for workers and demonstrated to be met.

There are no necessary actions by the worker outside the control room to maintain the facility in a safe condition under accident conditions. Upon actuation of a radiation area monitor alarm, facility personnel evacuate the RCA. As 10 CFR Part 50 does not contain an explicit requirement to evaluate accident dose to workers outside the control room, and there are no necessary actions by the worker outside the control room to maintain the facility in a safe condition under accident conditions, SHINE has removed the evaluation of potential radiological dose consequences to workers outside the control room from the FSAR.

As described in the SHINE Response to RAI 13-1.b, SHINE has replaced the evaluation of potential radiological dose consequences to workers outside the control room in the FSAR with an evaluation of potential radiological dose consequences to control room operators, demonstrating that the potential radiological dose received by control room operators does not exceed the 5 rem TEDE accident dose limit.

RAI 13-2

The regulations that are most relevant to radiation protection are contained in 10 CFR Part 20 and 10 CFR Part 50. Additional requirements, specific to particular uses or classes of facilities, are found in other portions of the regulations. Both 10 CFR Part 50 and 10 CFR Part 20 refer to various dose-based criteria and limits based on dosimetry methodologies defined by the International Commission on Radiological Protection (ICRP) in Publication 26, “Recommendations of the ICRP,” and Publication 30, “Limits for Intakes of Radionuclides by Workers.” The ICRP 30 dosimetry methodologies are applied in:

- 10 CFR Part 50 – through the TEDE criteria (defined in 10 CFR 50.2) for the design, construction, and operation of the facility under normal and accident conditions.
- 10 CFR Part 20 – through the TEDE limits (defined in 10 CFR 20.1003) to establish standards and practices for radiation protection purposes for occupational and public health during normal operation. 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,” to provides direction in how to determine external and internal exposures. The appendix provides an appropriate method to derive the Annual Limits on Intake and Derived Air Concentrations based on ICRP 30 tissue weighting factors.

The tissue weighing factors are directly codified by 10 CFR 20.1003, Definitions, within the table labeled, Organ Dose Weighting Factors, as follows:

Organ or Tissue	W_T
Gonads	0.25
Breast	0.15
Red Bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bong surfaces	0.03

Remainder	0.30
Whole Body	1.00

For both 10 CFR Parts 50 and 20, the TEDE is defined as the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). Acceptable practices for computing design-basis accident radiological consequences in terms of TEDE are to apply the exposure-to-committed effective dose equivalent factors for inhalation of radioactive material found in Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." The factors in the column headed "effective" yield doses corresponding to the committed effective dose equivalent. These tables are derived from the data provided in ICRP Publication 30 and have been found acceptable to the NRC staff as they meet the applicable regulatory requirements. Likewise, the exposure-to-effective dose equivalent factors for external exposure of radioactive material apply Federal Guidance Report No. 12, "External Exposure to Radionuclides."

Therefore, by default, compliance with the dose-related regulations of Parts 50 and 20 are demonstrated when applying the exposure-to-dose conversion factors of Federal Guidance Reports 11 and 12.

The SHINE FSAR Chapter 13 design basis accident analyses are evaluated against the applicant-selected accident dose criteria. The design basis accidents range from anticipated events, such as a loss of electrical power, to a postulated maximum hypothetical accident that exceeds the radiological consequences of any accident considered to be credible. To compute radiological consequences, the SHINE FSAR states that the dose conversion factors were taken from ICRP Publication 119, "Compendium of Dose Coefficients based on ICRP Publication 60," and Federal Guidance Report No. 12. It appears SHINE has applied a dosimetry methodology inconsistent with applicable dose-related regulations under 10 CFR Part 50. Therefore, by applying dose conversion factors based on ICRP Publication 60 dosimetry methodologies for a Part 50 license application, the applicant does not comply with the applicable regulations. To be compliant with the dose-related regulations of Parts 50, the exposure-to-committed effective dose equivalent factors for inhalation of radioactive material should apply those found in Table 2.1 of Federal Guidance Report No. 11 and 12.

Therefore, the NRC staff requests that SHINE discuss the following, with the support of any relevant reference calculations or documents, related to information provided in its operating license application:

Discuss how SHINE's selected dosimetry methodology satisfies applicable regulatory requirements and whether it will be necessary to re-compute the radiological consequences of all design-basis accidents in terms of TEDE to be in compliance with the NRC's regulations.

SHINE Response

To demonstrate compliance with the dose-related regulations in 10 CFR Part 20 and 10 CFR Part 50, SHINE has revised the radiological consequence analysis, applying the dose conversion factors of Federal Guidance Report (FGR) No. 11, Limiting Values of Radionuclides Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion (Reference 14) and FGR No. 12, External Exposure to Radionuclides in Air, Water, and Soil (Reference 15).

SHINE has revised Section 13a2.2 of the FSAR to describe the application of dose conversion factors from FGR No. 11 and FGR No. 12 in the revised radiological consequence analysis. SHINE has revised Sections 13a3 and 13b.2 of the FSAR to provide the results of the revised radiological consequence analysis. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 13-3

10 CFR Part 50.34 requires that each applicant for a construction permit or operating license provide an 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.

Regulatory Guide 1.145, Rev 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of design basis accidents radiological releases at the site boundary as they relate to the applicable siting requirements where short-term atmospheric dispersion factors (χ/Q values) are computed at the 95th-percentile value (i.e., χ/Q value that is equal to or exceeded no more than 5 percent of the total time). Both NUREG-1537 and the ISG Augmenting NUREG-1537 refer to RG 1.145 with respect to accident analyses.

Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," as it pertains to the development of technical specifications based on the SHINE FSAR for the purposes of crediting natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms. In addition, with respect to installed features credit taken for their effectiveness should depend on the adequacy of the related quality assurance procedures undertaken, including the extent to which surveillance tests simulate the conditions to be met in practice. If assumptions regarding atmospheric dilution are involved, they should not be less conservative than those used in the analysis of design basis accidents.

It is noted here for further discussion that RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," provides regulatory positions on long-term atmospheric dispersion estimated for routine releases of effluent. For these assessments, it is typical regulatory practice to accept 50th-percentile χ/Q value.

The SHINE FSAR Chapter 13 design basis accident analyses are evaluated against the applicant-selected accident dose criterion. As presented by SHINE, the design basis accidents range from anticipated events, such as a loss of electrical power, to a postulated maximum hypothetical accident that exceeds the radiological consequences of any accident considered to be credible. SHINE identified these design basis accidents using the following sources of information:

- NUREG-1537 and the ISG Augmenting NUREG-1537;
- Process hazard analysis method within the integrated safety analysis process; and,
- Experience of the hazard analysis team.

SHINE selected accident dose criteria for members of the public as follows:

- Radiological consequences to an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not exceed 500 mrem total effective dose equivalent (TEDE) for the duration of the accident; and,
- Radiological consequences to workers do not exceed 5 rem TEDE during the accident.

The intent of these analyses is to evaluate 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 including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

SHINE computed long-term 50th percentile (average) χ/Q values at the nearest point along the site boundary and at the nearest resident location. This is consistent with the staff guidance found in NUREG-1537, Chapter 11, Radiation Protection, for the purposes of demonstrating compliance with the limits of 10 CFR Part 20 to assess routine releases. These 50th percentile χ/Q values were also applied to the Chapter 13 design basis accident analyses which is nonconservative to demonstrate compliance with 10 CFR 50.34 when evaluating the radiological consequences of postulated design basis accidents (i.e., short-term events) for facility siting and operation.

It is acknowledged there can be a misinterpretation of certain statements found in NUREG-1537, Part 2, Chapter 13, since no explicit percentile χ/Q value is made for accident analysis purposes. However, Chapter 13, Accident Analysis, Subsection, Radiological Consequences, does refer to RG 1.145 as an acceptable method for demonstrating compliance with the applicable siting criteria. Regulatory Guide 1.145, Section 3, "Determinations of 5% [95th percentile] Overall Site χ/Q Values," states in part, "The χ/Q values that are exceeded no more than 5% of the total time around the exclusion area boundary... should be determined..."

In other words, the purpose of evaluating the radiological consequences at the 95th-percentile value reasonably assures radiological consequences at the site boundary are not exceeded more than 5 percent of the time. Therefore, by applying the long-term 50th-percentile χ/Q values imply the computed radiological consequences at the site boundary are met only 50 percent of the time. Staff experience with both long-term 50th- and short-term 95th percentile χ/Q values has shown non-linearity between the computed radiological consequence results which can range between three-orders-of-magnitude in difference depending on the site location.

Computing radiological consequences of design basis accidents at the 95th-percentile χ/Q value provides reasonable assurance that facilities' licensing bases will not be exceeded by more than 5.0 percent within any given year of operation.

SHINE calculation number 2012-03852 Rev 0, "Short-Term Diffusion Estimates for SHINE," provides the details of the analysis to calculate atmospheric dispersion factors to be used to assess the consequences of an accidental release of radioactive material. Both the overall bounding long-term and short-term 50th and 95th-percentile χ/Q values are reported to be $3.88E-4$ s/m³ and $5.66E-3$ s/m³ respectively. This difference in χ/Q values would impact the reported SHINE FSAR radiological consequences by about a factor of 15.

Therefore, the NRC staff requests that SHINE discuss the following, with the support of any relevant reference calculations or documents, related to information provided in its operating license application:

- a. Provide a postulated set of short-term atmospheric χ/Q values (95th-percentile) at the site boundary based on site-specific meteorological data to be presented in the SHINE FSAR.
- b. Recompute the radiological consequences of each design basis accident with the short-term atmospheric χ/Q values.

SHINE Response

- a. Table 13-1-1 provides the 95th percentile site boundary atmospheric dispersion (χ/Q) values for the SHINE site. SHINE has revised the radiological consequence analysis to incorporate the use of these 95th percentile χ/Q values. SHINE has revised Section 13a2.2 of the FSAR to describe the use of 95th percentile site boundary χ/Q values in the radiological consequence analysis. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

Table 13-1-1: 95th Percentile Site Boundary χ/Q Values

Time Bin (Hours)	0 – 2	2 – 8	8 – 24	24 – 96	96 – 720
Site Boundary χ/Q Value (s/m³)	5.66E-03	3.45E-03	2.70E-03	1.58E-03	7.31E-04

- b. SHINE has revised the radiological consequence analysis to incorporate the use of 95th percentile χ/Q values. SHINE has revised Tables 13a3-1 and 13b.2-2 of the FSAR to provide the radiological dose consequences of design basis accidents using the 95th percentile χ/Q values. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 13-4

10 CFR Part 50.34 requires that each applicant for a construction permit or operating license provide an 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. It is the staff’s understanding that the proposed radiological accident dose criterion serves the purpose of evaluating the suitability of the site from operation of the facility for the purposes of computing radiological consequences.

10 CFR, Section 50.36 requires an applicant for an operating license to include in the application proposed technical specifications as it relates to the evaluations and analysis of the offsite radiological consequences of postulated accidents with fission products.

10 CFR 50.36(c)(3), requires TSs to include items in the category of surveillance requirements, which are requirements relating to test, calibration, or inspection to assure that the necessary

quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met.

Regulatory Guide 1.145, Rev 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of design basis accidents radiological releases at the site boundary as they relate to the applicable siting requirements where short-term atmospheric dispersion factors (χ/Q values) are computed at the 95th-percentile value (i.e., χ/Q value that is equal to or exceeded no more than 5 percent of the total time). Both NUREG-1537 and the ISG Augmenting NUREG-1537 refer to RG 1.145 with respect to accident analyses.

Regulatory Guide 2.2, as it pertains to the development of technical specifications based on the SHINE FSAR for the purposes of crediting natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms. In addition, with respect to installed features credit taken for their effectiveness should depend on the adequacy of the related quality assurance procedures undertaken, including the extent to which surveillance tests simulate the conditions to be met in practice. If assumptions regarding atmospheric dilution are involved, they should not be less conservative than those used in the analysis of design basis accidents.

Design basis accidents are postulated accidents that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety. The design basis accidents are not intended to be actual event sequences, but rather, intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. They can be thought of as loosely defined 'classes' of accidents that bound a number of facility processes, activities, and/or accident sequences identified through a risk-assessment. The quantification of the accidental release of fission products into the atmosphere, or accident radiological "source term," is intended to be representative of a major accident involving significant damage which affects the design of plant systems and is one element used to determine site suitability. The safety margins contained within the design basis accidents are products of specific values and limits contained in the facilities technical specifications, as required by 10 CFR 50.36 and other values, such as assumed accident or transient initial conditions or assumed safety system response times.

Beyond design basis accident is a term used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. However, as the regulatory process strives to be as thorough as possible, "beyond design-basis accident" sequences are analyzed to fully understand the capability of a design. Beyond design basis accidents are considered more unlikely than design basis accidents, non-safety-related systems, structures, and components can be credited for accident mitigation. For example, the 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) events for light-water-cooled nuclear power plants," allows the use of non-safety-related equipment for accident mitigation. These analyses often include multiple failures beyond

those considered for design basis accident analyses, and thus more realistic assumptions are allowed in the analyses.

The staff reviews the radiological consequences of design basis accidents in six parts: (1) review of selected bounding design basis accidents; (2) review of accident source terms; (3) review of the major structures, systems, and components of the facility that are intended to mitigate the radiological consequences of a design basis accident; (4) review of the characteristics of fission product releases from the proposed site to the environment, (5) review of the meteorological characteristics of the proposed site; and, (6) review of the total calculated radiological consequence dose at the site boundary from the bounding design basis accidents.

The NRC staff generally does not accept design basis accident analyses that credit facility features that:

- are not safety-related;
- are not covered by technical specifications;
- do not meet single-failure criteria, or;
- rely on the availability of offsite power. Design basis delays in actuation of these features should be considered, especially for those features that rely on manual operator intervention.

Analysis inputs should be the most restrictive values of plant parameters selected from the range of design values possible during the specific event so that the postulated consequences of the event are maximized. It is generally inappropriate to use values characterized as “best estimates.” Other considerations should include:

- The range of values applicable during an accident may vary from accident to accident and will likely differ from the range that applies during normal operations.
- The use of different parameter values in different portions of the analyses or to perform a sensitivity analysis to determine the limiting value.
- Facility parameters associated with a technical specification limiting condition for operation. If the limiting condition for operation specifies a range, or a value with a tolerance band, the most restrictive value should be used.
- Consider situations where and how some parameters may change value during the accident. In these cases, the calculation should either assume the most restrictive value for the entire duration or the calculation should be performed in time steps, with the appropriate parameter values used for each time step.
- Parameters based on the results of less frequent surveillance testing, for example, efficiency testing of charcoal filters, the degradation that may occur between periodic tests should be considered in establishing the analysis value.
- Analysis parameters which affected by density changes that occur in the process stream. With regards to specified volumetric flow rates as limiting conditions of operations, the

density used should be consistent with the density that is assumed in the surveillance procedure that demonstrates compliance with the limiting conditions of operations.

Lastly, a point of discussion regarding the application of the Single Failure Criterion is made when developing design basis accidents. The Single Failure Criterion, as a design and analysis tool, has the direct objective of promoting reliability through the enforced provision of redundancy in those systems which must perform a safety-related function.⁷ As discussed in NUREG-1537, for the purposes of facility design and accident analysis, and the applicable SHINE Design Criteria, a single failure means an occurrence which results in the loss of capability of a component or protection system to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of capability of the system to perform its safety functions.

In principle, the Single Failure Criterion as applied in design basis accident analyses is straightforward. Simply stated, it is a requirement that a system which is designed to carry out a defined safety function must be capable of carrying out its mission in spite of the failure of any single active component within the system or in an associated system which supports its operation. Application of the Single Failure Criterion involves a systematic search for potential single failure points and their effects on the system. Such a search is required by the Standard Review Plan and the Standard Format for the Content of Safety Analysis Reports for specified safety systems and components. The objective is to search for design weaknesses which could be overcome by increased redundancy, use of alternate systems or use of alternate procedures. In general, only those systems or components which are judged to have a credible chance of failure are assumed to fail when the Single Failure Criterion is applied. Such failures would include, for example, the failure of a valve to open or close on demand, the failure of an emergency diesel generator to start or the failure of an instrument channel to function.

The SHINE FSAR Chapter 13 design basis accident analyses are evaluated against the applicant-selected accident dose criterion. As presented by SHINE, the design basis accidents range from anticipated events, such as a loss of electrical power, to a postulated maximum hypothetical accident that exceeds the radiological consequences of any accident considered to be credible. SHINE identified these using the following sources of information:

- NUREG-1537 and the ISG Augmenting NUREG-1537;
- Process hazard analysis method within the integrated safety analysis process; and,
- Experience of the hazard analysis team.

The NRC staff has reviewed a sampling for both the irradiation facility and radioisotope production facility design basis accidents with a focus on the two maximum hypothetical accidents. It appears these maximum hypothetical accidents fit the description of beyond design basis accidents where multiple failures are assumed which is beyond typical consideration for licensing purposes.

⁷ ADAMS Accession No. ML060260236

Therefore, the NRC staff requests that SHINE discuss the following, with the support of any relevant reference calculations or documents, related to information provided in its operating license application:

- a. Re-assess the maximum hypothetical accidents considering the discussions above. It may be necessary to redefine the design basis accident source terms and sequence of events to meet the applicable public accident dose criteria.
- b. As discussed above, the DBAs are not intended to represent actual event sequences, but surrogates to enable deterministic evaluations of the response of the facilities engineered safety features. Based on SHINE's use of a risk-assessment to define creditable accident sequences and the substantial operating experience of similar facilities, provide a discussion of the following:
 - How SHINE classified and binned the accident sequences from the SHINE safety analysis into each DBA;
 - Which technical specifications and limiting conditions of operations were developed from insights gained from the accident sequences identified from the SHINE safety analysis; and
 - How the accident sequences, which require workers to take preventive or mitigative actions in order to put the facility in a safe configuration, are reflected in the impacted DBA, including how these actions are controlled through procedures and programmatic controls that may be implemented in the Technical Specifications (Administrative Controls or otherwise).

SHINE Response

- a. The maximum hypothetical accidents (MHAs) currently described in the FSAR fit the definition of a beyond design basis accident, as defined in this request for additional information. In lieu of identifying a beyond design basis accident scenario as the MHA for the SHINE facility, SHINE has chosen to identify a credible fission product-based design basis accident (DBA) which bounds the radiological consequences to the public of all credible fission product-based accident scenarios as the MHA for the SHINE facility.

SHINE has revised Sections 13a2.1, 13a2.2, 13a3, 13b.1, and 13b.2 of the FSAR to re-define the MHA, as described above. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

- b. The types of accident sequences in the SHINE safety analysis (SSA) are derived from the guidance in Part 1 of the Interim Staff Guidance (ISG) augmenting NUREG-1537 (Reference 16). Additional accident sequences that are specific to the facility are derived from the process hazard analysis (PHA). This includes accident sequences specific to the irradiation facility (IF), the radioisotope production facility (RPF), and external events that are applicable to the entire facility. The SHINE Response to RAI 13-5 provides additional details of the categorization of accident sequences at the SHINE facility and a discussion of how the accident sequences are binned into the DBAs described in Chapter 13 of the FSAR.

The controls identified in the SSA include active and passive engineered controls and specific administrative controls (SACs). The engineered controls credited for prevention or

mitigation are incorporated into Section 3.0 (Limiting Conditions for Operation and Surveillance Requirements) or Section 5.5.4 (Configuration Management) of the technical specifications.

The accident sequences developed in the SSA do not credit operator actions for mitigation. Human actions credited for accident sequence prevention are identified as SACs, which are procedural controls incorporated into maintenance, inspection and testing, and operating procedures. These procedural controls are subject to the requirements of Section 5.4 (Procedures) of the technical specifications.

RAI 13-5

10 CFR 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

10 CFR Part 50, paragraph 50.57(a)(3) states that an operating license may be issued upon finding that, "There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public..."

The ISG to NUREG-1537 states in Section 13a2 that the information in this chapter should achieve the objectives stated in this chapter of NUREG-1537, Part 1 by demonstrating that the applicant has considered all potential accidents at the reactor facility and adequately evaluated their consequences.

The ISG to NUREG-1537 states NRC staff have determined that the use of Integrated Safety Analysis (ISA) methodologies as described in 10 CFR 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 items relied on for safety, 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.

SHINE has prepared a document entitled "SHINE Safety Analyses (SSA) Report (TECRPT-2020-016) which discusses the safety analyses methodology; however, this methodology is not discussed in the FSAR.

Revise the FSAR to include a description of the accident analysis methodology and criteria. Discuss the types of hazards considered (e.g., radiological, chemical), the phases of operation analyzed in the accident analysis (startup, normal operation, shutdown, non-routine operations), the receptors considered, and the criteria used to determine the acceptability of accident consequences for each type of hazard (e.g. chemical, radiological) and each receptor (e.g., public, worker, control room operator). Also discuss consideration of non-routine activities such as (1) unplanned maintenance activities; (2) periods of extended shutdown, or (3) conditions outside of the established Limiting Conditions of Operations (LCOs). Maintenance activities can

create situations where there could be reduced controls or barriers resulting in the release of hazardous material and extended shutdown periods or conditions exceeding LCOs could introduce new accident scenarios.

SHINE Response

The SHINE Response to RAI 13-5 will be provided by January 31, 2021.

RAI 13-6

10 CFR 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

10 CFR Part 50, paragraph 50.57(a)(3) states that an operating license may be issued upon finding that, "There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public..."

The ISG to NUREG-1537 states in Section 13a2 that the information in this chapter should achieve the objectives stated in this chapter of NUREG 1537, Part 1 by demonstrating that the applicant has considered all potential accidents at the reactor facility and adequately evaluated their consequences.

The SSA, as discussed in RAI 13-5, provides the results of the accident analyses which are summarized in Chapter 13 of the FSAR. The SSA also outlines the various programs related to the development, implementation and maintenance of the accident analysis.

Technical Specification Section 5.5.4 of FSAR Chapter (14) states that configuration management is applied to all safety related SSCs. This statement seems inconsistent with Item 7 in Section 5.3, "Programmatic Administrative Controls," of the SSA, which states that the configuration management program provides the means to evaluate "each change". The configuration management program should be applied to each proposed change because a change that involves a non-safety-related SSC could introduce an unanalyzed condition or a new hazard with significant consequences.

Revise the FSAR to clarify that configuration management is applied to "each change" not only to "safety-related SSCs" or explain why such a revision is not necessary.

SHINE Response

SHINE has revised Section 5.5.4 of the technical specifications to specify that the configuration management program is used to evaluate each change to the SHINE facility for the potential to affect safety-related structures, systems, and components (SSCs). A mark-up of the technical specifications is provided as Attachment 2. SHINE will provide a revision to the technical specifications incorporating the mark-up by February 28, 2021.

RAI 13-7

10 CFR 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

10 CFR Part 50, paragraph 50.57(a)(3) states that an operating license may be issued upon finding that, "There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public..."

The ISG to NUREG-1537 states in Section 13a2 that the information in this chapter should achieve the objectives stated in this chapter of NUREG 1537, Part 1 by demonstrating that the applicant has considered all potential accidents at the reactor facility and adequately evaluated their consequences.

The ISG to NUREG-1537 states NRC staff have determined that the use of Integrated Safety Analysis methodologies as described in 10 CFR 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 items relied on for safety, 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.

SHINE has stated in the SSA that they are using guidance from NUREG-1520 to support their accident analyses in the FSAR. The following items identified from SHINE's SSA summary are not consistent with the regulatory guidance in Chapter 3 of NUREG-1520. The staff needs this information to assess the completeness of the applicant's accident analyses and the adequacy of the applicant's accident analyses methodology. Furthermore, the staff needs this information to verify the applicant's implementation of the SSA methodology for reasonable assurance that the applicant will conduct operations without endangering the health and safety of the public.

- a. Section 2.5.2 of the SSA states that the dose calculations were made using both the site boundary and the location of the nearest resident as dose receptors. Revise the SSA dose calculations and FSAR, as necessary, to consider the distance to the end of the owner-controlled area, or the maximum exposed individual. Alternatively, justify use of the nearest resident.
- b. In the SSA, SHINE assigns a failure frequency index of -5 to some safe-by-design controls without further justification. Similarly, SHINE assigns a failure probability index of -4 or -5 to passive engineered controls with high design margin without further justification. Using these assumptions, failure of a safe-by-design component is inherently considered highly unlikely and therefore the accident sequence need not be developed and further analyzed. According to guidance in Chapter 3 of NUREG-1520, the default failure frequency or failure probability index for such controls is -3. The approach taken in the SSA is not consistent with the guidance in NUREG-1520. Re-evaluate the applicable accident sequences using

the assumptions from NUREG-1520. Alternatively, provide the analysis that justifies assigning the associated failure frequencies or failure probability indices.

- c. According to Tables 2.4.1 and 2.4.2 of the SSA, SHINE may assign a failure frequency index of -4 and a failure probability index of -3 to an enhanced specific administrative control. Given that this facility is first of a kind and the reliability of human actions in its operation has not been studied to the extent of those in a nuclear reactor or typical fuel cycle facility, it is unlikely these indices could be justified without a detailed analysis. According to the guidance in NUREG-1520, the default failure frequency or failure probability index for such controls is -2. Re-evaluate the applicable accident sequences using the assumptions from NUREG-1520 for administrative controls. Alternatively, provide the analysis that justifies assigning the associated failure frequencies or failure probability indices.
- d. As cited in NUREG-1520, the methodology should contain information on management measures applied to ensure designated safety controls are reliable and available to perform their intended safety function, i.e., management measures are necessarily distinct from the IROFS to which they are applied. The applicant's SSA describes "Reliability Management Measures" as programmatic administrative controls that are applied to credited controls. These Reliability Management Measures include maintenance, inspections, and testing. Appendix A appears to credit those measures as safety related. For those accident sequences in Appendix A that credit Reliability Management Measures as preventing or mitigating an accident sequence, the staff needs clarification on the credited controls to which the Reliability Management Measures are applied. If the credited controls are also Reliability Management Measures, the applicant should reevaluate the applicable accident sequences to identify and evaluate the failure likelihood of the controls to which the Reliability Management Measures are applied.

SHINE Response

The SHINE Response to RAI 13-7 will be provided by January 31, 2021.

RAI 13-8

10 CFR 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

10 CFR Part 50, paragraph 50.57(a)(3) states that an operating license may be issued upon finding that, "There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public..."

The ISG to NUREG-1537 states in Section 13a2 that the information in this chapter should achieve the objectives stated in this chapter of NUREG 1537, Part 1 by demonstrating that the applicant has considered all potential accidents at the reactor facility and adequately evaluated their consequences.

The ISG to NUREG-1537 states NRC staff have determined that the use of Integrated Safety Analysis methodologies as described in 10 CFR 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 items relied on for safety, 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.

The SSA included consequence categories comparable to the performance requirements in 10 CFR 70.61(b)(1) – (4), 70.61(c)(1) – (2) and 70.61(c)(4). However, the SSA does not discuss a comparable consequence category as provided in performance requirement 70.61(c)(3), i.e., a 24-hour release of radioactive material outside the restricted area in concentrations of 5000 times the values in Table 2 of Appendix B to Part 20. Furthermore, the SSA does not include credible accident sequences exceeding a comparable threshold. This threshold, as put forth in 70.61(c)(3), protects the public from releases that may result in intermediate consequences as described in Section 2.3.2 of the SSA.

Describe how the SSA considers a consequence category comparable to performance requirement 70.61(c)(3). Alternatively, justify its exclusion as a consequence category in the SSA.

SHINE Response

The SHINE Response to RAI 13-8 will be provided by January 31, 2021.

RAI 13-9

10 CFR 50.34(b) states that the FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analyses of the structures, systems and components and of the facility.

10 CFR 50.34(b)(2) states that a description and analyses of the structures, systems and components of the facility, with emphasis upon the performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.

The ISG to NUREG-1537 states in Section 6b.3 that the applicant has designed a facility that will provide adequate protection against criticality hazards related to the storage, handling, and processing of licensed materials. The facility design must adequately protect the health and safety of workers and the public during normal operations and credible accident conditions from the accidental criticality risks in the facility. It should also protect against facility conditions that could affect the safety of licensed materials and thus present an increased risk of criticality or radiation release.

The ISG to NUREG-1537 states NRC staff have determined that the use of Integrated Safety Analysis methodologies as described in 10 CFR 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 items relied on for safety, 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.

- a. FSAR Section 6b.3 states that the CSP is intended to meet the applicable criticality safety requirements of 10 CFR 70. Explicitly state which 10 CFR 70 requirements the applicant considers applicable and intends to meet. Explicitly state whether the CSP meets, not intends to meet, these requirements.
- b. The accident analyses methodology contained in the SSA (see RAI 13-5) states the risk of criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes within the RPF are subcritical, including use of an approved margin of subcriticality for safety. Additionally, FSAR Section 6b.3, "Nuclear Criticality Safety in the RPF," suggests that the CSP is only applicable to activities performed in the RPF. However, SSA Summary Table 2.8-1, "FSAR Accident Analyses for the Irradiation Facility," includes accident sequences in the Irradiation Facility (IF) that could result in inadvertent criticality. Describe how subcriticality is assured under normal and credible abnormal conditions for all nuclear processes performed within the IF, excluding the target solution vessels (TSVs). Specifically, describe how subcriticality is assured in the event of failure of a target solution vessel, TSV dump tank, and/or connected systems that can result in target solution migration into unintended or unanticipated locations.

SHINE Response

The SHINE Response to RAI 13-9 will be provided by January 31, 2021.

RAI 13-10

10 CFR 50.34(b)(6) requires the FSAR to include:

- i. the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements,
- ii. managerial and administrative controls to be used to assure safe operation,
- iii. plans for preoperational testing and initial operations,
- iv. plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems and components,
- v. plans for coping with emergencies, which shall include items specified in appendix E,
- vi. proposed technical specifications prepared in accordance with the requirements of 50.36.

This type of information forms the basis for safety programs that identify and manage the spectrum of hazards at the applicant's facility including chemical hazards. Chemical safety is specifically discussed in the ISG augmenting NUREG-1537, Part 1, as follows:

- Section 4b.4.2, "Processing of Unirradiated Special Nuclear Material," states that the application should provide chemical accident prevention measures as appropriate"
- Section 12.1.6, "Production Facility Safety Program," states that the radioisotope production facility must have an established safety program that includes chemical hazards
- Section 13b.3, "Analyses of Accidents with Hazardous Chemicals," states that the analyses of accidents for the production facility should include chemical hazards

- Section 14b, “Radioisotope Production Facility Technical Specifications,” states that the technical specifications should consider chemical hazards

Technical Specification, Section 5.5.1, “Nuclear Safety Program,” states, in part, the following:

The SHINE nuclear safety program documents and describes the methods used to minimize the probability and consequences of accidents resulting in radiological or chemical release.

Technical Specification, Section 5.5.8, “Chemical Control,” states the following:

The SHINE chemical control program ensures that on-site chemicals are stored and used appropriately to prevent undue risk to workers and the facility. The chemical control program implements the following activities, as required by the accident analysis:

1. Control of chemical quantities permitted in designated areas and processes;
2. Chemical labeling, storage and handling; and
3. Laboratory safe practices.

However, there is no description in the FSAR how the nuclear safety program or chemical control program identifies and manages chemical hazards.

Provide a description of the activities associated with the nuclear safety program and chemical control program that minimizes the probability and consequences of accidents resulting in a hazardous chemical release. Additionally, provide an explanation regarding the relationship between the nuclear safety program and the chemical control program as it relates to the identification and management of chemical hazards under NRC’s regulatory jurisdiction.

SHINE Response

The SHINE Response to RAI 13-10 will be provided by January 31, 2021.

References

1. NRC letter to SHINE Medical Technologies, LLC, "Issuance of Request for Additional Information Related to the SHINE Medical Technologies, LLC Operating License Application (EPID No. L-2019-NEW-0004)," dated November 10, 2020
2. SHINE Medical Technologies, LLC letter to the NRC, "SHINE Medical Technologies, LLC Application for an Operating License," dated July 17, 2019 (ML19211C143)
3. American National Standards Institute/American Nuclear Society, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement," ANSI/ANS-8.10-2015, La Grange Park, IL
4. American National Standards Institute/American Nuclear Society, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-2014, La Grange Park, IL
5. American National Standards Institute/American Nuclear Society, "Nuclear Criticality Safety in the Storage of Fissile Materials," ANSI/ANS-8.7-1998 (R2017), La Grange Park, IL
6. American National Standards Institute/American Nuclear Society, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors," ANSI/ANS-8.21-1995 (R2011), La Grange Park, IL
7. Nuclear Energy Agency, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)03, September 2014
8. Nuclear Regulatory Commission, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, January 2001
9. American National Standards Institute/American Nuclear Society, "Criticality Safety Engineer Training and Qualification Program," ANSI/ANS-8.26-2007 (R2016), La Grange Park, IL
10. American National Standards Institute/American Nuclear Society, "Administrative Practices for Nuclear Criticality Safety," ANSI/ANS-8.19-2014, La Grange Park, IL
11. U.S. Nuclear Regulatory Commission, "Nuclear Criticality Safety Standards for Nuclear Materials Outside Reactor Cores," Regulatory Guide 3.71, Revision 3 (ML18169A258)
12. U.S. Nuclear Regulatory Commission, "Non-Power Production and Utilization Facility License Renewal," *Federal Register*, Vol. 82, No. 60, March 30, 2017, pp.15643-15660
13. U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 400-R-92-001, May 1992
14. U.S. Environmental Protection Agency, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report No. 11, September 1988
15. U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, September 1993

16. U.S. Nuclear Regulatory Commission, "FINAL Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," October 17, 2012 (ML12156A069)

**ENCLOSURE 3
ATTACHMENT 1**

SHINE MEDICAL TECHNOLOGIES, LLC

**SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION
SUPPLEMENT NO. 5 AND RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

**FINAL SAFETY ANALYSIS REPORT CHANGES
PUBLIC VERSION
(MARK-UP)**

CHAPTER 3 – DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 DESIGN CRITERIA

Structures, systems, and components (SSCs) present in the SHINE facility are identified in [Tables 3.1-1](#) and [3.1-2](#), including the applicable FSAR section(s) which describe each SSC and the applicable SHINE design criteria. Design criteria derived from external codes, guides, and standards specific to the design, construction, or inspection of SSCs are included in the applicable FSAR section describing those SSCs. For each SSC, the FSAR section identifies location, function, modes of operation, and type of actuation for specific SSCs, as applicable.

Nuclear Safety Classification

Safety-related SSCs at SHINE are those physical SSCs whose intended functions are to prevent accidents that could cause undue risk to health and safety of workers and the public; and to control or mitigate the consequences of such accidents.

Acceptable risk is achieved by ensuring that events are highly unlikely or by reducing consequences less than the SHINE safety criteria. The SHINE safety criteria are:

- An acute worker dose of five rem or greater total effective dose equivalent (TEDE).
- An acute dose of ~~500 milli~~1 rem or greater TEDE to any individual located outside the owner controlled area.
- An intake of 30 milligrams or greater of uranium in a soluble form by any individual located outside the owner controlled area.
- An acute chemical exposure to an individual from licensed material or hazardous chemicals produced from licensed material that could lead to irreversible or other serious, long-lasting health effects to a worker or could cause mild transient health effects to any individual located outside the owner controlled area.
- Criticality in the radioisotope production facility (RPF).
- Loss of capability to reach safe shutdown conditions.

Some SSCs are nonsafety-related but perform functions that impact safety-related SSCs. These nonsafety-related SSCs have design basis requirements necessary to prevent unfavorable interactions with safety-related SSCs due to failure of the nonsafety-related SSCs.

Safety-related SSCs are identified in [Table 3.1-1](#) and nonsafety-related SSCs are identified in [Table 3.1-2](#).

SHINE Design Criteria

The SHINE facility uses design criteria to ensure that the SSCs within the facility demonstrate adequate protection against the hazards present. The design criteria are selected to cover:

- The complete range of irradiation facility and radioisotope production facility operating conditions.
- The response of SSCs to anticipated transients and potential accidents.
- Design features for safety-related SSCs including redundancy, environmental qualification, and seismic qualification.
- Inspection, testing, and maintenance of safety-related SSCs.

the primary system boundary (PSB) should the safety parameter trip points be exceeded, including high source range neutron flux.

In addition to TSV fill volumes and reactivity, the temperature of the target solution is monitored via the temperature of the PCLS water. Due to the low decay power of the target solution, its temperature is approximately equal to the cooling water temperature during startup mode. Due to the operating characteristics of the SHINE system, a decrease in the temperature of the target solution results in an increase in system reactivity. Excessive cooldown of the target solution during startup is prevented by the TRPS initiating an IU Cell Safety Actuation on low PCLS temperature and high source range neutron flux. The IU Cell Safety Actuation results in drainage of the target solution in the TSV to the TSV dump tank, which maintains the k_{eff} below 0.954 for the most reactive uranium concentration.

If at any time during the fill process neutron flux, TSV fill volume, or target solution temperatures are determined to be outside allowable parameters, operators will transfer the entire contents of the TSV to the TSV dump tank via gravity by opening the TSV dump valves. Due to the location of the TSV dump tank in the light water pool, decay heat removal requirements from the target solution are satisfied.

Mode 2: Irradiation Mode

After filling the TSV with target solution, the TSV is isolated from the TSV fill lift tank and the target solution hold tank by closing two redundant (in series) fill valves. During Mode 2, there is no capability to increase reactivity by adding target solution to the TSV. Given the aqueous target solution negative void and temperature coefficients, reactivity decreases as the irradiation process begins. Furthermore, any increase in operating power levels beyond normal operating conditions results in a temperature increase and a corresponding increase in the void fraction of the target solution itself, reducing the power level.

Testing has demonstrated that the pH of the uranyl sulfate remains stable during full power operation. The TSV, TSV dump tank, and TOGS are operated as a closed system, except for gas adjustments in TOGS for pressure and oxygen concentration control, to prevent an inadvertent addition of material that could affect reactivity or system chemistry. Malfunctions in the TOGS gas adjustments are evaluated for potential reactivity effects in the accident analysis discussed in [Subsection 13a2.1.2](#). The introduction of water into the system as a result of the failure of the pressure boundary is also analyzed in [Subsection 13a2.1.2](#).

During irradiation of the subcritical assembly, the TOGS is used to purge radiolytic hydrogen from the headspace in the TSV. [Section 4a2.8](#) provides a detailed discussion of the TOGS. The PCLS has the capability to remove approximately 137.5 kilowatts (kW) (469,000 British thermal units per hour [Btu/hr]) of heat from the TSV during irradiation. Cooling water is supplied to the external surfaces of the TSV and neutron multiplier at approximately 68°F (20°C) and exits the TSV and neutron multiplier at a maximum temperature of approximately 77°F (25°C). The TRPS monitors the PCLS during irradiation for low flow, high temperature, and low temperature and initiates an IU Cell Safety Actuation if the limits are exceeded.

The light water pool is not directly cooled. The light water pool provides a large thermal mass that absorbs heat and passively rejects heat to the PCLS-cooled components submerged in the pool and the surrounding concrete and air in the IU cell. The operating temperature of the pool ranges between 50°F and 95°F (10°C and 35°C).

out shape (Clayton, 1985)). This ensures that early predictions of critical volume are lower than the actual critical volume and result in lower actual k_{eff} values.

Uncertainty in critical volume for SHINE's normal approach to critical is a combination of instrumentation and system design parameters. To calculate the uncertainty in the critical volume during a normal startup following the 1/M process, the following assumptions are made:

- Detector uncertainty of 2 percent
- Level measurement uncertainty of 0.3 percent
- Minimum volume between 1/M points of 5 L

The uncertainty in the prediction of critical volume of the subcritical assembly using the 1/M method is estimated to be less than $[\quad]^{PROP/ECI}$ for the nominal core and $[\quad]^{PROP/ECI}$ for the limiting core.

The reactivity worth of that solution addition is calculated by MCNP5 and validated during startup physics testing. The calculational uncertainty in the worth of that solution addition is described above in this section.

TRPS high source range flux trip setpoints for Mode 1 are determined in accordance with requirements in [Subsection 4a2.6.2.7](#). The TRPS high source range flux trip setpoints provides an engineered control to initiate dumping of the target solution to the TSV dump tank prior to exceeding any allowable limits in anticipated transients and postulated accidents. Normal operating procedures and administrative controls provide the first barrier to maintaining a subcritical system, as described in [Subsection 4a2.6.2.7](#).

The facility startup physics testing confirms that reactivity coefficients, worths, and k_{eff} predictions are within the expected uncertainties. Estimates of reactivity parameter uncertainties from the startup plan must be bounded by those used for the determination of safety limits or trip setpoints, or if not, they are specifically evaluated for their effect on safety. Following completion of the startup plan, target solution uranium concentration is adjusted based on the startup testing results (see [Section 12.11](#)).

Due to the statistical nature of MCNP5, results from MCNP5 also contain a statistical error. This statistical error is included in calculations of bounding and analytical limits, and is generally small in comparison with other sources of error.

4a2.6.2.6.1 Uncertainties in k_{eff} Values Relying on MCNP Calculation

The TSV dump tank and TOGS are designed to ensure that any target solution contained within them is sufficiently shutdown. The TSV dump tank and TOGS k_{eff} values are directly calculated by MCNP5.

These direct calculations of k_{eff} account for the bias and bias uncertainty of MCNP5, a subcritical margin of 0.056 Δk , and uncertainty in MCNP5's statistical analysis, as described below.

The MCNP bias and bias uncertainty was evaluated using greater than 120 critical and near-critical benchmark data points. The benchmark data points were selected based on applicability to the SHINE uranyl sulfate solution system. Using the results of the benchmark calculations, an average k_{eff} is calculated, weighted by the combined calculated and experimental uncertainties.

The difference between the weighted average k_{eff} and 1 is the bias. The bias is calculated using the methodology in Section 2.4.1 of NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001b).

For conservatism, positive bias (i.e., where MCNP is found on average to over-predict k_{eff}) is assumed to be zero for the purposes of determining TSV dump tank and TOGS reactivity.

Bias uncertainty is calculated based on the pooled variance of the data used to calculate the bias and a one-sided tolerance factor. The bias uncertainty is calculated using the methodology described in Section 2.4.1 of NUREG/CR-6698 (NRC, 2001b).

MCNP statistical uncertainty is accounted for in the calculation by adding two times the standard deviation in k_{eff} reported by MCNP ($\sigma_{k_{\text{eff}},MCNP}$) to the k_{eff} reported by MCNP ($k_{\text{eff},MCNP}$).

The TSV dump tank and TOGS are designed to a k_{eff} value of less than 0.954 at the most reactive uranium concentration and at cold conditions. Reactivity analysis for the TSV dump tank and TOGS satisfies the following inequality:

$$k_{\text{eff},MCNP} + 2\sigma_{k_{\text{eff}},MCNP} \leq K_L - 0.06 - \Delta A0A$$

Where:

- K_L is the weighted single-sided lower tolerance limit.
- $\Delta A0A$ is an additional margin of subcriticality that may be necessary as a result of extensions to the area of applicability.

Both of these values are determined following the methodology of Section 2.4.4 of NUREG/CR-6698 (NRC, 2001b). K_L includes the effects of bias and bias uncertainty.

The methodology ensures with a high degree of confidence that the target solution is safely shut down by appropriately accounting for uncertainty in MCNP and providing margin to criticality.

See [Subsection 4a2.6.3.4](#) for detailed discussion on TSV dump tank subcriticality.

See [Section 4a2.8](#) for detailed discussion on TOGS subcriticality.

4a2.6.2.7 Trip Requirements to Limit Reactivity in Mode 1

In conjunction with the additional engineered and administrative controls described below, the limiting trip setpoint for TRPS high source range neutron flux signal is designed such that during normal operation and anticipated transients, the subcritical assembly k_{eff} remains below 1.0.

Anticipated transients in the subcritical assembly are described in [Subsection 4a2.6.3.3](#). Postulated accidents that could add reactivity to the system are described in [Subsection 13a2.1.2](#).

The trip setpoint is set to ensure a trip occurs prior to exceeding a percentage above the normal startup flux as measured by the neutron detection system, per the equation below:

collapse show that the transient occurs on a very short time scale, therefore a bounding analysis was made for the maximum power.

For the limiting event, a full void collapse was considered with the system operating at 125 kW. The maximum power reached was []^{PROP/ECI}, which results in a target solution power density below the limit for a transient event.

Loss of Neutron Source During Irradiation

Loss of neutron source during Mode 2 irradiation can occur due to intermittent decreases in the fusion neutron yield by the neutron driver. Fusion neutron yield can temporarily drop to any fraction from 100 percent and 0 percent due to focusing issues, electrical arcing, or other malfunctions within the accelerator.

As the neutron source effects generate substantial reactivity defect in the system, this loss of source creates positive reactivity insertion. During a loss of neutron source, void fractions decrease as bubbles leave the solution and temperatures begin to cool down, leading to reactivity increase.

The timing of this event was modeled for nominal conditions using TRIAD. Results showed that the transient happens on a short time scale, therefore a bounding analysis was made for the maximum power.

For this event, it is assumed that the accelerator decreases to 0 percent neutron yield. During this event, it is assumed all void leaves the target solution, leading to a reactivity increase of up to []^{PROP/ECI}. The temperature decreases, and within []^{PROP/ECI}, the TSV temperature has decreased by up to 11.2°F (6.2°C). This creates a reactivity increase of up to []^{PROP/ECI}. The final reactivity of the system is at least as negative as []^{PROP/ECI} below the startup k_{eff} .

It is assumed that the accelerator instantaneously returns to 122 percent output at peak reactivity. This results in a peak TSV power of []^{PROP/ECI}. The peak TSV power density is []^{PROP/ECI}. Following the peak power, the thermal and void feedback effects rapidly result in a decrease in power and the assembly output stabilizes with normal operating parameters. The safety limits of the assembly are not challenged, and therefore, there is no damage to the PSB.

Additional Target Solution Injection During Fill/Startup and Irradiation Operations

Target solution injection directly adds fissile material to the TSV, and results in reactivity increase. Multiple administrative controls, passive safety features, and active safety features are in place to limit the reactivity effects from additional target solution injection at TSV fill (Mode 1). Target solution injection during irradiation (Mode 2) is not considered credible due to redundant isolation valves interlocked with Mode 1 and the location of the TSV hold tank physically below the TSV.

In Mode 1, the limiting transient event (i.e., the one calculated to yield the highest reactivity values) is an inadvertent fill of solution after the target solution has already been filled to normal startup k_{eff} values. This could occur due to operator error or equipment malfunction.

The second postulated scenario is precipitation of uranium solids from the solution. Precipitation of uranium solids due to uranyl peroxide formation is possible in aqueous reactors. In the SHINE system, chemistry, power density, and temperature limits have been placed on the target solution as described in [Subsection 4a2.6.3.5](#). Given these limits, no significant precipitation is expected.

Summary of Credible Inadvertent Insertions of Reactivity

The controls that prevent or provide mitigation for the consequences of an excess reactivity insertion event are listed in [Subsection 13a2.2.2](#) for the seven initiating events discussed. Given these controls, none of the analyzed events result in damage to the PSB.

4a2.6.3.4 Negative Reactivity

While typical reactors must have sufficient negative reactivity available in control rods to ensure the reactor can be shut down safely, the subcritical assembly operates safely without control rods. Reactors are required to characterize the amount of negative reactivity available to be added to the core. In the SHINE system, the target solution is moved for shutdown rather than adding poisons or control rod worth. Therefore, the concept of negative reactivity is different in the SHINE system.

In the SHINE system, the subcritical assembly is shut down by moving the target solution to a subcritical location (i.e., the TSV dump tank). The SHINE subcritical assembly is capable of being shutdown to $k_{\text{eff}} < 0.954$ for any analyzed condition by transferring the target solution to the TSV dump tank.

The important parameters for ensuring safe shutdown are: (1) the redundancy and reliability of transferring the solution to the TSV dump tank (described in [Subsection 4a2.6.3.7](#)), (2) the rate of target solution transfer to the TSV dump tank (described below), and (3) the reactivity analysis once the solution is in the TSV dump tank (described below).

Rate of Target Solution Transfer to the TSV Dump Tank

When analyzing transient or accident scenarios involving the shutdown of the TSV, the following parameters, or more conservative values, are used for the drain time, delay time, and opening time of the dump valves. The TSV drain system must drain the TSV within 183 seconds with a liquid volume in the TSV of []^{PROP/ECI}, which is the limiting core configuration. The design drain time is conservatively based on only one drain line being available.

The delay time between neutron flux levels exceeding predetermined limits and the start of the dump valves opening is a maximum of one second. The duration of time it takes for the dump valves to fully open is less than 2.0 seconds.

The specified drain time, delay time, and opening time ensure safe shutdown of the subcritical assembly in anticipated transients and postulated accident conditions.

Reactivity Analysis in the TSV Dump Tank

The TSV dump tank reactivity has been analyzed using MCNP5. The V&V for MCNP5 is discussed in [Subsection 4a2.6.2.1](#). The k_{eff} calculation methodology and how uncertainties in the

methodology are accounted for are discussed in [Subsection 4a2.6.2.6.1](#). The methodology ensures that k_{eff} in the dump tank remains less than 0.954.

The TSV dump tank is analyzed to remain safely shutdown for the most reactive uranium concentration. The most reactive uranium concentration was found by calculating k_{eff} for a range of uranium concentrations that spanned the peak reactivity (approximately 1000 gU/L). Then, SHINE selected the concentration that resulted in the highest reactivity and used this concentration for calculating dump tank k_{eff} .

The most reactive uranium concentration results in an increase of approximately []^{PROPECI} relative to the nominal concentration. This methodology ensures that the TSV dump tank will be subcritical at any uranium concentration, which provides very high confidence in shutdown margin for the range of normal conditions and accident scenarios. This significantly increases margins when the system contains expected uranium concentrations, as specified in [Table 4a2.2-2](#).

The TSV dump tank reactivity increases as the target solution temperature cools down from its operating temperature of approximately 118°F (48°C) to the light water pool temperature of approximately 68°F (20°C). Dump tank reactivity is calculated assuming the target solution has cooled down and achieved equilibrium with the pool.

Abnormal conditions were also evaluated within the TSV dump tank, including a design basis seismic event, excessive corrosion, overfilling, salt accumulation, and water intrusion. The increases in k_{eff} due to the single abnormal conditions analyzed do not result in k_{eff} values exceeding 0.954. Therefore, the dump tank is able to maintain the solution in a subcritical state when undergoing these analyzed single abnormal conditions.

Normal electrical power is not required to shut down the subcritical assembly or maintain it in a safe shutdown condition. After a loss of normal electrical power, the target solution is allowed to remain in the TSV for up to 3 minutes, with the dump valves receiving power from the uninterruptible electrical power supply system (UPSS). After this period of time, the TSV dump valves are automatically opened by TRPS disconnecting power to the valves, resulting in a dump of the solution to the TSV dump tank. The TSV dump tank does not require active cooling given the low decay heat loading of the target solution. Rejection of decay heat is achieved through passive convection with the light water pool.

Safety-related electrical power from the UPSS is required by the TOGS for 5 minutes following a loss of off-site power in order to maintain hydrogen concentrations at acceptable levels in the PSB. See [Section 4a2.8](#).

Transient poisons, such as xenon, are not credited in the reactivity analysis.

Verification of the k_{eff} and shutdown margin in the TSV dump tank is not required. This approach to safety is acceptable given the [large subcritical](#) margin ~~to critical~~ of 0.056 Δk , the consideration of relevant uncertainties in the calculation process, and the consideration of abnormal conditions to which the vessel may be exposed while still maintaining [the subcritical](#) margin ~~to critical~~ of 0.05 Δk .

The subcritical assembly is capable of being safely shutdown for any postulated reactivity loading in the TSV. In the TSV dump tank, the target solution is maintained below a k_{eff} of 0.954.

The dump system consists of two completely independent flow paths between the TSV and the TSV dump tank. The physical design connects the flow paths to different parts of the TSV and dump tank. Each path consists of a dump line from the TSV to the TSV dump tank, and a dump valve to control the drainage of the target solution into the TSV dump tank. Two completely independent overflow lines are also present, which serve as vent lines from the TSV dump tank to the TSV to equalize gas pressures during solution dumps.

The dump valves are highly reliable fail-open units designed for service in the environmental conditions present. Both valves are actuated by isolating power to them, which is a method resistant to common mode failures. Both TRPS and PICS can independently open the TSV dump valves. The PICS and TRPS are configured in series configuration for the TSV dump valve control. When the TRPS output is energized, the PICS has control of the TSV dump valves independent from the TRPS. If either system de-energizes the output, the dump valves open.

Each dump valve is equipped with a valve position indicator, which immediately alerts the operator of a failure of the valve to respond. Any failure of a valve to respond to a commanded signal will be thoroughly investigated and corrected, as part of the corrective action program, to ensure the valves can be relied upon when required.

Valves are maintained appropriately to ensure high reliability. Design considerations allow for underwater maintenance of the valves, when needed. The valves are designed for a lifetime of 30 years.

There is internal redundancy within the TRPS such that a single failure does not result in a spurious actuation. Either TRPS or PICS can open the dump valves.

As the dump valves are actuated with each irradiation cycle, they undergo regular normal cycling. This frequent actuation provides data that could indicate degraded performance prior to failure to perform their safety function. A decrease in drain rate indicates potential for dump line blockage, overflow line blockage, or valve failure to fully open. An increase in valve opening time indicates potential future valve failure. SHINE will monitor drain rates and opening time at least yearly to ensure early indication of failures are identified.

Given the high valve reliability, automatic valve opening on control system or electrical power failure, and ability to frequently actuate and trend performance of the dump valves, no additional shutdown mechanisms are required for ensuring target solution can be shut down safely.

The target solution is maintained in a criticality-safe shutdown condition (k_{eff} less than 0.954) in locations outside the TSV by passive engineered controls. TRPS IU Cell Safety Actuations also lead to de-energizing the HVPS of the NDAS, which eliminates fusion neutron production and terminates the fission process within the subcritical assembly.

The TSV dump valves and TPS provide a high degree of confidence in the ability to drain the target solution to a safe shutdown configuration.

4a2.6.3.8 Technical Specifications

Certain material in this subsection provides information that is used in the technical specifications. This includes limiting conditions for operation, setpoints, design features, and

Table 4a2.6-6 – Calculated Core Effective Multiplication Factors

	k_{eff} in Mode 1 (After Filling)	k_{eff} in Mode 2 (Steady-State^(a))	k_{eff} in Mode 3 (After Solution has Dumped)
Nominal core configuration	[] ^{PROP/ECI}	[] ^{PROP/ECI}	< 0.9 5 <u>4</u>
Limiting core configuration	[] ^{PROP/ECI}	[] ^{PROP/ECI}	< 0.9 5 <u>4</u>

(a) At 125 kW fission power, using best estimate temperatures and void fractions

There is no significant nitrogen oxide (NO_x) gas present in the off-gas; therefore, there is no postulated accident scenario resulting in the release or accumulation of NO_x gas. The SHINE target solution is a sulfuric/sulfate system. Nitric acid is not used to prepare the target solution or to adjust the target solution chemistry.

Additionally, no significant amount of SO_x gas is present in the off-gas. Sulfuric/sulfate was chosen as the acid/counter ion system because of the stability it maintains in the presence of radiation. Furthermore, the vapor pressure of sulfuric acid is known to be extremely low, so very little SO_x gases will leave the liquid phase. Therefore, it is not necessary to consider scenarios related to SO_x gas in the abnormal conditions of the TOGS.

Pressure safety valves are connected to the PSB piping to passively prevent an over-pressurization of the PSB, which may cause structural damage to the IU or malfunction of TOGS. The setpoint of the pressure safety valves does not exceed the design pressure of the PSB components. The pressure safety valves are connected to the PVVS. PVVS is capable of receiving the calculated maximum gas relieving rate from TOGS. The relief gas is then processed through the PVVS filters, guard beds, and carbon delay beds to remove particulates, remove iodine, and sufficiently delay noble gas release. This process ensures that the radioactive release and dose requirements of 10 CFR 20 are met. See [Subsection 9b.6.1](#) for a discussion on the PVVS.

Transients can occur in the nuclear system due to pressure fluctuations, neutron driver interruptions, cooling system malfunctions, and other causes. See [Subsection 4a2.6.1](#) for kinetic behavior of the TSV. Variations in TSV power lead to variations in hydrogen and oxygen generation rates. TOGS is designed to handle transient and accident hydrogen generation rates while maintaining hydrogen concentrations in the PSB below those that could cause damage to the PSB.

SHINE has considered the long-term accumulation of fissionable material entrained in the system. Long term accumulation of material could lead to flow blockages and subsequent system malfunction, or it could present a hazard for inadvertent criticality. Inadvertent criticality is discussed below. Monitoring is performed for flow blockages due to long term accumulation of material by periodically trending system flow rates. Long term accumulation would result in changes in pressure drops in the system, especially in the demisters and catalytic recombiner beds.

4a2.8.5.1 Protection Against Inadvertent Criticality

The potential exists for fissile material from the TSV, such as uranium solution droplets, to enter TOGS. Water leakage from the light water pool could cause flooding of the target solution into TOGS. Droplet carryover from TOGS could lead to uranium entrainment in TOGS. Fissile material, without proper design, could lead to inadvertent criticality in TOGS.

To prevent the potential for an inadvertent criticality in TOGS, the sections of TOGS that form a portion of the PSB are designed to be geometrically favorable if fully flooded. Analyses are performed in accordance with the methodology described in [Subsection 4a2.6.2.6.1](#). The TOGS k_{eff} analysis is evaluated at the most reactive uranium concentration, which ensures the system will be subcritical at any uranium concentration. As discussed in [Subsection 4a2.6.2.6.1](#), TOGS is designed to a k_{eff} value of less than 0.954 at the most reactive uranium concentration and at cold conditions.

Since TOGS is designed to be at a k_{eff} below 0.954 even if fully flooded at the most reactive uranium concentration, TOGS is protected against inadvertent criticality.

4a2.8.6 RADIATION AND HYDROGEN CONCENTRATION CONTROL/MONITORING

The TOGS is connected to the vacuum transfer system (VTS) for purging between irradiations, as needed. The VTS discharges collected TOGS gases to the PVVS, where they are treated through the PVVS filters, guard beds, and carbon delay beds. Purging the off-gas to the VTS allows SHINE to decrease gaseous fission products contained within the system. TOGS is purged to VTS prior to maintenance operations in which lower dose rates in and surrounding the TOGS equipment are desired. The TOGS components are designed and shielded to limit personnel exposure to radiation.

Hydrogen concentration monitoring instrumentation is included to measure the concentration of hydrogen in the TOGS piping. The TOGS is designed to maintain hydrogen concentrations at or less than 2 percent during normal operation.

If the hydrogen concentration exceeds 2.5 percent by volume, an alarm alerts the operator to take action. If the neutron driver is shut down, the blowers and recombiners remain active to circulate and recombine the hydrogen and oxygen 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 operating limit of 3 percent while not resulting in excessive alarms that distract the operators in the control room. The hydrogen concentration limit of 3.0 percent provides sufficient margin to hydrogen concentrations that could result in a deflagration pressure exceeding 65 psia should the failure of a single active component occur. A minimum TOGS mainstream flow of []^{PROP/ECI} and a TOGS dump tank flow of []^{PROP/ECI} is required to ensure hydrogen can be maintained below this limit.

The worst postulated single active failure is that of the blower ventilating the TSV dump tank. The TSV reactivity protection system (TRPS) detects loss of flow and initiates an IU Cell Safety Actuation and an IU Cell Nitrogen Purge. This opens the TSV dump valves and de-energizes the high voltage power supply to the neutron driver, rapidly reducing hydrogen production. Conservatively assuming that the TOGS and TSV were uniformly at 3.0 percent hydrogen concentration prior to the trip, the peak hydrogen concentration has been calculated. This peak hydrogen concentration results in deflagration pressures less than 65 psia assuming a deflagration occurred immediately at the peak concentration.

Oxygen concentration monitoring instrumentation is also included to measure the concentration of oxygen in the TOGS piping. Oxygen holdup in the target solution can lead to non-stoichiometric releases of hydrogen and oxygen from the solution. A minimum oxygen concentration of 10 percent is required to ensure hydrogen recombination in the TSV off-gas recombiner occurs satisfactorily.

TOGS condenser demister outlet temperature sensors monitor the health of the condenser demisters. A temperature over 25°C is indicative of a failure of the condenser demister, which could lead to increased water holdup in TOGS and potential reduction in hydrogen recombination. In the event of a failure, the TRPS would initiate an IU Cell Nitrogen Purge.

Table 6a2.1-2 – Comparison of Unmitigated and Mitigated Radiological Doses for Select Irradiation Facility DBAs

Representative DBA	Unmitigated Public Dose (rem)			Mitigated Public Dose (rem)		
	Public TEDE	Worker TEDE	Worker Limiting Organ	Public TEDE	Worker TEDE	Worker Limiting Organ
Mishandling or Malfunction of Target Solution (Primary Confinement Boundary – IU Cell)	5.03E+00 1	4.13.7E+02 1	2.48.6E+03 2	64.5E-02 1	1.52E+00	2.3.0E+00 1
Mishandling or Malfunction of Equipment (Primary Confinement Boundary – TOGS Cell)	4.95.3E+00 1	4.03.7E+02 1	2.38.6E+03 2	27.3E-01	4.81.9E+00	4.2.8E+01
Facility-Specific Events (Tritium Confinement Boundary)	2.75E+00 1	3.18.6E+02 1	3.08.6E+02 1	4.8.0E-01	2.51.4E- +040	21.4E-+040

Table 6b.1-2 – Comparison of Unmitigated and Mitigated Radiological Doses for Select Radioisotope Production Facility DBAs

Representative DBA	Unmitigated Public Dose (rem)			Mitigated Public Dose (rem)		
	Public TEDE	Worker TEDE	Worker Limiting Organ	Public TEDE	Worker TEDE	Worker Limiting Organ
Critical Equipment Malfunction (Process Confinement Boundary - Supercell)	7.9E-04 8.0E+00	5.2E+01 1.7E+01	4.6E+02 2.5E+02	6.8E-03 4.2E-03	3.2E-04 7.6E-04	5.2E+00 9E+00
Critical Equipment Malfunction (Process Confinement Boundary - Below Grade)	7.6E-04 8.0E+00	5.1E+01 1.7E+01	4.2E+02 4E+02	5.2E-03 2.4E-03	4.0E-04 2E-04	2.9E+00 9E+00

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6b.1-1	Summary of Engineered Safety Features and Design Basis Accidents Mitigated
6b.1-2	Comparison of Unmitigated and Mitigated Radiological Doses for Select Radioisotope Production Facility DBAs
<u>6b.3-1</u>	<u>Summary of Benchmarks Selected for the SHINE Validation Report</u>
<u>6b.3-2</u>	<u>Area of Applicability Summary</u>

6b.3.1.2 Nuclear Criticality Safety Staff Qualifications

The minimum qualification entry requirements for NCS staff are:

~~Fissile Material Handler: Qualification as a radiation worker~~

NCS Analyst: Baccalaureate degree in science or engineering from an accredited college or university, or at least five years of directly applicable experience, or an equivalent combination of education and experience.

NCS Engineer: Same as for an NCS Analyst

Senior NCS Engineer: Current qualifications as an NCS Engineer, plus three years of experience as an NCS Engineer

NCS qualifications use a tiered approach, with three qualification levels for NCS Staff and specific functional area qualifications for Fissile Material Handlers. The specific training requirements are taken from ANSI/ANS-8.26-2007 (R2016), *Criticality Safety Engineer Training and Qualification Program* (ANSI/ANS, 2007a). SHINE uses qualification cards to record an individual's progress towards qualification. Qualification cards list the necessary knowledge and performance requirements for NCS staff and provide a record of completion for qualification activities. Assignment of personnel for qualification is made by an engineering manager. Maintenance of qualifications is required for NCS staff.

Qualifications granted by external organizations may be recognized based on verification and completion of SHINE facility-specific portions of the appropriate qualification card. Experience in NCS may be used to exempt individual training and qualification requirements. Where experience is used for exemptions, appropriate documentation is attached to the qualification card and retained. Facility familiarity and walk-through requirements may not be exempted and are required in addition to recognition of externally-completed qualifications. Maintenance of qualifications is required for NCS staff.

6b.3.1.3 Use of National Consensus Standards

The CSP commits to the requirements of the following national consensus standards, subject to the clarifications and exceptions identified in RG 3.71, with certain SHINE-specific limitations described below:

Standards endorsed without clarifications or exceptions by the Nuclear Regulatory Commission (NRC) in RG 3.71:

- ANSI/ANS-8.6-1983 (R2017), Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ (ANSI/ANS, 1983)
- ANSI/ANS-8.7-1998 (R2017), Nuclear Criticality Safety in the Storage of Fissile Materials (ANSI/ANS, 1998)
- ANSI/ANS-8.19-2014, Administrative Practices for Nuclear Criticality Safety (ANSI/ANS, 2014a)
- ANSI/ANS-8.20-1991 (R2015), Nuclear Criticality Safety Training (ANSI/ANS, 1991)
- ~~ANSI/ANS-8.21-1995 (R2011), Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors (ANSI/ANS, 1995)~~

- ANSI/ANS-8.22-1997 (R2016), Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI/ANS, 1997a)
- ANSI/ANS-8.26-2007 (R2016), Criticality Safety Engineer Training and Qualification Program

Standards endorsed in RG 3.71 with clarifications or exceptions:

- ANSI/ANS-8.1-2014, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI/ANS, 2014b)
The clarification applied to this standard is related to subcritical limits for plutonium isotopes and is not applicable to the SHINE facility.
- ANSI/ANS-8.3-1997 (R2017), Criticality Accident Alarm System (ANSI/ANS, 1997b)
The clarifications and exceptions applied to this standard are applicable to the SHINE facility.
- ~~ANSI/ANS-8.10-2015, Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement (ANSI/ANS, 2015)
The clarifications applied to this standard are applicable to the SHINE facility.~~
- ANSI/ANS-8.23-2007 (R2012), Nuclear Criticality Accident Emergency Planning and Response (ANSI/ANS, 2007b)
The clarification applied to this standard is applicable to the SHINE facility.
- ANSI/ANS-8.24-2017, Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations (ANSI/ANS, 2017)
The clarifications applied to this standard are applicable to the SHINE facility.

The following ANSI/ANS Series 8 Standards are not used by the SHINE CSP. For each standard, the basis for non-implementation is provided:

- ANSI/ANS-8.5-1996 (R2017), Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material.
Borosilicate-glass Raschig rings are not used in the SHINE facility.
- ANSI/ANS-8.10-2015, Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement (ANSI/ANS, 2015).
SHINE does not apply the criteria provided in this standard for determining the adequacy of shielding and confinement.
- ANSI/ANS-8.12-1987 (R2016), Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors.
Plutonium is not used as a fuel component at SHINE. Only small quantities are present due to burnup.
- ANSI/ANS-8.14-2004 (R2016), Use of Soluble Neutron Absorbers in Nuclear Facilities Outside Reactors.
SHINE does not use soluble neutron absorbers for control of criticality.
- ANSI/ANS-8.15-2014, Nuclear Criticality Control of Selected Actinide Nuclides.
SHINE does not conduct operations with non-negligible quantities of the selected actinide nuclides.
- ANSI/ANS-8.17-2004 (R2014), Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
SHINE does not handle, store, or transport LWR fuel rods or units.
- ANSI/ANS-8.21-1995 (R2011), Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors (ANSI/ANS, 1995)
SHINE does not use fixed neutron absorbers for control of criticality.

- ANSI/ANS-8.27-2015, Burnup Credit for LWR Fuel.
SHINE does not possess irradiated LWR fuel assemblies.

6b.3.1.4 Nuclear Criticality Safety Evaluations

NCSEs are conducted for each FMO within the radioisotope production facility (RPF) to ensure that under normal and credible abnormal conditions, all nuclear processes remain subcritical with an approved margin of subcriticality for safety. An FMO is any process or system that has the potential to contain more than 250 g of non-exempt fissile material. This limit is selected based on one-half of the single parameter mass limit for uranium-233 identified in ANSI/ANS-8.1-2014. For the purposes of application of this limit, all fissionable isotopes in the process or system are considered to be fissile.

Exempt fissile material is defined as special nuclear material (SNM) that meets the requirements from classification as fissile nuclear material as specified in 10 CFR 71.15. The limits specified in 10 CFR 71.15 are derived for use in nuclear material transport and long-term storage and are acceptably conservative. When 10 CFR 71.15 is invoked to exempt a process or system, the NCSE must show that there are no credible means of changing the physical composition or configuration of the material.

NCS limits are derived based on assuming optimum or most-reactive credible parameter values unless specific controls are implemented to limit parameters to a particular range. If less-than-optimum values are used, the basis for use is included in the NCSE. Operating limits which take process variability and uncertainty into account are used to ensure NCS limits are unlikely to be exceeded. Controls used to enforce safety and operating limits are specified in the NCSEs.

~~The subcritical margin used for RPF solution processes is 0.06. An initial 0.05 margin was selected based on completion of a validation study that meets the guidelines of NUREG/CR 6698, Guide for Validation of Nuclear Criticality Safety Methodology, using a wide variety of critical experiment benchmarks that encompass the material compositions, neutron energy spectra, and geometric forms (USNRC, 2001) to bound SHINE solution processes. An additional subcritical margin of 0.01 is applied for conservatism to account for the limited number of experimental benchmarks for uranyl sulfate systems.~~

The NCSEs are conducted using appropriate hazard evaluation techniques, including "What-if," "What-if Checklist," and Event Tree Analysis, to determine potential scenarios which could result in an inadvertent criticality event. Process hazards evaluations are referenced to identify additional potential scenarios that have been determined to have potential criticality safety implications (e.g. chemical safety, fire, radiological events). The identified scenarios are screened based on a qualitative determination of likelihood and those events which are deemed to be credible are evaluated for appropriate control selection. For the purposes of NCSEs, criticality events are always considered to be "high" consequence, with a strict emphasis on selection of controls to prevent criticality. Where the double contingency principle (DCP) is employed, the NCSE contains a description of its implementation.

The NCS limits used in the evaluations are derived from industry-accepted and peer-reviewed references, including ANS standards; from hand calculations using industry-accepted and peer-reviewed techniques, such as solid-angle or surface density calculation; or from computational methods. In cases where hand calculations are used, each technique is used consistent with any limitations.

6b.3.1.5 Computational System Validation

Where computational methods are employed, the computational system is verified and validated using the guidance in NUREG/CR-6698. ~~Reverification of the computational system is conducted following any changes to the hardware or operating system (USNRC, 2001).~~

A written validation report for the computational systems used for NCS calculations is documented and maintained in accordance with the SHINE document control process. The validation process was performed using Monte Carlo n-Particle (MCNP) software, version MCNP5-1.60. Verification of the MCNP software installation was performed using developer-supplied verification tools, and re-verification of the computational system is conducted following any changes to the hardware or operating system.

The validation report uses benchmarks from the Handbook of the International Criticality Safety Benchmark Evaluation Project (ICBEP). Benchmarks were selected for evaluation based on their similarity to the SHINE solution system, as no plant-specific benchmark experiments are available. The fissile material, enrichment, chemical form, range of concentration, and reflector materials were considered in the selection of benchmarks. The selected benchmarks series, number of cases selected from each benchmark series, and a description of each physical system is provided in Table 6b.3-1. A summary of the area of applicability covered by the validation report is provided in Table 6b.3-2.

The bias and bias uncertainty were calculated using the methodology described in NUREG/CR-6698. The benchmark data were tested using a modified Shapiro-Wilk test for normality and were determined to be normally distributed. A single-sided tolerance limit approach was used to determine the bias uncertainty. The upper subcritical limit is the difference between unity and the sum of the bias (zero, because a positive bias was determined), the bias uncertainty, and the subcritical margin.

The margin of subcriticality used for SHINE solution processes is 0.06. A subcritical margin of 0.05 was conservatively selected based on the quantity and quality of the selected benchmarks. An additional subcritical margin of 0.01 is applied to provide additional conservatism to account for the limited number of experimental benchmarks specific to uranyl sulfate systems. NCSEs ensure that the evaluated processes fall within the range of the validated computational system. The validation range may be extended beyond the range of the benchmark data using additional subcritical margin or bias trending analysis to ensure that the existing subcritical margin is appropriate. Where extrapolation or wide interpolations are used to extend the validation range, the recommendations of NUREG/CR-6698 are used. When a positive bias is encountered, it is set to 0 for the purposes of calculating subcritical limits, and data outliers are only rejected based on inconsistency with known physical behavior; statistical rejection methods for outliers are not used. NCS limits are selected to incorporate appropriate margins to protect against uncertainty in process variables and to prevent a limit being accidentally exceeded. Allowances for uncertainty in the methods, data, and bias are included in the selected limits. Studies are conducted to correlate the effects of changing one controlled parameter on other controlled parameters, such as to evaluate compliance with the ~~double contingency principle (DCP).~~

~~In general, RPF processes comply with the DCP. If the DCP cannot be employed, consideration is given to the use of ANSI/ANS-8.10-2015, to allow single-contingency operations or mitigation of consequences. Where the DCP is employed, the evaluation reports contain an explicit description of its implementation.~~

NCS program documentation, evaluations, and calculations are maintained in accordance with the SHINE records management system. Equipment characteristics relied on to maintain NCS limits are identified as NCS controls and are maintained by the SHINE configuration management system.

Process or design changes that could affect NCS limits or controls are evaluated using the facility change process requirements of 10 CFR 50.59. ~~Such changes include new design, operation, or modification to existing SSCs; computer programs; processes; operating procedures; or administrative controls.~~ Prior to implementing the change, the NCSE is reviewed and updated if needed to determine that the entire process will be subcritical under both normal and credible accident scenarios.

6b.3.1.6 Nuclear Criticality Safety Training

In support of SHINE's CSP, a two-tiered NCS training program is established and maintained. The first-tier training program includes the Program Content identified in ANSI/ANS-8.20-1991 (R2015), and is directed toward those who manage, work in, or work near areas where the potential exists for a criticality accident. The second-tier training is specific to NCS staff. NCS staff training meets the requirements identified in ANSI/ANS-8.26-2007 (R2016). Both tiers of NCS training include procedural compliance, stop-work authority, response to criticality alarms, and reporting of defective conditions.

6b.3.1.7 Criticality Safety Program Oversight

Operations are reviewed at least annually to verify that procedures are being followed and that process conditions have not been altered to affect the NCSE. NCS staff conduct walkthroughs of facility processes and procedures as part of the annual operational review. These reviews are conducted, in consultation with operating personnel, by individuals who are knowledgeable in NCS and who, to the extent practicable, are not immediately responsible for the operation, and are documented. Active procedures are reviewed periodically by supervisors.

The NCS Lead schedules and coordinates routine NCS oversight activities:

- NCS staff conduct and participate in routine audits of NCS practices, including compliance with procedures.
- NCS staff examine reports of procedural violations and other deficiencies for possible improvement of safety practices and procedural requirements. Findings are reported to management.
- NCS staff periodically review NCSEs to determine their continued applicability and validity. This should include a review of elements of the evaluation such as scope, assumptions, normal conditions, credible abnormal conditions, controls, and limits. Annual reviews of NCSEs and calculations are conducted, with each evaluation and calculation being reviewed at least once every three years.
- At least every three years, an audit of the overall effectiveness of the CSP is performed. Management participates actively in this activity.

Equipment and procedures needed for NCS controls are clearly identified. Activities involving fissile material are conducted using written and approved procedures. For situations in which approved procedures are inadequate or do not exist, personnel are required to take no action until the NCS staff has evaluated the situation and provided recovery instructions. Procedures

are supplemented by appropriate material labeling and postings, specifying material identification and limits on parameters, in areas, operations, workstations, and storage locations subject to procedural controls. Equipment and procedures are maintained as part of the facility management measures.

6b.3.1.8 Criticality Safety Nonconformances

The adequacy of engineered and administrative NCS controls is routinely assessed as part of the SHINE facility audits and inspections. Deviations from procedures and unintended alterations in process conditions that affect NCS are promptly reported to management using the corrective action program, investigated promptly, corrected as appropriate, and documented. Action to correct such deviations or alterations is taken in accordance with procedural requirements and with guidance obtained from the NCS staff. Action is taken to prevent recurrence for significant conditions adverse to quality. Records of NCS deficiencies and associated corrective actions are maintained in the corrective action program.

Upon the loss of double contingency protection, operations are suspended and processes rendered safe until double contingency protection can be restored. Adequacy of the affected controls is subsequently assessed as part of the corrective actions.

NCS events are reported to the NRC in accordance with the reporting requirements of 10 CFR 70.50, 10 CFR 70.52, and 10 CFR Part 70, Appendix A.

6b.3.1.8.1 Planned Response to Criticality Accidents

The CAAS is described in [Subsection 6b.3.3](#).

SHINE maintains an emergency plan which includes the planned response to criticality accidents. The emergency plan contains information on the provision of personnel accident dosimeters in areas that require the CAAS and arrangements for on-site decontamination of personnel and the transport and medical treatment of exposed individuals. The SHINE emergency plan is further described in [Section 12.7](#).

6b.3.1.8.2 Criticality Safety Event Reporting

Facility procedures include provisions for rapid evaluation of the significance of NCS events, including immediate notifications of facility NCS staff and the assessment of events with respect to the loss or degradation of double contingency protection.

The significance and reportability of NCS events is based on the loss or degradation of NCS controls and not on the event sequence with respect to whether or not limits were exceeded.

If an NCS event cannot be affirmatively determined to not require a one-hour report within one hour, it is reported as an event requiring a one-hour report.

6b.3.2 CRITICALITY SAFETY CONTROLS

General

Table 6b.3-1 – Summary of Benchmarks Selected for the SHINE Validation Report

<u>Benchmark Series</u>	<u>Cases</u>	<u>Description of Physical Systems</u>
<u>LEU-SOL-THERM-003</u>	<u>9</u>	<u>10.06% enriched uranyl nitrate, un-reflected</u>
<u>IEU-SOL-THERM-002</u>	<u>13</u>	<u>30.45% enriched uranyl fluoride, water-reflected and un-reflected</u>
<u>IEU-SOL-THERM-003</u>	<u>46</u>	<u>30.3% uranyl fluoride, water-reflected and un-reflected</u>
<u>IEU-SOL-THERM-004</u>	<u>1</u>	<u>14.7% uranyl sulfate, reflected by beryllium oxide</u>
<u>LEU-SOL-THERM-004</u>	<u>7</u>	<u>9.97% enriched uranyl nitrate, water-reflected</u>
<u>LEU-SOL-THERM-007</u>	<u>5</u>	<u>9.97% enriched uranyl nitrate, un-reflected</u>
<u>LEU-SOL-THERM-008</u>	<u>4</u>	<u>9.97% enriched uranyl nitrate, concrete-reflected</u>
<u>LEU-SOL-THERM-016</u>	<u>7</u>	<u>9.97% enriched uranyl nitrate, water-reflected</u>
<u>LEU-SOL-THERM-017</u>	<u>6</u>	<u>9.97% enriched uranyl nitrate, un-reflected</u>
<u>LEU-SOL-THERM-018</u>	<u>6</u>	<u>9.97% enriched uranyl nitrate, concrete-reflected</u>
<u>LEU-SOL-THERM-020</u>	<u>4</u>	<u>9.97% enriched uranyl nitrate, water-reflected</u>
<u>LEU-SOL-THERM-021</u>	<u>4</u>	<u>9.97% enriched uranyl nitrate, un-reflected</u>
<u>LEU-SOL-THERM-023</u>	<u>9</u>	<u>9.97% enriched uranyl nitrate, un-reflected</u>
<u>LEU-SOL-THERM-025</u>	<u>7</u>	<u>9.97% enriched uranyl nitrate, concrete-reflected</u>

Table 6b.3-2 – Area of Applicability Summary

<u>Parameter</u>	<u>Area of Applicability</u>
<u>Fissile Material and Composition</u>	<u>Uranyl Sulfate</u> <u>Uranyl Nitrate</u> <u>Uranyl Fluoride</u>
<u>Chemical Form</u>	<u>Solution</u>
<u>Average Neutron Energy Causing Fission (ANECF) (MeV)</u>	<u>0.004-0.064</u>
<u>Enrichment (wt. %)</u>	<u>10-30.5</u>
<u>Reflector Materials</u>	<u>None</u> <u>Water</u> <u>Graphite</u> <u>Beryllium Oxide</u> <u>Concrete</u>
<u>Uranium Concentration (g-U/L)</u>	<u>52.8-960</u>
<u>H/²³⁵U Ratio</u>	<u>75-1610</u>

13a2 IRRADIATION FACILITY ACCIDENT ANALYSIS

The purpose of this section is to identify the postulated initiating events and credible accidents that form the design basis for the irradiation facility (IF), which includes the irradiation units (IUs) and supporting systems. [Section 13b](#) identifies the postulated initiating events and credible accidents within the radioisotope production facility.

Design basis accidents were identified using the following sources of information:

- NUREG-1537 (USNRC, 1996) and the Interim Staff Guidance Augmenting NUREG-1537 (USNRC, 2012a);
- Process hazard analysis method within the safety analysis; and
- Experience of the hazard analysis team.

Each identified accident scenario was qualitatively evaluated for its potential chemical or radiological consequences. For accident scenarios with potential consequences that could exceed the appropriate evaluation guidelines for worker or public exposure, controls were applied to ensure that the scenario is prevented or that consequences are mitigated to within acceptable limits. For accident scenarios which are not prevented, the radiological or chemical consequences were quantitatively evaluated to demonstrate the effectiveness of the selected mitigative controls or shown to be bounded by other quantitative analysis.

The quantitative analysis includes:

- 1) Identification of the limiting initiating event, initial conditions, and boundary conditions.
- 2) Review of the sequence of events for functions and actions that change the course of the accident or mitigate the consequences.
- 3) Identification of damage to equipment or the facility that affects the consequences of the accident.
- 4) Review of the potential radiation source term and radiological consequences.
- 5) Identification of safety controls to prevent or mitigate the consequences of the accident.

The results of these analyses are provided in [Section 13a3](#). The analyses identify those safety-related structures, systems, and components (SSCs) and engineered safety features for each accident, and demonstrate that the mitigated consequences do not exceed the radiological accident dose criteria, described in [Section 13a2.2](#).

13a2.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

The design basis accidents (DBAs) identified in this section are credible accident scenarios that range from anticipated events, such as a loss of electrical power, to events that are still credible, but considered unlikely to occur during the lifetime of the plant. The ~~IF~~-maximum hypothetical accident (MHA) is also defined to result in the bounding radiological consequences for the ~~IF~~[SHINE facility](#).

Based on the guidance provided in the Interim Staff Guidance (ISG) Augmenting NUREG-1537 (USNRC, 2012a), the following accident categories were used to identify potential accident sequences:

- Maximum hypothetical accident ([Subsection 13a2.1.1](#))
- Excess reactivity insertion ([Subsection 13a2.1.2](#))
- Reduction in cooling ([Subsection 13a2.1.3](#))
- Mishandling or malfunction of target solution ([Subsection 13a2.1.4](#))
- Loss of off-site power (LOOP) ([Subsection 13a2.1.5](#))
- External events ([Subsection 13a2.1.6](#))
- Mishandling or malfunction of equipment ([Subsection 13a2.1.7](#))
- Large undamped power oscillations ([Subsection 13a2.1.8](#))
- Detonation and deflagration in the primary system boundary ([Subsection 13a2.1.9](#))
- Unintended exothermic chemical reactions other than detonation ([Subsection 13a2.1.10](#))
- System interaction events ([Subsection 13a2.1.11](#))
- Facility-specific events ([Subsection 13a2.1.12](#))

The effects of losses of electrical power and operator errors were considered as initiating events within the scope of the process hazard analysis (PHA) process and are therefore considered within each event category.

13a2.1.1 ~~IF~~-MAXIMUM HYPOTHETICAL ACCIDENT

~~In accordance with the guidance in the ISG Augmenting NUREG-1537 (USNRC, 2012a1996), a postulated fission product release with radiological consequences that exceed those of any accident considered to be credible is analyzed. describes the MHA as a postulated accident scenario whose potential consequences are shown to exceed those of any credible accidents, and that such a scenario need not be entirely credible. SHINE considers such a scenario to be a beyond design basis accident (BDBA).~~

In lieu of identifying a BDBA scenario as the MHA for the SHINE facility, SHINE has chosen to identify a credible fission product-based DBA which bounds the radiological consequences to the public of all credible fission product-based accident scenarios as the MHA for the SHINE facility. The MHA for the SHINE facility is identified as the failure of the target solution vessel (TSV) off-gas system (TOGS) pressure boundary resulting in a release of off-gas into the TOGS cell. A general description of this scenario is provided in Subsection 13a2.1.7.2, Scenario 1. A detailed description of this scenario and an evaluation of the radiological consequences is provided in Subsection 13a2.2.7.

~~The IF MHA is an accident scenario defined to result in the most limiting consequences for the target solution and associated fission products in the IF. Although the MHA is an accident scenario, it does not need to have a credible or defined initiating event or accident progression, except as necessary to evaluate the consequences. The MHA itself is therefore not a DBA; however, it is used as a metric for understanding radiological risk from the facility.~~

~~The main production facility is divided into two major process areas: the IF and the radioisotope production facility (RPF). The IF includes eight irradiation units (IUs) each containing, among other components, a subcritical assembly system (SCAS) (including the target solution vessel [TSV] and TSV dump tank), light water pool system (LWPS), and the TSV off-gas system (TOGS). The TSV, TOGS, TSV dump tank, and associated components make up the primary system boundary (PSB). The RPF consists of several process areas that prepare target solution, extract and purify the radioisotope products, and process waste streams. The major process systems include the uranium receipt and storage system (URSS), target solution preparation system (TSPS), target solution staging system (TSSS), vacuum transfer system (VTS), process~~

vessel vent system (PVVS), radioactive liquid waste storage (RLWS) system, and the radioactive liquid waste immobilization (RLWI) system. The RPF also includes the supercell which is comprised of several internal cells, including the molybdenum extraction areas, purification areas, an iodine and xenon purification and packaging (IXP) cell, PVVS equipment, and packaging areas, that form one hot cell structure.

The ISG Augmenting NUREG-1537, Part 1 (USNRC, 2012a) identifies several possible MHAs that could be considered. The ISG Augmenting NUREG-1537, Part 2 (USNRC, 2012b), Section 13a2.1, indicates that the MHA should release fission products from the uranium target solution (fuel). As such, SHINE has selected the IF MHA based on accidents involving the PSB. Accidents that only involve tritium are not considered for the MHA. SHINE has established the MHA based on the maximum consequence to the public. Worker doses are also calculated for the same accident.

Several potential MHA scenarios were considered, including:

- Energetic dispersal of contents of the PSB with bypass of the light water pool scrubbing capacity;
- Failure of the TOGS pressure boundary and release of some or all of the TSV radioactive gases into the TOGS cell;
- Complete loss of target solution inventory (e.g., TSV break);
- Man-made external event that breaches the PSB of more than one IU, and
- Facility-wide external event that breaches various systems containing radioactive fluids.

The main production facility is designed to withstand external events such as tornado, seismic, and man-made external events. The structure protects the equipment inside its seismic envelope from external events. With this protection, it is not credible for an external event such as an aircraft impact, tornado, flood, earthquake, or tornado missile to initiate an accident involving a safety-related SSC on one or multiple IUs within the structure.

Multiple non-seismic SSCs within the structure can still be affected by and initiate an accident due to seismic events. The neutron drivers are nonsafety related components within the IUs. The neutron drivers do not contain fission products and are not part of the PSB. The failure of multiple neutron drivers during a seismic event is evaluated in Subsection 13a2.1.6.

Events between IUs that could lead to one accident propagating to another unit are not credible because there is no potential interaction between units that supports this propagation. Therefore, scenarios that involve multiple IUs are not analyzed further.

13a2.1.1.1 Identification of Causes, Initial Conditions, and Assumptions

The IF MHA is a failure of the PSB leading to a release of TSV radioactive gases into the TOGS cell. The failure of the TOGS pressure boundary is assumed to cause a failure of the TOGS to function, which initiates the nitrogen purge system (N2PS). The N2PS causes the pressure in the TOGS cell to increase as the nitrogen gas also leaks into the TOGS cell. The MHA assumes that the normal N2PS flow path from the TOGS through the PVVS system is completely blocked, resulting in a higher pressurization of the TOGS cell than would occur for the credible design-basis accident. Therefore, the MHA is a pressurized release from the PSB, which sufficiently bounds credible radiological releases.

~~The selection of this event was determined to be not credible based on a number of factors:~~

- ~~• PSB piping and valves are fabricated and installed according to codes and standards appropriate to their application and safety classification.~~
- ~~• Corrosion allowances on the pressure vessel and piping wall thicknesses ensure that corrosion expected over component lifetime does not impact the pressure retaining capability of the pressure boundary.~~
- ~~• The N2PS flow path from TOGS to the vent release point is designed with redundant valves in parallel to ensure a flow path is available.~~

~~Failure of the TOGS pressure boundary resulting in release of off-gas into the TOGS cell is described in Subsection 13a2.1.7.~~

~~The initial conditions and assumptions used to analyze the IF MHA described above are:~~

- ~~• The material at risk for this scenario is 100 percent of the iodine and noble gases circulating through TOGS at the end of the bounding operating cycle, as described in Section 13a2.2. The TOGS zeolite bed removes a portion of the gaseous iodine during operation.~~
- ~~• Only one IU is affected by the event.~~
- ~~• The integrated primary confinement leakage is within specifications to limit the release of radioactive materials from the cells.~~
- ~~• No credit is taken for the TOGS cell filtration (radiological ventilation zone 1 recirculating subsystem [RVZ1r]), which provides normal cooling and filtration and may continue to function following the accident.~~
- ~~• No credit is taken for the TOGS blowers and zeolite bed once the leak has occurred, which normally function to scrub iodine out of the gases present in the PSB.~~
- ~~• The radiological ventilation zone 1 exhaust subsystem (RVZ1e) redundant ventilation isolation from the IU cell are closed on detection of high radiation in the RVZ1e duct. A sufficient time delay is provided by design to prevent significant radioactive gases from exiting through this path prior to isolation.~~
- ~~• The flow rate of compressed nitrogen from N2PS into the TOGS cell is assumed to be twice the flow rate required for hydrogen mitigation.~~

13a2.1.1.2 ~~General Scenario Description~~

~~A failure in the PSB in the TOGS cell causes radioactive gases to be released from the PSB into the TOGS cell. The release of radioactive material to the IF is mitigated by the primary confinement boundary and by the radiation monitoring of the primary closed loop cooling system (PCLS) expansion tank exhaust, which provide a signal to the TSV reactivity protection system (TRPS) to perform an IU Cell Safety Actuation and isolate the RVZ1e ventilation flow path from the IU cell. The N2PS actuates and causes nitrogen to flow through the affected PSB and out of the break in TOGS into the TOGS cell. The blockage of the PVVS flowpath results in a higher pressurization of the TOGS cell.~~

~~The pressurized flow of nitrogen and radioactive gases from the PSB to the IF and the environment results in a long term release.~~

~~The main facility ventilation system (i.e., radiological ventilation zone 2 [RVZ2]) is isolated within 30 seconds of detectable accident conditions. However, it is assumed that the release is a-~~

~~ground release through unfiltered leakage pathways through the facility leak paths (e.g., door gaps). An unfiltered ground release is conservative.~~

~~Detailed analysis of the IF MHA is provided in Subsection 13a2.2.1.~~

13a2.1.1.3 ~~Accident Consequences~~

~~The accident consequences associated with the IF MHA are evaluated further in Subsection 13a2.2.1.~~

13a2.1.2 INSERTION OF EXCESS REACTIVITY

The excess reactivity insertion event during normal operations is identified as a potential initiating event for accidents in the accident analysis. The potential for excess reactivity insertions during the startup process and irradiation mode of the TSV was identified as scenarios to be evaluated.

Two operating modes that have potential reactivity impacts were evaluated for the TSV:

- Mode 1 - Startup Mode: filling the TSV
- Mode 2 - Irradiation Mode: operating mode (neutron driver active)

Excess reactivity insertion events can challenge the integrity of the primary system boundary (PSB) by causing increased power density, temperature, and pressure.

The subcritical assembly system (SCAS) is designed to operate in a subcritical state without available excess reactivity. Reactors normally have engineered reactivity control mechanisms and load excess reactivity into the core to accommodate power defect, fuel burnup, and uncertainty in k_{eff} . There are no reactivity control systems in the SHINE system. Analyzing the inadvertent withdrawal of the most reactive control element as performed for reactors is not possible. SHINE will not perform experiments with the IUs, so there are no reactivity effects from experiment malfunctions.

For the subcritical assembly being driven by the neutron driver (such as in Mode 2), excess reactivity insertion (i.e., reactivity inserted beyond planned operations) has similar effects to excess reactivity insertions in a reactor, including increases in power, temperature, and gas generation. As substantial power can be generated even if reactivity remains subcritical in a driven system, the effects of excess insertions of reactivity were considered in the safety analysis.

For the subcritical assembly, when it is not being driven by the neutron driver (such as in Mode 1 or Mode 2 during Driver Dropout), excess reactivity insertion could lead to inadvertent criticality and unplanned fission power generation, temperature increase, and gas generation.

The assembly is designed to be in a subcritical condition during each mode of operation, with multiple safety controls to prevent or mitigate an excess reactivity insertion or inadvertent criticality. The potential for an inadvertent criticality is greater during fill operations. However, as discussed in the following subsections, controls are in place to safely limit excess reactivity insertions.

Inadvertent criticality events outside the IF (i.e., in the [radioisotope production facility \[RPF\]](#)) are prevented by the nuclear criticality safety program, as described in [Section 6b.3](#).

13a2.1.2.1 Identification of Causes, Initial Conditions, and Assumptions

The following postulated initiating events and scenarios that could lead to an excess reactivity insertion or power transient during operation were identified using the guidance in the ISG Augmenting NUREG-1537 (USNRC, 2012a):

- Increase in the target solution density during operations (e.g., due to pressurization)
- Target solution temperature reduction during fill/startup (e.g., excessive cooldown)
- Target solution temperature reduction during irradiation (e.g., excessive cooldown)
- High reactivity and power due to high neutron production at cold conditions
- Moderator addition due to cooling system malfunction (e.g., cooling water in-leakage)
- Additional target solution injection during fill/startup and irradiation operations
- Realistic, adverse geometry changes
- Reactivity insertion due to moderator lumping effects (e.g., voiding in the cooling system)
- Inadvertent introduction of other materials into the TSV (e.g., uranium solids introduction or precipitation of uranium from target solution)
- Concentration changes of the TSV target solution (e.g., through boiling or evaporation)
- Failure to control temperature during 1/M measurements at startup

The following initial conditions or assumptions are made with respect to the Mode 1 and Mode 2 operations:

- TSV is filled to an approximate k_{eff} of [$\beta^{\text{PROP/ECI}}$] at a cold startup temperature range of 59°F to 77°F (15°C to 25°C).
- The TSV is operated in a subcritical state with a nominal k_{eff} of approximately [$\beta^{\text{PROP/ECI}}$] during steady-state irradiation operations. The TSV is designed to operate with the neutron driver in service with a source strength yielding a maximum value of 125 kilowatts (kW) power within the target solution.
- During irradiation, the TSV is designed to operate with a maximum average temperature below 176°F (80°C).
- The target solution has high negative temperature and void coefficients, as described in [Section 4a2.6](#)).
- The TRPS is designed to dump the TSV on high neutron flux level (source, wide range, and time-averaged) to protect the PSB.
- Redundant, fail-open TSV dump valves ensure target solution can be dumped and are cycled each irradiation cycle.
- The TRPS is designed to dump the TSV on high PCLS temperature, low PCLS temperature, and low PCLS flow.

13a2.1.2.2 General Scenario Descriptions

The general scenarios for each of the potential excess reactivity insertion events listed in [Subsection 13a2.1.2.1](#) are discussed in detail below.

Scenario 1 – Increase in the Target Solution Density During Operations

During irradiation operations, the limiting target solution cooldown scenario occurs when the TSV is operating normally at licensed power of 125 kW and then PCLS temperature instantaneously decreases from 25°C to 15°C. Given the thermal mass of the PCLS, the instantaneous change is a conservative approximation. The thermal mass of the TSV and target solution is slow to respond, allowing sufficient time for the TRPS IU Cell Safety Actuation on high time-averaged neutron flux at 104 percent of licensed power. This drains the target solution to the TSV dump tank, terminating the event.

Greater PCLS temperature changes are prevented by the TRPS IU Cell Safety Actuation on high and low PCLS temperature, resulting in a dump of the target solution and termination of the neutron generation by the neutron driver assembly system (NDAS). The draining of the target solution to the TSV dump tank results in safe shutdown of the target solution. No damage to the PSB occurs and there are no radiological consequences.

Scenario 4 – High Power Due to High Neutron Production and High Reactivity at Cold Conditions

A high reactivity and power event can occur due to excess tritium injection into the NDAS during cold conditions. This can occur as a result of a tritium purification system (TPS) control system or component failure during startup that injects excess tritium before the TSV is at operating temperature. The TRPS initiates an IU shutdown on high wide range neutron flux.

A high reactivity and power event can also occur if the NDAS neutron production drops to a lower flux than expected due to focusing issues, electrical arcing, or other malfunctions. This loss of neutron source during irradiation results in a decrease in void fraction and a target solution cooldown in the TSV. If the NDAS neutron production were to rapidly return to full output subsequent to a loss of void fraction and cooldown, excessive power generation could occur that could challenge target solution power density limits or PSB integrity.

To prevent excessive power pulses at the start of the irradiation cycle, ~~TPS permissives prevent transitioning from Mode 1 to Mode 2 until the~~ [

] ^{PROP/ECI}. This prevents the driver from producing excessive neutrons concurrent with high system reactivity.

To prevent excessive power pulses during driver ramp-up as the target solution has not yet reached operating temperature, the rate of tritium concentration increase in the NDAS target chamber is limited by the achievable flow rate of tritium from the TPS. This design characteristic is passive and designed to prevent a TPS failure that could result in rapid tritium concentration increase in the target chamber.

To prevent excessive power pulses during irradiation, the TRPS de-energizes the NDAS high voltage power supply (HVPS) redundant breakers on a driver dropout signal after [^{PROP/ECI} of low power range neutron flux in Mode 2 are detected. This prevents the driver from producing excessive neutrons concurrent with high system reactivity.

As described in [Subsection 4a2.6.3](#), the cooldown and void loss during this event creates a reactivity insertion of up to [^{PROP/ECI} from loss of void and up to [^{PROP/ECI} in [^{PROP/ECI} from cooldown. The final k_{eff} of the system remains below the initial startup k_{eff} of the system. It is assumed the driver instantaneously returns to full output. The

Regarding the target solution itself, uranium solids used in the target solution preparation process are prevented from reaching the TSV by a filter in the [target solution preparation system \(TSPS\)](#) process.

Water could potentially be introduced into the TSV through a leak from the PCLS, light water pool, or from the RPCS-cooled components in TOGS. Dilution of the target solution in the TSV is discussed in [Subsection 13a2.1.2.2](#), Scenario 5.

Material Entering the TSV from Sources Internal to the PSB

Two potential sources of uranium solids entering the TSV and resulting in reactivity addition were evaluated: uranyl salt crystal buildup in the TSV or TOGS components and precipitation of uranium solids.

The first two postulated scenarios are a buildup of uranium-bearing salt crystals in the TSV (such as a "bathtub" ring) or in TOGS components. These salt crystals could become rewetted or otherwise dislodged and reenter the TSV. The buildup of salt crystals in the TSV is not expected due to the high humidity of the TSV and the cold walls of the TSV. In addition, periodic inspection of the TSV is performed which would allow for detection of salt crystal buildup.

If salt crystals did accumulate, their release could lead to an unexpected reactivity increase due to the increase in fissile material in the target solution. To quantify reactivity effects, it is postulated that a piece of deposited salt containing 100 grams of uranium is dislodged from the upper TSV surfaces and falls into the target solution. The re-dissolution of the salt adds approximately []^{PROP/ECI} of reactivity to the system. This reactivity effect does not result in significant consequences and does not lead to an inadvertent criticality. If additional salt pieces were to continue to enter the TSV, they could continue to re-dissolve and lead to further concentration increases, and power could increase in the TSV. The TRPS would dump the target solution on high time-averaged neutron flux, terminating any reactivity increase. The TSV dump tank is favorable geometry at any uranium concentration. No damage to the PSB occurs and there are no radiological consequences.

The second postulated scenario is precipitation of uranium solids from the solution. Precipitation of uranium solids due to uranyl peroxide formation is possible in aqueous reactors. In the SHINE system, chemistry, power density, and temperature limits have been placed on the target solution as described in [Subsection 4a2.6.3](#). Given these limits, no significant precipitation is expected. For transient events, precipitation has not been seen in transient operations of historic uranyl sulfate systems. Therefore, the dump of the target solution by TRPS on high time-averaged or wide range neutron flux occurs prior to significant precipitation developing in the target solution.

The accumulation of small amounts of precipitation over many cycles has been considered. This could lead to chemical effects on the TSV surface, which may have the potential to lead to a failure of the PSB. A failure of the PSB is analyzed in [Subsection 13a2.1.4](#).

Scenario 10 – Concentration of the TSV Target Solution

Postulated scenarios where the uranium concentration of the target solution could increase were evaluated. One identified scenario requiring control was the TOGS pressure control failure

Additional defense-in-depth measures are also in place to avoid a leak and detect leaks, which include:

- control of solution pH through target solution sampling in the target solution hold tank;
- chemistry controls of PCLS to limit corrosion (see [Section 5a2.5](#)); and
- conductivity instrumentation in PCLS, which detects intrusion of target solution.

The small amount of target solution that could diffuse into the PCLS cooling water after the pressure between the PCLS and the PSB is equalized, combined with the dilution of the leaked material by the cooling water, minimizes the potential for criticality in the PCLS and dose to workers or the public.

Because of the system characteristics and preventative controls in place, further analysis is not required.

Scenario 5 – Failure in the TOGS Causes High Pressure in the TSV during Fill Mode

A failure by the TOGS to control pressure, and a resulting pressure increase during TSV filling operations, may result in a backflow of target solution. Target solution may flow through the fill line into the TSV fill lift tank, into the [vacuum transfer system \(VTS\)](#) header, and into the VTS buffer tank. This failure potentially results in radiological exposures to workers or a criticality accident in non-favorable-geometry components in the VTS.

The protection in place for this scenario is the configuration of the TSV fill line to prevent significant volume of target solution from backflowing from the TSV into the VTS lift tank. The TSV fill line connects to the TSV with an air gap. The connection is located at the approximate elevation of the TSV overflow lines. The fill line is sloped to allow it to drain after fill operations have occurred. Therefore, no significant volume of target solution will backflow from the TSV to the VTS lift tank in the event of pressurization of the TSV.

Defense-in-depth measures are also present to mitigate this scenario, which include:

- the VTS vacuum valve to lift tank closes from high liquid level in the lift tank, and
- a drain valve for the buffer tank opens and drains to [radioactive liquid waste system \(RLWS\)](#) if a high level in the buffer tank is detected.

Because of the system characteristics and preventative controls in place, further analysis is not required.

Scenario 6 – Target Solution Leakage within a Valve Pit

A pipe or valve failure in the valve pit may be caused by overpressurization due to thermal expansion of target solution in an isolated section of piping. This pipe or valve failure results in leakage of target solution from the system into the valve pit, which subsequently could result in: (1) increased worker or public dose, or (2) a criticality accident in the valve pit. The protections in place to mitigate the consequences of target solution leakage within a valve pit are: (1) drip pans and drains to the radioactive drain system (RDS), which prevent accumulation of solution within the valve pit and prevent criticality, and (2) valve pit shielding and confinement for fission products that could result from leakage, reducing potential dose to workers and the public.

- The N2PS begins passively injecting nitrogen gas into the primary system boundary. Nitrogen gas is injected in the eight SCAS systems via a connection to the dump tank. The gas purges the primary system boundary leaving through a vent connection from the TOGS to the [process vessel vent system \(PVVS\)](#) header. The gas then passes through the PVVS carbon delay beds for removal of fission product gases before release to the environment. The nitrogen purge system has enough capacity for three days, after which the system is resupplied.

In addition to the above sequence of events in the IU, the following actions also occur simultaneously:

- In the event that any transfer of uranyl sulfate solution is in progress, VTS transfer operations stop.
- Nitrogen gas sweeps RPF process tank and lift tank headspaces to dilute radiolytic hydrogen. Nitrogen from the N2PS is routed to the PVVS carbon beds for removal of fission product gases before release to the environment. The N2PS has enough capacity for three days, after which the system is resupplied.
- The UPSS supplies essential facility loads their required runtime as provided in [Table 8a2.2-1](#). The 120 VAC UPSS buses automatically maintain power to essential instrumentation and equipment, including radiation monitoring systems.

13a2.1.5.3 Accident Consequences

The accident consequences associated with a LOOP are discussed further in [Subsection 13a2.2.5](#).

13a2.1.6 EXTERNAL EVENTS

This section discusses external events that impact the IF. This class of accident initiators represent natural or man-made events that occur outside the facility and have the potential to impact facility SSCs. Scenario descriptions are provided in this section for the range of accident initiators that were considered during the accident analysis.

13a2.1.6.1 Identification of Scenarios, Initial Conditions, and Assumptions

The following potential external events were evaluated:

- Seismic event affecting the IF and RPF (see [Section 3.4](#)).
- Severe weather events affecting the IF and RPF (see [Section 3.2](#)).
- Transportation accidents, including small aircraft crash into the IF or RPF (see [Subsection 3.4.5](#)), toxic gas releases (see [Subsection 2.2.3](#)), or explosions (see [Subsection 2.2.3](#)).
- External flooding affecting the IF and RPF ([Subsection 2.4.2](#)).
- External fires from natural sources (see [Subsection 2.2.3](#)).

The initial conditions and assumptions associated with these external events include:

- Prior to an external event occurring, the facility is assumed to be running at nominal conditions.
- Unless otherwise noted, these scenarios only consider single failure mechanisms.

Scenario 7 – Seismic Event Causing PVVS/VTs Failure

A seismic event may cause the failure of the PVVS/VTs. The limiting postulated failure occurs during target solution transfer from the TSV dump tank to the molybdenum extraction and purification system (MEPS). The PVVS is assumed to fail, resulting in a loss of sweep gas in the vacuum transfer tanks. Due to the lack of circulation, the hydrogen concentration in the vacuum transfer tanks increases, approaching the deflagration limit. In this event, the N2PS dilutes the hydrogen gas concentration and prevents hydrogen deflagration.

In addition, the loss of PVVS also results in the loss of the VTs, stopping the movement of target solution. The target solution remains in the lift tanks or drains back into the TSV dump tank. The lift tanks and the TSV dump tanks are passively-cooled and geometrically-favorable tanks. Therefore, there are no consequences resulting from this event.

The radiological consequences of deflagrations within the primary system boundary are discussed in [Subsection 13a2.1.9](#).

Scenario 8 – Seismic Event Causing Crane Failure

A seismic event may cause the failure of the IF crane. A failure of a crane during a heavy lift of a vault plug or neutron driver in the IF could result in the heavy load dropping onto the NDAS or SCAS components. Potential consequences of the crane failing include radiological dose.

To prevent crane failure, the crane is a single failure proof design and has been seismically qualified. Additional information on heavy load drops is provided in [Subsection 13a2.1.12](#).

Scenario 9 – Seismic Event Causing Chemical Spill

A seismic event may cause uranium oxide powder to become airborne during target solution preparation activities or may overturn a uranium storage rack causing multiple canisters to spill, resulting in a worker uptake of uranium oxide.

Oxide handling operations occur within the TSPS and [uranium receipt and storage system \(URSS\)](#) gloveboxes, which are seismically qualified and have installed filtered ventilation. The quantity of uranium used in handling operations is limited and is insufficient to cause chemical dose consequences that exceed the chemical exposure criteria in the event of a single canister spill.

Discussion of the consequences of an overturned uranium storage rack and additional discussion of accidents with chemical dose consequences is provided in [Section 13b.3](#).

Severe Weather Events Affecting the IF and RPF

Scenario 10 – Tornado or High Winds Affecting the IF and RPF

The main production facility is designed to withstand credible wind and tornado loads, including missiles, as described in [Section 3.2](#) and [Subsection 3.4.2.6](#), respectively.

A tornado or high wind event may cause an N2PS tube failure. Potential consequences of a N2PS failure include damage to the components containing radioactive materials.

that are discussed in this section are those that may result from failures in support systems or other shared systems that could result in an adverse impact on the primary system boundary.

PVVS is connected to the eight IUs via connections to TOGS. Accidents with PVVS failure are considered in [Section 13b.2](#).

The functional interactions considered in this analysis are the following:

Loss of Off-Site Power

The NPSS provides electrical power to SSCs in the IF and the RPF.

Reduction of cooling

- The RPCS is the common heat sink for the independent instances of PCLS, which are the primary cooling systems for each TSV. Each PCLS removes generated heat from its associated TSV during normal and shutdown operations. The generated heat is transferred to the RPCS via the PCLS heat exchangers. The RPCS is served by the PCHS, which exhausts heat to the environment.
- RPCS additionally provides cooling for several heat exchangers in the IF and the RPF, including:
 - TOGS condenser-demisters
 - TOGS recombiner condensers
 - TSPS dissolution tank reflux condensers
 - [}PROP/ECI
 - PVVS condensers
 - NDAS cooling cabinets
 - RVZ1r
 - Radiological ventilation zone 2 recirculating subsystem (RVZ2r)

Loss of ventilation

- The ventilation systems (RVZ1, RVZ2) are described in [Section 9a2.1](#).
- Loss of RVZ1 flow may result in maloperation of multiple systems in the IF and RPF, such as the:
 - TPS glovebox pressure control exhaust and the vacuum/impurity treatment subsystem (VAC/ITS) process vents
 - [Radioactive liquid waste immobilization \(RLWI\) system](#) shielded enclosure,
 - Individual cells of the supercell,
 - URSS glovebox,
 - TSPS gloveboxes, or
 - Vent exhausts from the PCLS expansion tanks.
- Loss of RVZ2 to common areas of the IF and the RPF.
- Loss of ventilation to the primary cooling rooms.

Spatial Interactions

Spatial interactions are interactions resulting from the presence of two or more systems in locations. Spatial interactions include a single event that could impact the operation of the adjacent systems, or the failure of one system that may impact the operation of another system.

Scenario 1 – Loss of Normal Ventilation to the IU or TOGS Cells

A failure of RVZ1 may be caused by failure of a blower or cooler, including loss of cooling water. It may also be caused by a failed-shut or mispositioned damper or other equipment failure. A loss of cooling may cause instrumentation inaccuracies or failures which may lead to TOGS maloperation or loss of function. This can result in a potential deflagration and release of radiological material.

The protections in place to prevent a TOGS failure due to loss of ventilation are redundant and environmentally qualified TOGS instrumentation (e.g., low flow) that initiates a TRPS signal if TOGS failures are detected. The TRPS signal opens redundant TSV dump valves draining target solution to the TSV dump tank and shuts down the irradiation unit. Decay heat from the target solution is removed by the light water pool.

Scenario 2 – Loss of Normal Ventilation to Primary Cooling Rooms

A failure of RVZ2 may be caused by failure of a blower or cooler, including loss of cooling water. Loss of ventilation to individual primary cooling rooms may also be caused by a failed-shut or mispositioned damper. A failure of normal ventilation may lead to increased environmental temperatures within the primary cooling room with potential for increased instrument inaccuracies or failure. The consequences of an RVZ2 failure leading to equipment malfunction result in TSV overcooling causing a reactivity insertion in the TSV. Excess reactivity additions are discussed further in [Subsection 13a2.1.2](#).

The protections in place to prevent TSV malfunctions related to ventilation failures are redundant low and high PCLS temperature trip that initiates a TRPS signal (separate from the control system). The TRPS signal opens redundant TSV dump valves draining target solution to the TSV dump tank and shuts down the irradiation unit. Decay heat from the target solution is removed by the [light water pool system \(LWPS\)](#).

Based on the preventive controls the failure of normal ventilation does not have radiological consequences, and no further analysis is required.

Loss of ventilation due to a LOOP is described in [Subsection 13a2.1.5](#).

Loss of ventilation due to external events is described in [Subsection 13a2.1.6](#).

Spatial Interactions

Fires

The fire hazards analysis (FHA) evaluates the fire hazards and fire protection features for each fire area in the SHINE facility. The fire protection features in the IF rely on low combustible loading, fire detection, manual fire-fighting capabilities, and rated fire barriers to limit the potential for fire initiation and spread within the IF. The fire protection program and the FHA are described in [Section 9a2.3](#).

Potential fire scenarios in the IF have been evaluated. The principle fire hazards in the IF are: (1) the HVPS used for the NDAS service cell, (2) hydrogen located in the TPS and within the PSB for each IU cell, and (3) the carbon filters in the radiologically controlled area (RCA) exhaust

described in [Subsection 9b.7.2](#). Therefore, a heavy load drop into an in-service IU cell is not credible.

Heavy Load Drop Scenario 3 – Heavy Load Drop onto TPS Equipment

A crane mechanical failure or operator error during a lift may result in a heavy load drop onto TPS equipment. The heavy load can damage the equipment and result in a release of radioactive material.

SHINE has applied the applicable guidance from NUREG-0612, Control of Heavy Loads at Nuclear Power Plants (USNRC, 1980), for control of heavy loads at the SHINE facility, as described in [Subsection 9b.7.2](#). Therefore, a heavy load drop onto TPS equipment is not credible.

13a2.1.12.3 Accident Consequences

Neutron Driver Assembly System

The dose consequences of an NDAS failure are evaluated in [Section 13a2.2.12](#).

Tritium Purification System

The dose consequences of a release of tritium from TPS Scenario 1 are described in [Section 13a2.2.12](#). This scenario bounds the dose consequences for the release of tritium from TPS Scenario 3 and TPS Scenario 4.

TPS Scenario [2 and TPS Scenario 5](#) ~~is~~[are](#) not credible; therefore, accident consequences are not evaluated. |

Heavy Load Drop

Heavy load drop scenarios are not credible; therefore, accident consequences are not evaluated.

13a2.2 ACCIDENT ANALYSIS AND DETERMINATION OF CONSEQUENCES

This section describes the accident analysis for the limiting scenarios described in [Section 13a2.1](#) and provides a determination of the radiological consequences. Chemical consequences are analyzed in [Section 13b.3](#).

Radiological consequences are determined for members of the public and workers (i.e., control room operators) and are provided in Table 13a3-1 and Table 13b.2-2. Radiological consequences to control room operators are determined for the duration of a postulated event, accounting for shift change outs, and demonstrate that SHINE Design Criterion 6 (Control room) is met.

The analyses in this section evaluate the applicable radiological consequences of these accidents to demonstrate that the SHINE accident dose criteria are met. The SHINE accident dose criteria are defined as follows:

- Radiological consequences to an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not exceed ~~500-~~ ~~m~~1 rem total effective dose equivalent (TEDE) for the duration of the accident, and
- Radiological consequences to workers do not exceed 5 rem TEDE during the accident.

Radiological Consequence Assessment Development

The radiological consequence assessment is a multi-step process. [Figure 13a2.2-1](#) provides a graphical representation of the process, which is further described in this section. The process involves: (1) calculation of radionuclide inventories, (2) definition of the accident-specific materials-at-risk (MAR), (3) transport methods of radionuclides, (4) development of accident source terms, and (5) determination of radiological consequences.

Radionuclide Inventories

For most accident scenarios, the MAR were derived from the target solution vessel (TSV) target solution inventory at the end of []^{PROP/ECI} of continuous 30-day irradiation cycles with a []^{PROP/ECI} downtime between cycles. The constant power level used for the analysis was 137.5 kW, which is 110 percent of design operating power. The TSV inventory calculation includes effects from fission, transmutation, activation, and decay. The calculation contains time steps from the start of irradiation through the end of the approximately []^{PROP/ECI} irradiation cycle and additional time steps that account for decay post-shutdown, as needed. []^{PROP/ECI} was selected for the irradiation cycle based on the anticipated replacement period for target solution.

Accident-Specific Materials-At-Risk Partitioning

For accident scenarios involving the release of radionuclides produced in the target solution, a portion of the inventory was released based on various factors unique to each scenario. The starting inventory was selected based on the assumed start time for each scenario and was then partitioned based on scenario specific nuclide removal mechanisms. For the source term determination and the determination of resulting dose, the radionuclides are grouped into three groups: iodine, noble gases, and non-volatiles. The non-volatile group encompasses the radionuclides which do not fall into the other groups.

For scenarios involving the release of tritium, the available MAR was determined based on the limiting operational values for the affected systems or components.

Radionuclide Transport Methods

The transport of radioactive material for the accident analysis was quantitatively evaluated using a five-step process. The result of the five-step method is a combination of the airborne release fraction (ARF) and the leak path factor (LPF) into a single parameter (i.e., the LPF model) which was then used to replace these two terms in the typical five-factor formula found in NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook (USNRC, 1998).

Step One: Identify control volumes and leakage paths

This step identifies the control volumes, leakage paths, and heat sinks that define the model geometry. The performance of this step establishes the geometric information for the LPF calculations including gas and liquid volumes, surface areas for deposition, flow path areas and characteristics that influence leakage, elevations, surface areas for liquid and air heat sinks, and system information for ventilation rates and filtering. The control volumes used in the analysis include the source volume, the building volume, and the environment. Leakage paths are treated as junctions between the control volumes.

Step Two: Quantify scenario progression source histories

This step defines the specific physical phenomena for each scenario. This includes the accident initiator and source volume location to quantify the amount or rate at which materials and energy can be released. Where pertinent, this also includes the amounts or rates at which gases and aerosols are evolved. It also includes the initial conditions including gas temperature and pressure, liquid mass and temperature as appropriate, and initial released activity. Further rates of activity release are also specified for evolution of iodine and non-volatile evolution. The performance of this step quantifies initial sources and source rate histories for radionuclides and other mass and energy sources that drive material transport.

- Iodine partitioning was calculated using equations based on the definitions found in NUREG/CR-5950, Iodine Evolution and pH Control (USNRC, 1992), without modifications.
- Bursting bubble aerosols are treated using a linear relationship between entrainment coefficients and the volumetric flow rate with entrainment coefficients consistent with the small amount of dissolved materials present in the target solution.
- Spray and free fall aerosols use a constant airborne release fraction found in NUREG/CR-6410 (USNRC, 1998).
- Radiolysis was treated as either an instantaneous release of hydrogen or as entrainment with source generation based on decay power inputs for long-term problems.

Step Three: Quantify leakage rates between volumes

In this step, leakage rates between volumes are defined in either closed-form expressions for input into the LPF model or in terms of time histories generated by the LPF model. Leakage rates are driven by pressure, gas density differences, and barometric breathing.

The gas pressure in each volume was determined using a combination of conservation laws for mass and energy, temperature-dependent specific heats, and the ideal gas law. In addition to intra-volume flows, the following heat and mass transfer rates are considered:

- Natural convection heat transfer between the gas space and the wall using a constant or calculated wall temperature and temperature profile and standard correlations accounting for laminar and turbulent conditions
- Natural convection heat transfer between a liquid pool and the wall or floor using the heat sink temperature and standard correlations accounting for laminar and turbulent conditions
- Water pool evaporation using standard mass transfer correlations accounting for laminar and turbulent flow
- Heat sources to the gas and water pool to account for equipment
- Decay power in the gas and liquid based on the local activity

Pressure-driven flow through each junction was calculated using the standard compressible flow equation for pressure-driven flow. Pressure-driven flow through shielding cover plug gaps was calculated using the equation for plane Poiseuille flow. In the absence of pressure-driven flow, counter-current flow was assumed to be induced and was calculated based on the difference in pressure from one side of the cover plug to the other.

Step Four: Quantify removal mechanisms

This step establishes expressions for removal rates of radionuclides for use in the LRF model. Of particular interest are iodine adsorption, non-volatile aerosol deposition, and removal by filters along a flow path. Radioactive decay and build-in are not considered in this step.

- Removal by aerosol settling was evaluated using Stokes' law and the calculated equilibrium particle size for a hygroscopic particle. The use of the equilibrium particle size is valid based on the time scale of the accident sequences.
- Consideration of removal by filters in the accident flow paths was done by assigning decontamination factors for noble gases, iodine, and aerosols to each filter.
- Removal due to barometric breathing was based on an analysis of the peak-to-peak magnitude of environmental pressure fluctuations. Eight years of pressure data were collected and analyzed to provide a bounding estimate of the barometric breathing flow rate for the facility.

Iodine removal through absorption was determined using models developed for Library of Iodine Reactions in Containment (LIRIC) and Iodine Model for Containment Codes (IMOD) codes written and validated by Atomic Energy of Canada Limited (AECL). Typical values for adsorption and desorption rate constants determined from experimental data were used in the analyses. Experimentally determined saturation capacities were also used. Desorption was neglected because it has been shown to have little impact on equilibrium iodine concentrations since the rate constants for desorption are more than a factor of 1,000 lower than adsorption. The deposition velocity was estimated from correlations which serve as the basis for the models used in NUREG-75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (USNRC, 1975) (WASH-1400). Similarly, the diffusion coefficients were found using standard methods which form the basis for the values in WASH-1400.

Step Five: Generate LPF model

~~The final LPF model consists of a coupled set of differential equations that express the masses or mass fractions of radionuclides in each region as a function of time, including airborne and removed quantities.~~

~~The resulting LPF model 30-day cumulative leakage factors and the 10-minute leakage factors are provided in Table 13a2.2-1 and Table 13a2.2-2, respectively. These values represent the product of the leak path factors and the airborne release fractions for each group of radioisotopes.~~

control volume and tracer method.

The control volume method considers each part of the facility as a fixed volume that the material is free to disperse into. Dispersion within these volumes is assumed to be instantaneous. Each volume is connected by one or more junctions which allows flow in one direction at a volumetric flow rate, either pressure-driven or constant. Counter-current flow, or flow back into the previous control volume, is conservatively neglected. Flow from the irradiation facility (IF) to the radioisotope production facility (RPF) or vice versa is not modeled; material present in the IF or RPF control volumes is assumed to exit the SHINE facility to the environment without further dilution.

The tracer method is a modeling tool that is representative of the kind of material being tracked. Because an output of the tracer method is a fraction of material released, any material can be used as a tracer for any other kind of material as long as the tracer's properties (density, molar mass, etc.) are used consistently throughout the scenario. For example, a gas may be used as a tracer for an airborne non-volatile because the output of the analysis is normalized to the amount of material initially released. Therefore, the physical properties of the tracer itself are not important as long as they are applied consistently. In this calculation, iodine is used as a tracer for iodine, krypton for noble gases, xenon for non-volatiles, nitrogen or air for nitrogen, air for air, and tritium for tritium.

For each control volume, the amount and volume fraction of each tracer, in moles, is calculated using the density and molar weight of each tracer and the volume of the space. Material flowing out of the control volume is calculated as the product of the volumetric flow rate multiplied by the volume fraction of the tracer. Flow can be due to pressure-driven flow, barometric breathing, or a constant flow rate based on the design of the cell or glovebox.

Flow Between Control Volumes

There are three types of flow between control volumes. The first is pressure-driven Poiseuille flow, which is calculated using the following equation:

$$v_{flow}(t) = \frac{p(t)}{k}$$

Where:

- v_{flow} is the volumetric flow rate at a given time (m^3/s)
- $p(t)$ is the pressure difference between the control volumes (Pa)

- k is the conversion factor from pressure to volumetric flow. This represents the tightness of the seal on the cell being pressurized ($\text{Pa}\cdot\text{s}/\text{m}^3$).

This kind of flow is used in scenarios that model the pressurization of cells due to the nitrogen purge system (N2PS) actuation.

The second kind of flow is a prescribed leak rate. In lieu of modeling the pressure of the system, these scenarios consider the cell as flowing at its maximum design leak rate for the duration of the design basis accident (DBA). These prescribed leak rates are some fraction of the volume per hour, converted into a fraction of volume per second and multiplied by the volume of the cell to convert to m^3/s .

The third kind of flow is due to barometric breathing, which is the gas flow driven by cyclical changes in the atmospheric pressure. Barometric breathing is determined using meteorological data from the Southern Wisconsin Regional Airport. The barometric breathing rate is converted to a volume fraction per second which is multiplied by the volume of the cell to produce a volumetric flow rate in m^3/s . This kind of flow is considered for all cells that are not pressurized and do not have a designed leak rate.

In some cases, more than one flow type is modeled. For example, a transient may use a combination of pressure-driven flow and barometric breathing. This is due to the system initially being dominated by pressure-driven flow due to a gas release, but eventually achieves a pressure equilibrium between control volumes or between a control volume and the environment.

Deposition of Iodine and Non-Volatiles

In DBA scenarios involving uranyl sulfate and leakage through the radiologically controlled area (RCA), both iodine and non-volatile deposition surfaces are considered. Where possible, iodine and non-volatile absorption coefficients are calculated using the following equation:

$$\lambda = v_d \frac{A}{V}$$

Where:

- λ is the absorption coefficient (s^{-1})
- v_d is the settling velocity (m/s)
- A is the surface area that the radionuclide cloud encounters (m^2)
- V is the volume of the gas (m^3)

For the IF and RPF, the area that the gas encounters is only considered to be the floor area of the IF or RPF. Because this eliminates the surface area of the walls and ceiling of the RCA, this is a conservative assumption. The free volume of the IF and RPF is used as the volume of the gas.

The settling velocity is assumed to be 10^{-4} m/s, consistent with the 'dry conditions' velocity for epoxy paint of 10^{-3} m/s. Desorption of the iodine back into the RCA or cell gas space is not considered in this analysis.

No iodine or non-volatile adsorption is modeled once the radionuclides exit the RCA.

Once the absorption coefficient is determined, the removal of isotopes is calculated using the following equation:

$$\frac{da}{dt} = -\lambda a(t)$$

Where $a(t)$ is the moles of the corresponding tracer.

Receptor Activity Fractions

A receptor activity fraction (RAF) represents the fraction of a tracer that is present in a control volume at a specific time interval. The RAF for a control room operator at time bin j is therefore:

$$RAF_{CR,j} = \frac{IA_j O_j}{a_{initial}}$$

Where IA_j is the integrated activity in the control room at time bin j , O_j is an occupancy factor for operators in the control room, and $a_{initial}$ is the initial tracer moles released. The integrated activity is calculated for each time bin j using one second time steps as:

$$IA_{total}(t_j) = \sum_{t=0}^{t_j} a(t)$$

The integrated moles for a given time bin j , defined by the initial time j_1 and concluding time j_2 , is then calculated by the following equation:

$$IA_j = IA_{total}(j_2) - IA_{total}(j_1)$$

The total receptor activity (RA) is then calculated by summing the products of IA_j and O_j from the beginning of the DBA to the end of the desired time period, dividing that value by the initial moles released for that tracer, and multiplying the resultant RAF by the activities included in the scenario's MAR.

The public RAF is calculated in a similar manner to the control room RAF, with the dispersion factor $(\chi/Q)_j$ replaces the control room occupancy factor.

$$RAF_{p,j} = \frac{IA_j \left(\frac{\chi}{Q} \right)}{a_{initial}}$$

The RAF_p for a given time period is calculated by summing the calculated public RAFs from the beginning of the DBA to the end of the desired time period and multiplying the summed RAFs by the activities in the scenario's MAR.

For determination of RAF to the public, 95th percentile site boundary time-dependent χ/Q values are used. For the determination of RAF to the worker, 95th percentile control room time-dependent χ/Q values are used. The maximum calculated value over all directions of the 95th percentile χ/Q was used for both receptor locations. The use of time-dependent χ/Q values is consistent with the methodology presented in Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (USNRC, 2003a). The environmental and meteorological conditions used to develop the atmospheric dispersion factors are discussed in Section 2.3.

Accident Source Terms

The accident source terms for each accident scenario ~~was then determined by the use of~~ are consistent with the five-factor formula methodology as described in Section 3.2.5.2 of NUREG/CR-6410 (USNRC, 1998). ~~The LPF model described above replaces~~ However, the combined ARF x LPFRAF term in the formula as it combines both of these parameters described in the previous section accounts for the leak path factor, airborne release fraction, and atmospheric dispersion factors. The cumulative ~~ARF x LPFRAF~~ values represent the time-dependent leakage of radionuclides. The ~~ARF x LPFRAF~~ values are calculated for the leakage from the source volume to the building for the worker dose (~~10 minutes~~ duration of the event), and the source volume to the environment for the public dose (duration of the event).

~~The respirable fraction (RF) and damage ratio (DR) are conservatively set at 1.0. Radioactive decay of the MAR for various times after the initiation of an accident was included in the source-term development. The MAR inventory was tabulated based on various decay times. Determination of the accident source term was accomplished by linear interpolation between the tabulated values, which is a conservative estimate.~~

~~The atmospheric dispersion factor (χ/Q) was also applied for releases to the environment as discussed below.~~

Radiological Consequences

The radiological consequences for each accident are presented in terms of TEDE.

The methodology uses external and internal radiation sources to calculate the effective external dose equivalent and dose equivalent for external sources and committed effective dose equivalent and committed dose equivalent for internal sources. The TEDE and the total dose equivalent (TDE) are measures of the total body and organ doses respectively, received from external and internal radiation sources.

External doses are calculated for submersion in contaminated air for both the public and worker with appropriate dose conversion factors (DCF) values for submersion for each radionuclide. Inhalation doses are calculated based on the respirable fraction, DCF for inhalation, and breathing rate. ~~Worker dose was calculated based on a facility evacuation time of 10 minutes.~~ The ~~dose conversion factors~~ DCF values used in the analysis are taken from ~~ICRP Publication 119, Compendium of Dose Coefficients based on ICRP Publication 60 (ICRP, 2012)~~ Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors (DCF) for Inhalation, Submersion, and Ingestion (EPA, 1988), and Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil (EPA, 1993).

Worker dose ~~prior to initial evacuation has been evaluated. Immediate~~ was generally calculated over a 30-day interval. The scenario resulting in the release of tritium from the tritium purification system (TPS) gloveboxes uses a 10-day release interval because it is expected that tritium recovery can be accomplished within this time frame. Worker dose also includes the control room occupancy factor used in the calculation of RAF. Operator action inside the facility is not required to stabilize accident conditions.

The public dose was generally calculated over a 30-day interval at the site boundary. The scenario resulting in the release of tritium from TPS gloveboxes uses a 10-day release interval because it is expected that tritium recovery can be accomplished within this time frame. ~~The χ/Q~~

~~values are calculated at the nearest point along the site boundary and at the nearest resident location. The maximum calculated value over all directions of the 50th percentile χ/Q was used for both receptor locations.~~ A ground release was used as the release point.

~~The environmental and meteorological conditions used to develop the atmospheric dispersion factors are discussed in Section 2.3.~~

Conservatism

Additional areas of conservatism included in the determination of radiological consequences include:

- Conservative TSV power history and operational cycle: The TSV power history was derived from nearly continuous TSV operation over a []^{PROP/ECI} period at a power level that exceeds the design power level by ten percent. No credit was taken for medical isotope extraction activities that normally occur during the operation of the SHINE facility.
- Conservative statistical bounding of nuclide inventory: Due to inherent uncertainties in MCNP5, multiple unique sets of results were run through ORIGEN-S to determine the nuclide inventories. The nuclide inventories were analyzed such that a 95 percent confident 95th percentile upper bound was determined for each nuclide. These uncertainties on individual nuclides, 0 to 35 percent, were added to the safety basis inventory to account for the uncertainties inherent to the methods used.
- Conservative estimation of nuclide decay (linear interpolation in lieu of exponential decay): Analyses which account for the decay of nuclides between time steps use linear interpolation in lieu of exponential decay, which increases the available radionuclide inventory at the intervening points.
- Condensation was conservatively neglected in the [LPF radiation transport](#) model.

Uncertainties

Uncertainty in the radionuclide inventory was evaluated using statistical modeling to account for uncertainties associated with the use of Monte Carlo N-Particle Transport Code (MCNP) (LANL, 2011) in the SHINE Best Estimate Neutronics Model (BENM). The modeling produced a nuclide-dependent multiplication factor ranging from approximately 0 to 35 percent increase in the nuclide inventory per nuclide. For the radionuclides which were increased, the average increase was approximately 2.5 percent, and the total estimated increase in inventory was approximately 1 percent. The unweighted uncertainty associated with the multiplication factors was approximately 12 percent. Given that the majority of radionuclides either did not receive an increase or received an increase less than 10 percent and that the multiplication factor only increased the inventory this uncertainty is considered to be negligible.

~~Based on the results of the validation activities for the LPF model, described below, there is no additional uncertainty associated with the LPF model used in the analysis.~~

The DCFs used in the analysis are well-recognized and are used without consideration of uncertainty in the values.

Uncertainty in the χ/Q calculation was estimated by calculating the mean and standard deviation of the 95th percentile values for the 16 sectors. The result of the estimation is ± 25 percent. However, SHINE conservatively uses the value for the highest sector.

Use of Computer Codes

The PAVAN computer code was used to calculate the short-term atmospheric dispersion (χ/Q) factors for an effluent release to the public. PAVAN is described in NUREG/CR-2858, PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations (USNRC, 1982). The code was used as prescribed in Regulatory Guide 1.145, Revision 1, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (USNRC, 1983). No additional validation was performed for the PAVAN code.

~~The LPF model implementation was performed using Mathcad Version 15. The LPF model calculation was validated at three levels: (1) function tests, consisting of tests of individual functions for rate processes, heat transfer, mass transfer, and aerosol behavior; (2) integral tests, which used the model for a simplified transient using one or more rate processes which is then compared against independent solutions to simplified differential equations; (3) use of a simplified LPF model, which notes key leakage rates and integrates the equations for activity for each of the modeled regions assuming the leakage rates are constant. The results of the validation for the LPF model showed good agreement in each of the test categories and revealed no significant sources of uncertainty due to the use of the model. The LIRIC and IMOD correlations used in the LPF model have been extensively validated against experimental data. No additional validation was performed.~~ The ARCON96 computer code was used to calculate the short-term atmospheric dispersion (χ/Q) factors for an effluent release to the control room. ARCON96 is described in NUREG/CR-6331, Revision 1, Atmospheric Relative Concentrations in Building Wakes (USNRC, 1997). The code was used as prescribed in Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (USNRC, 2003b). No additional validation was performed for the ARCON96 code.

The radionuclides included in the target solution inventory are determined using ORIGEN-S (ORNL, 2011) with input from the SHINE BENM which provided the neutron flux and cross-sections for the data library used by ORIGEN-S. ORIGEN-S has been extensively validated for use in calculating burnup in a variety of applications. No additional validation for ORIGEN-S was performed. Additional discussion of the use of ORIGEN-S is provided in [Section 4a2.6](#).

13a2.2.1 ~~IF~~ MAXIMUM HYPOTHETICAL ACCIDENT

~~As described in Subsection 13a2.1.1, the postulated maximum hypothetical accident (MHA) for the irradiation SHINE facility (IF) is a failure of the TSV off-gas system (TOGS) pressure boundary leading to resulting in a release of TSV radioactive off-gases into the TOGS confinement cell. The nitrogen gas purge system (N2PS) actuates, but the process vessel vent system (PVVS) flow path is assumed to be completely blocked, causing a maximum pressurization of the irradiation unit (IU) cell. Therefore, the release of radioactive material from the pressurized primary confinement boundary is hypothetical and bounds other releases of fission products from the IF. A detailed description of this scenario and an evaluation of the radiological consequences is provided in Subsection 13a2.2.7.~~ As described in Subsection 13a2.1.1, the postulated maximum hypothetical accident (MHA) for the irradiation SHINE facility (IF) is a failure of the TSV off-gas system (TOGS) pressure boundary leading to resulting in a release of TSV radioactive off-gases into the TOGS confinement cell. The nitrogen gas purge system (N2PS) actuates, but the process vessel vent system (PVVS) flow path is assumed to be completely blocked, causing a maximum pressurization of the irradiation unit (IU) cell. Therefore, the release of radioactive material from the pressurized primary confinement boundary is hypothetical and bounds other releases of fission products from the IF. A detailed description of this scenario and an evaluation of the radiological consequences is provided in Subsection 13a2.2.7.

13a2.2.1.1 Initial Conditions

~~Initial conditions of the accident are described in Subsection 13a2.1.1.1.~~

13a2.2.1.2 Initiating Event

~~The initiating event for the MHA is the non-credible failure of the N2PS flow path through the PVVS system following a credible TOGS failure (described in Subsection 13a2.2.7). The MHA scenario results in the release of radioactive gases at a higher pressure than analyzed in the credible TOGS failure scenario.~~

13a2.2.1.3 Sequence of Events

~~The accident sequence proceeds as follows:~~

- ~~1. A failure of the primary system boundary (PSB) in the TOGS causes a release of noble gases and iodine into the TOGS cell.~~
- ~~2. The radioactive material is confined by the primary confinement boundary, which is described in Subsection 6a2.2.1.~~
- ~~3. The TSV reactivity protection system (TRPS) is assumed to actuate N2PS based on changes in TOGS flow. It is conservative to assume N2PS actuates, as it pressurizes and increases leak rates from the TOGS cell.~~
- ~~4. The N2PS flow path through the PVVS is assumed to be completely blocked, increasing the total pressurization of the TOGS cell.~~
- ~~5. The released TOGS gases and nitrogen gas pressurize the TOGS cell and begin to flow into the IF by pressure-driven flow through leak paths in the primary confinement boundary.~~
- ~~6. The radioactive material is then dispersed throughout the IF and exits to the environment through building penetrations.~~
- ~~7. Detection of high radiation in the radiological ventilation zone 1 exhaust subsystem (RVZ1e) from the IU cell actuates redundant ventilation dampers and prevents the transport of radioactive material to the environment through this path. The assumed response time for RVZ1e ventilation is 20 seconds from detection of high radiation. A sufficient time delay is provided by design to prevent significant radioactive gases from exiting through this path prior to isolation.~~
- ~~8. The TRPS initiates an IU Cell Safety Actuation signal, which terminates irradiation operations and isolates the primary confinement boundary. The TRPS signal may be initiated by a TOGS failure or a RVZ1e high radiation signal.~~
- ~~9. The main facility ventilation system (i.e., radiological ventilation zone 2 [RVZ2]) is isolated by the engineered safety features actuation system (ESFAS) within 30 seconds of detectable accident conditions. Leakage to the environment continues through unfiltered leakage pathways.~~
- ~~10. Personal dosimeters, local radiation alarms, and alarms in the facility control room notify facility personnel of radiation leakage.~~
- ~~11. Facility personnel evacuate the immediate area within 10 minutes upon actuation of the radiation area monitor alarms.~~
- ~~12. Pressurized flow due to nitrogen is considered to continue for the full accident duration of 30 days.~~

Safety Controls

~~The safety controls credited to mitigate the release of radioactive material in this scenario are:~~

- ~~• Primary confinement boundary~~
- ~~• Ventilation radiation monitors~~
- ~~• Ventilation isolation mechanisms~~
- ~~• TRPS IU Cell Safety Actuation~~

13a2.2.1.4 ~~Damage to Equipment~~

~~The TOGS pressure boundary leak only affects the TOGS operability. No other damage is considered.~~

13a2.2.1.5 ~~Radiation Source Terms~~

~~The initial MAR for the MHA is 100 percent of the noble gases and iodine present in the target solution at the end of []^{PROP/ECI} of continuous 30-day irradiation cycles with a []^{PROP/ECI} downtime between cycles. The power level used for the analysis is 137.5 kW, which is 110 percent of design operating power. The complete inventory of radioactive gases in the PSB is instantaneously transported into the TOGS cell at the beginning of the event sequence.~~

~~The accident source term development is discussed in Section 13a2.2. The LPF model values used in the source term development for the public and worker doses are provided in Table 13a2.2-1 and Table 13a2.2-2, respectively.~~

13a2.2.1.6 ~~Radiological Consequences~~

~~The radiological consequences of this accident scenario are determined as described in Section 13a2.2. The public and worker dose consequences from the MHA are provided in Table 13a3-1 and meet the acceptance criteria.~~

13a2.2.2 INSERTION OF EXCESS REACTIVITY

As discussed in [Subsection 13a2.1.2.3](#), no releases are expected to occur as a result of insertion of excess reactivity events. There are no consequences to the workers or the public from excess reactivity events as discussed below. Accident consequences resulting from excess reactivity events that reference other subsections are evaluated in those respective subsections.

13a2.2.2.1 Initial Conditions

Initial conditions for insertion of excess reactivity events are described in [Subsection 13a2.1.2.1](#).

13a2.2.2.2 Initiating Event

[Subsection 13a2.1.2](#) identifies the postulated initiating events and scenarios with respect to an insertion of excess reactivity.

The subcritical assembly is protected from excessive power with actuation signals from the TRPS on high flux in Mode 1 and Mode 2. When a power excursion occurs, the strong negative

Because the postulated reduction in cooling events do not exceed any design limits or cause damage to the PSB, there is no radiation source term.

13a2.2.3.6 Radiological Consequences

Because the postulated reduction in cooling events do not exceed any design limits or cause damage to the PSB, there are no radiological consequences to workers or the public from a reduction in cooling event.

13a2.2.4 MISHANDLING OR MALFUNCTION OF TARGET SOLUTION

The bounding scenario analyzed as a ~~design-basis-accident (DBA)~~ for mishandling or malfunction of target solution is a loss of the PSB integrity which results in a release of target solution into the IU cell. This scenario is described in [Subsection 13a2.1.4.2](#) as Scenario 1b.

13a2.2.4.1 Initial Conditions

The TSV is operating at 110 percent of its design power limit at the time of the initiating event. Additional initial accident conditions are described in [Subsection 13a2.1.4.1](#).

13a2.2.4.2 Initiating Event

The accident sequence is initiated by a catastrophic loss of PSB integrity. Potential causes of the initiating event are discussed in [Subsection 13a2.1.4.1](#).

13a2.2.4.3 Sequence of Events

It is assumed that the primary confinement boundary is intact and performs a mitigation function with respect to radionuclide transport from the IU cell to the IF. The primary confinement boundary components are designed to maintain their integrity under postulated accident conditions and are maintained in accordance with the facility configuration management and maintenance requirements.

1. A failure of the PSB leads to mixing of irradiated target solution with the IU cell light water pool.
2. Radioactive material enters the gas space above the light water pool and is confined by the primary confinement boundary, which is described in [Section 6a2.2](#).
3. Some radioactive material is transported into the IF through minor leakage paths around penetrations in the confinement boundary.
4. Detection of airborne radiation in RVZ1e actuates the primary confinement boundary isolation valves and an IU trip within 20 seconds of detection. A sufficient time delay is provided by the holdup volume in RVZ1e to prevent radioactive gases from exiting through RVZ1e prior to isolation.
5. The radioactive material is then dispersed throughout the IF and exits the facility to the environment through building penetrations.
6. Detection of high radiation in the RCA actuates ventilation dampers between the RCA and the environment and minimizes the transport of radioactive material to the environment.
7. Personal dosimeters, local radiation alarms, and alarms in the facility control room notify facility personnel of radiation leakage.

8. Facility personnel evacuate the immediate area upon actuation of the radiation alarms.

No operator actions are taken or required to reach a stabilized condition or to mitigate dose consequences.

Following the failure of the PSB, it is assumed that the MAR is instantly well-mixed with the light water pool. Gases immediately evolve out of the pool and into the IU cell gas space. For the purposes of the accident analysis, it is assumed that the N2PS is operating and causes pressurization of the IU cell. Radiation transport is driven by pressure-driven flow between the IU cell and the IF. Reduction in the MAR occurs during the release due to adsorption of iodine onto the IU cell walls and other surfaces until equilibrium conditions are established. The majority of the MAR is transported to the IF through leakage through the primary confinement boundary. Transport to the environment occurs through leakage around penetrations in the RCA boundary.

Safety Controls

The safety controls credited for mitigation of the dose consequences for this accident are:

- Primary confinement boundary
- Ventilation radiation monitors
- Ventilation isolation mechanisms
- Holdup volume in the RVZ1e

13a2.2.4.4 Damage to Equipment

Chemical and radiological contamination may occur to systems within the IU cell. The contamination does not affect the safety function of the affected systems.

Following isolation of the primary confinement boundary, leakage between the IU cell and the IF is driven primarily by pressure-driven flow caused by N2PS. The IU cell sealing is a significant contributor to the function of the primary confinement boundary and will maintain its function under accident conditions.

The light water pool is required to act as a passive heat sink to remove decay heat from the irradiated target solution. The light water pool is constructed with a stainless steel liner surrounded by concrete and maintains the light water pool water inventory and will not be affected by the release of target solution.

13a2.2.4.5 Radiation Source Terms

The initial MAR for this scenario is the TSV target solution inventory at the end of approximately []^{PROP/ECI} of continuous 30-day irradiation cycles with a []^{PROP/ECI} downtime between cycles. The power level used for the analysis is 137.5 kW, which is 110 percent of design operating power. The entire radionuclide inventory in the TSV is instantaneously released to the light water pool and dispersed uniformly throughout the pool.

The accident source term development is discussed in [Section 13a2.2](#). The [LPFRAF](#) model values used in the source term development for the public and worker doses are provided in [Table 13a2.2-1](#) and [Table 13a2.2-2](#), respectively.

Following isolation of the primary confinement boundary, leakage between the TOGS cell and the IF is driven primarily by pressure-driven flow caused by the N2PS. The leakage paths between the cell and the IF are not impacted by the accident sequence. The TOGS cell seals are a significant contributor to the function of the primary confinement boundary and maintains its function under accident conditions.

13a2.2.7.5 Radiation Source Terms

The initial MAR for this scenario is a fraction of the TSV target solution inventory described in [Section 13a2.2](#). The initial MAR for this accident sequence is 100 percent of the noble gases and iodine present in the TOGS gas space while it is operating. Non-volatiles are not included in this accident sequence because the system is designed as a gas-handling system.

The [LPFRAF](#) model values used in the source term development for the public and worker doses are provided in [Table 13a2.2-1](#) and [Table 13a2.2-2](#), respectively.

13a2.2.7.6 Radiological Consequences

The radiological consequences of this accident scenario are determined as described in [Section 13a2.2](#). The results of the determination are shown in [Table 13a3-1](#) and meet the accident dose criteria.

13a2.2.8 LARGE UNDAMPED POWER OSCILLATION

As described in [Subsection 13a2.1.8](#), power oscillations that occur in the subcritical assembly are self-limiting as a result of the inherent design and safety characteristics of the subcritical assembly, operating parameters, and plant response to transients. TRPS setpoints for high wide range and high time-averaged neutron flux are set to actuate on high neutron flux before a large power oscillation occurs that challenges design limits. The IU Cell Safety Actuation results in the TSV dump valves opening and target solution draining from the TSV to the TSV dump tank. Thus, there are no consequences to workers or the public.

13a2.2.8.1 Initial Conditions

Initial accident conditions are described in [Subsection 13a2.1.8.1](#).

13a2.2.8.2 Initiating Event

Potential causes of power oscillations in the TSV are described in [Subsection 13a2.1.8.1](#).

13a2.2.8.3 Sequence of Events

The accident sequence proceeds as follows:

1. An oscillation in power occurs as a result of one of the potential causes described in [Subsection 13a2.1.8.1](#).
2. TSV reactivity oscillates due to the power oscillation but does not become undamped due to inherent design and safety characteristics of the TSV and operating parameters.
3. TRPS high neutron flux limits cause the IU to shutdown before a power oscillation challenges design limits.

- Administrative controls on maintenance and use of combustible materials
- Catchment pans for the high voltage power supplies

Exothermic Chemical Reaction

Exothermic chemical reaction scenarios are described in [Subsection 13a2.1.10](#).

Internal Flooding

Postulated internal flooding scenarios in the IF do not result in radiological consequences, as described in [Subsection 13a2.1.11](#).

Dynamic Effects

Dynamic effects are not present at the main production facility, as described in [Subsection 13a2.1.11](#).

Human Intervention Interactions

As described in [Subsection 13a2.1.11](#), human intervention interactions as accident scenario initiating events are described in other sections in this chapter as applicable.

13a2.2.11.4 Damage to Equipment

No damage to equipment occurs due to system interaction events since the TRPS initiates an IU Cell Safety Actuation or IU Cell Nitrogen Purge as needed prior to exceeding any design limits.

13a2.2.11.5 Radiation Source Terms

Because the postulated system interactions do not exceed any design limits or cause damage to the PSB, there is no radiation source term.

13a2.2.11.6 Radiological Consequences

Because the postulated system interactions do not exceed any design limits or cause damage to the PSB, there are no radiological consequences to workers or the public. Accident consequences resulting from system interactions that are referenced to other subsections in [Chapter 13](#) are evaluated in those subsections.

13a2.2.12 FACILITY-SPECIFIC EVENTS

The majority of the evaluated facility-specific events do not have radiological consequences. The events which do have radiological consequences are related to the release of tritium into the facility from the neutron driver assemblies or from the ~~tritium purification system~~TPS. Three potential locations for the release of tritium were analyzed to determine the dose consequences and necessary controls. The results of the analysis are presented in this subsection.

Safety Controls

The safety controls credited for mitigation of the dose consequences for this accident are:

- Primary confinement boundary (IU cell plugs and seals)
- TPS Train Isolation on high TPS target chamber supply pressure or high TPS target chamber exhaust pressure
- IU cell ventilation isolations
- Holdup volume in the RVZ1e

13a2.2.12.1.4 Damage to Equipment

Failure of the NDAS vacuum boundary does not cause subsequent damage to equipment.

After the initial IU cell pressurization has reached equilibrium, leakage between the IU cells and the IF is driven primarily by barometric breathing. The leakage paths between the cells and the IF are not impacted by the accident sequence.

13a2.2.12.1.5 Radiation Source Terms

The initial MAR for this scenario is []^{PROP/ECI} of tritium from the neutron driver assembly in the IU cell.

The accident source term development is discussed in [Section 13a2.2](#). The [LPFRAF](#) model values used in the source term development for the public and worker doses are provided in [Table 13a2.2-1](#) and [Table 13a2.2-2](#), respectively.

13a2.2.12.1.6 Radiological Consequences

The radiological consequences of this accident scenario are determined as described in [Section 13a2.2](#).

The radiological consequences of this accident scenario are provided in [Table 13a3-1](#) and meet the accident dose criteria.

13a2.2.12.2 Tritium Release into the Tritium Purification System Glove-Box

A release of the tritium inventory from the TPS is analyzed as a DBA. This accident is described in [Subsection 13a2.1.12.3](#) as TPS Scenario 1. This analysis establishes bounding radiological conditions for a release of tritium due to a TPS process deflagration, release of tritium to the facility stack, and release of tritium from the tritium storage bed.

13a2.2.12.2.1 Initial Conditions

Initial conditions for facility-specific events are described in [Subsection 13a2.1.12.1](#).

13a2.2.12.2.2 Initiating Event

An event causes a break in the tritium piping and vessels such that the uncontrolled release of the entire tritium in-process inventory occurs within the tritium confinement boundary. The tritium

confinement boundary is described in detail in [Section 6a2.2](#). Potential causes of the initiating event are discussed in [Subsection 13a2.1.12.3](#).

13a2.2.12.2.3 Sequence of Events

It is assumed that the tritium confinement boundary is intact and performs a mitigation function with respect to radionuclide transport from the TPS to the IF. The tritium confinement boundary components are designed to maintain their integrity under postulated accident conditions and are maintained in accordance with the facility configuration management and maintenance programs.

1. The initiating event is a seismic event that causes a break in ~~the tritium piping and vessels which~~two TPS trains and instantaneously releases the ~~entire~~ tritium inventory of ~~the TPS system~~ into a their respective TPS gloveboxes.
2. For the first 20 seconds, tritium escapes from each of the gloveboxes to the IF through the glovebox pressure control exhaust process vent to RVZ1.
3. The glovebox ventilation shuts down after 20 seconds due to the glovebox tritium monitors.
4. During the 30 seconds after the initiating event, the TPS room vents to the IF at an elevated rate due to the facility RVZ2 ventilation system.
5. The RVZ2 ventilation damper from the TPS room isolates after 30 seconds due to the glovebox tritium monitors.
6. The radioactive material is then dispersed throughout the IF and exits the facility to the environment through building penetrations.
7. Personal dosimeters, local radiation alarms, and alarms in the facility control room notify facility personnel of radiation leakage.
8. Facility personnel evacuate the immediate area within 10 minutes upon actuation of the radiation area monitor alarms.

Throughout the accident sequence, the leakage rate between ~~the~~each TPS glovebox and the TPS room is constant. After the TPS room ventilation is isolated, radiation transport is driven by air exchange between ~~the~~each TPS glovebox and the IF. Transport to the environment occurs through RCA boundary leak paths. The accident duration used in this analysis is 10 days, after which it is assumed that recovery actions will have occurred to stop further release and dispersion of radioactive material.

Safety Controls

The safety controls credited for mitigation of this accident are:

- TPS room ventilation isolations
- Glovebox pressure control and VAC/ITS ventilation isolations
- TPS glovebox tritium radiation monitors
- Tritium confinement boundary, as described in [Section 6a2.2](#)

In addition, TPS glovebox deflagration is prevented by:

- TPS glovebox gas space inerted with helium
- TSP glovebox minimum volume prevents deflagration conditions

13a2.2.12.2.4 Damage to Equipment

Failure of the TPS piping and vessels does not cause subsequent damage to other equipment.

13a2.2.12.2.5 Radiation Source Terms

The initial MAR for this scenario is ~~3~~200,000 curies of tritium from the TPS equipment in the TPS glovebox. |

The accident source term development is discussed in Section 13a2.2. The LPFRAF model values used in the source term development for the public and worker doses are provided in Table 13a2.2-1 and Table 13a2.2-2, respectively. |

13a2.2.12.2.6 Radiological Consequences

The radiological consequences of this accident scenario are determined as described in Section 13a2.2. The radiological consequences of this accident scenario are provided in Table 13a3-1 and meet the accident dose criteria.

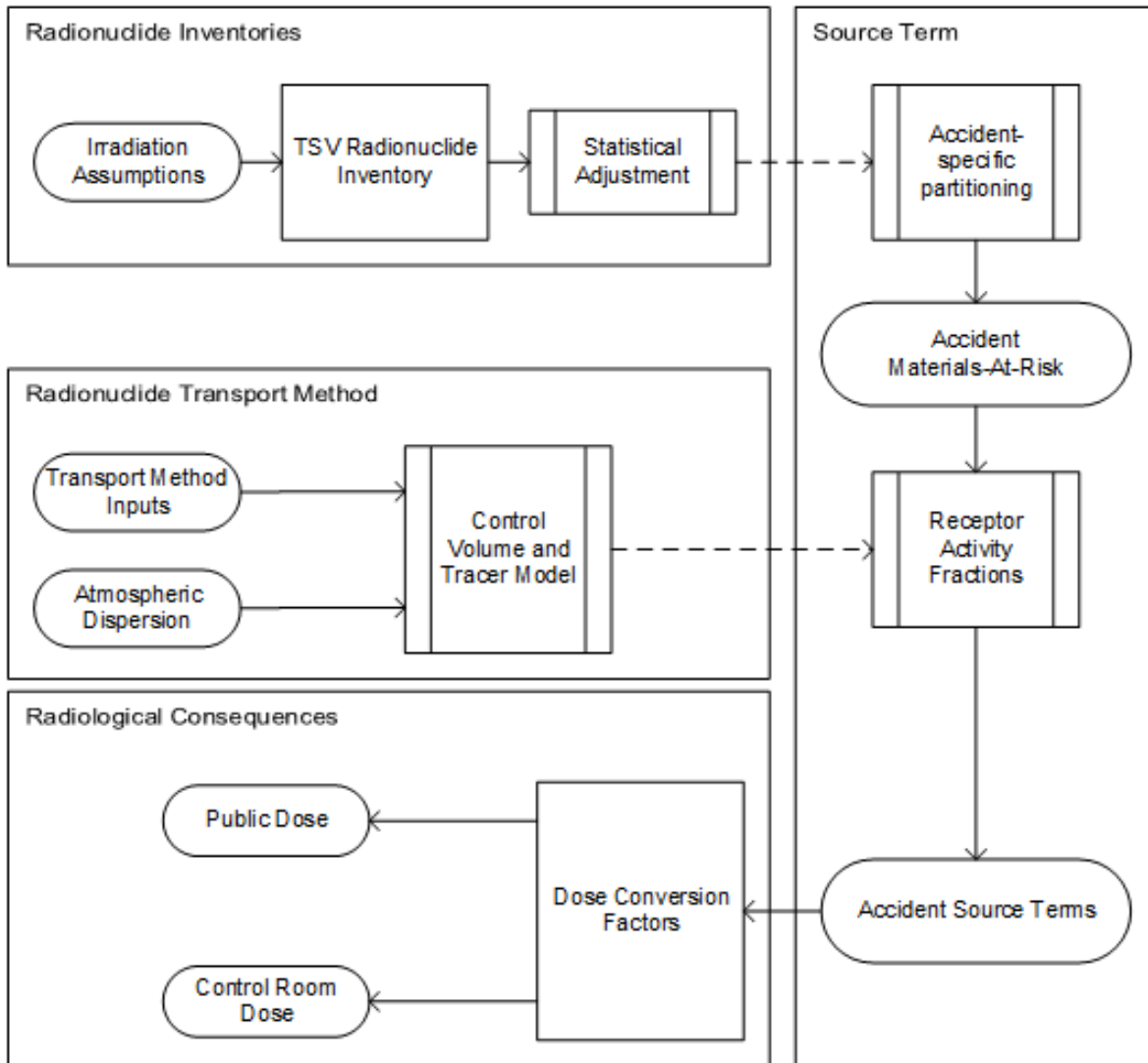
Table 13a2.2-1 – Summary of Radiation Transport Terms (Public)

Accident Category	ARF x LPF Receptor Activity Fraction (RAF)				
	Nobles (30-day)	Iodine (30-day)	Non-volatiles (30-day)	Tritium (10-day)	Tritium (30-day)
Maximum Hypothetical Accident (Subsection 13a2.2.1)	9.98E-01	9.98E-01	0	N/A	N/A
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	9.98 1.30 E-01 3	1.22 7.64 E-01 5	8.39 1.16 E-01 9	N/A	N/A
External Events (Subsection 13a2.2.6)	N/A	N/A	N/A	N/A	3.66 4.07 E-01 4
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	9.98 1.41 E-01 3	5.72 3.69 E-01 4	0	N/A	N/A
Facility-Specific Events (Subsection 13a2.2.12)					
• Tritium Release into an IU Cell	N/A	N/A	N/A	N/A	3.66 4.07 E-01 4
• Tritium Release into the Tritium Purification System Glovebox	N/A	N/A	N/A	1.78E-01 4	N/A

Table 13a2.2-2 – Summary of Radiation Transport Terms (Worker)

Accident Category	ARF x LPF (10 minute) <u>Receptor Activity Fraction (RAF)</u> <u>(30 days)</u>			
	Nobles	Iodine	Non-volatiles	Tritium
Maximum Hypothetical Accident (Subsection 13a2.2.1)	1.19E-02	1.19E-02	0	N/A
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	8.2455E-03 <u>1</u>	6.403E-05 <u>2</u>	9.697.45E-11 <u>07</u>	N/A
External Events (Subsection 13a2.2.6)	N/A	N/A	N/A	4.42.87E-01
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	1.199.92E-02 <u>1</u>	1.173.23E-02 <u>1</u>	0	N/A
Facility-Specific Events (Subsection 13a2.2.12)				
• Tritium Release into an IU Cell	N/A	N/A	N/A	4.42.87E-01
• Tritium Release into the Tritium Purification System Glovebox	N/A	N/A	N/A	1.08.03E-04 <u>1</u> <u>(10 days)</u>

Figure 13a2.2-1 – Radiological Consequence Assessment



13a3 SUMMARY AND CONCLUSIONS

This section presents the summary and conclusions for the accident analysis for the irradiation facility (IF).

The following accident categories were addressed for the irradiation facility:

- Maximum hypothetical accident (MHA)
- Insertion of excess reactivity
- Reduction in cooling
- Mishandling or malfunction of target solution
- Loss of off-site power
- External events
- Mishandling or malfunction of equipment
- Large undamped power oscillations
- Detonation and deflagration affecting the primary system boundary
- Unintended exothermic chemical reactions other than detonation
- System interaction events
- Facility-specific events

The dose consequences of the bounding accident scenarios evaluated for each accident category are provided in [Table 13a3-1](#).

The analyses in this section evaluated the applicable radiological consequences of these accidents and demonstrated that an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not receive a radiation dose in excess of ~~500 m~~1 rem total effective dose equivalent (TEDE) for the duration of the accident. |

Radiological consequences to workers were also evaluated and shown to not exceed 5 rem TEDE during the accident.

SHINE has established the MHA based on the maximum consequence to the public. The MHA itself is not a DBA; however, it is used as a metric for understanding radiological risk from the facility.

Table 13a3-1 – Irradiation Facility Accident Dose Consequences

Accident Category (Bounding Scenario)	Public Dose TEDE (mrem)	Worker Dose TEDE (mrem)
Maximum Hypothetical Accident (Subsection 13a2.2.1)		
TOGS failure with complete PVVS blockage	366	4800
Insertion of Excess Reactivity (Subsection 13a2.2.2)	No consequences	
Reduction in Cooling (Subsection 13a2.2.3)	No consequences	
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)		
<ul style="list-style-type: none"> Primary system boundary leak into an IU cell 	65372	4480555
Loss of Off-Site Power (LOOP) (Subsection 13a2.2.5)	No consequences	
External Events (Subsection 13a2.2.6)	406292	4930588
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	234727	47601940
Large Undamped Power Oscillations (Subsection 13a2.2.8)	No consequences	
Detonation and Deflagration affecting the Primary System Boundary (Subsection 13a2.2.9)	No consequences	
Unintended Exothermic Chemical Reactions other than Detonation (Subsection 13a2.2.10)	No consequences	
System Interaction Events (Subsection 13a2.2.11)	No consequences	
Facility-Specific Events (Subsection 13a2.2.12)		
<ul style="list-style-type: none"> Tritium Release into an IU Cell 	437	64674
<ul style="list-style-type: none"> Tritium Release into the Tritium Purification System Glove Box 	482798	2521380

13a4 REFERENCES

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The PVVS handles the off-gas resulting from the processes of the IF and the RPF. The PVVS is classified as a radiochemical operation and poses a radiological hazard. This process contains radionuclides removed from the off-gas.

The FCRS stores and supplies reagents to the processes of the RPF. The FCRS is classified as an operation with hazardous chemicals and poses a chemical hazard. The system contains no SNM or radionuclides.

13b.1.2 ACCIDENT INITIATING EVENTS

The design basis accidents (DBAs) identified in this section are initiating events (IEs) followed by credible accident scenarios that range from anticipated events, such as a loss of electrical power, to events that, while still credible, are considered unlikely to occur during the lifetime of the facility. The maximum hypothetical accident (MHA) is ~~an accident scenario~~ also defined to result in ~~the worst case (bounding)~~ radiological consequences for the SHINE facility. ~~Although the MHA is an accident scenario, it does not define a credible initiating event or accident progression, except for what is necessary to evaluate the consequences. Its purpose is to provide the most limiting consequence for the facility that bounds all credible DBAs.~~

DBAs were identified using the following sources of information:

- IEs and accidents identified in the Interim Staff Guidance Augmenting NUREG-1537 (USNRC, 2012)
- Hazard and operability (HAZOP) studies, failure modes and effects analyses (FMEA), and the PHA methods
- Experience of the hazard analysis team

The DBA identification process resulted in a series of accident sequences that were then categorized into the following accident types:

- MHA
- External Events
- Critical Equipment Malfunction (i.e., Malfunction or Mishandling of Equipment)
- Inadvertent Nuclear Criticality in the RPF
- RPF Fire
- Hazardous Chemical Accidents

The effects of a loss of off-site power (LOOP) and operator errors were considered as initiating events within the scope of the PHA and were not classified as separate accident types. Qualitative evaluations are performed on the DBAs to further identify the bounding or limiting accidents and scenarios, including the partial loss of systems or functions that could result in the highest potential consequences. These evaluations are based on a review of identification of causes, the initial conditions, and assumptions for each accident.

Using the range of accident scenarios identified, each scenario was qualitatively evaluated for its potential chemical or radiological consequences. Scenarios that presented potential consequences above the appropriate evaluation guidelines for worker or public exposure were then subject to control selection. Appropriate preventative or mitigative controls were identified to reduce the overall risk of the evaluated scenarios to within acceptable limits. For accident sequences that are not prevented and have mitigative controls applied, the radiological or

chemical consequences were quantitatively evaluated to demonstrate the effectiveness of the selected controls. The radiological consequences of accidents that were selected for additional evaluation are further evaluated in [Section 13b.2](#). The accident analysis for chemical exposures is provided in [Section 13b.3](#).

13b.1.2.1 Maximum Hypothetical Accident ~~in the RPF~~

The MHA ~~in the RPF is a fire in a carbon guard bed with degraded performance of the downstream carbon delay beds.~~

~~The initiating event for this accident is ignition of transient combustibles or exothermic chemical reaction in the bed resulting the formation of a hot spot and eventually a fire. Temperature indication normally detects an increase in temperature and the affected carbon guard bed is isolated. The carbon guard bed releases its inventory to the downstream carbon delay beds which are normally credited with adsorbing 99 percent of the released iodine. For the MHA, the carbon delay beds are assumed to be operating at a reduced efficiency of 95 percent. Analysis demonstrates that the carbon guard bed fire will not propagate to the carbon delay beds. This scenario is described further in Subsection 13b.2.1 for the SHINE facility is identified in Subsection 13a2.1.1.~~

13b.1.2.2 External Events

The external initiating events for the RPF that were evaluated include seismic events, tornados or high winds, small aircraft impacts, flooding, fires, and chemical releases. The SHINE main production facility is designed to withstand credible external events, as described in [Subsection 13a2.1.6](#). External events were considered as potential IEs for a number of accident scenarios that fall within the other accident categories. The design basis seismic event results in potential chemical consequences, as described below and in [Section 13b.3](#).

A design basis flooding event could result in potential flooding of internal vaults, trenches, and pits, as well as the URSS and TSPS rooms. Flooding of the areas that contain fissile material reduces the margin to criticality and challenges the double-contingency principle. Water intrusion into these areas is minimized by sealed covers for the below-grade locations and by elevated room floors for the URSS and TSPS rooms. The local maximum probable precipitation event resulting in a 100-year flood will not exceed the first-floor entrance elevations, providing additional margin.

External event scenarios are further described in [Subsection 13b.2.3](#).

13b.1.2.3 RPF Critical Equipment Malfunction

Critical equipment malfunctions in the RPF were evaluated as part of the accident analysis. Multiple scenarios were identified as having potential radiological consequences and were selected for additional evaluation. The identified scenarios are described below. For each scenario, the controls that act to reduce the likelihood or consequences of the accident are listed. For scenarios that require mitigative controls, the radiological consequence assessments for limiting exposures are presented in [Subsection 13b.2.4](#).

13b.2 ANALYSES OF ACCIDENTS WITH RADIOLOGICAL CONSEQUENCES

Several design basis accidents described in [Section 13b.1](#) result in a release of radioactive materials into or outside the controlled areas of the facility.

The analyses in this section evaluate the applicable radiological consequences of these accidents to demonstrate that an individual located in the unrestricted area following the onset of a postulated accidental release of licensed material would not receive a radiation dose in excess of ~~500 m~~1 rem total effective dose equivalent (TEDE) for the duration of the accident.

Radiological consequences to workers are also evaluated and are shown to not exceed 5 rem TEDE during the accident.

13b.2.1 MAXIMUM HYPOTHETICAL ACCIDENT ~~IN THE RPF~~

~~As described in Subsection 13a2.1.1, the postulated maximum hypothetical accident (MHA) in for the radioisotope production SHINE facility (RPF) is a fire in a carbon guard bed with degraded carbon delay bed efficiency. It is postulated that 100 percent of the radionuclide inventory is released from the guard bed and flows downstream into the carbon delay beds and is then released to the facility stack. The performance of the carbon delay beds is assumed to be degraded to 95 percent. The automatically mitigated release from a credible carbon guard bed fire is discussed in Subsection 13b.2.6.2 failure of the target solution vessel (TSV) off-gas system (TOGS) pressure boundary resulting in a release of off-gas into the TOGS cell. A detailed description of this scenario and an evaluation of the radiological consequences is provided in Subsection 13a2.2.12.2.~~

Initial Conditions

~~The process vessel vent system (PVVS) is operating normally, with nominal flow through one carbon guard bed.~~

~~The affected carbon guard bed contains radioactive iodine from RPF process streams. The material at risk (MAR) in this scenario is a combination of gases from eight irradiation units (IU), with various modifiers applied to account for decay and operational sequencing.~~

Initiating Event

~~An upset or malfunction in the PVVS results in high moisture or high temperature flow through the carbon guard bed. The high moisture or high temperature results in ignition of the carbon guard bed absorber media. Potential initiating events for this scenario are discussed further in Subsection 13b.1.2.1.~~

Sequence of Events

- ~~1. Ignition of one of the carbon guard beds occurs, resulting in an exothermic release of 100 percent of the stored radioactive material to the gas piping downstream of the guard bed.~~
- ~~2. The downstream carbon delay beds adsorb 95 percent of the radioactive material.~~
- ~~3. The radioactive material exiting the carbon delay beds is released to the environment through the PVVS and facility stack.~~

- ~~4. The maximum gas temperature entering the carbon delay beds is 130°C, which is insufficient to propagate the fire condition to the carbon delay bed. Temperature-instrumentation in the carbon guard bed detects increased temperature and the carbon-guard beds are isolated. However, the isolation of the carbon guard bed is not credited for limiting the release of radioactive materials from the carbon guard bed in this scenario.~~

~~*Damage to Equipment*~~

~~The occurrence of fire damages the affected carbon guard bed and eliminates its ability to function. No other damage to the PVVS system or its components occurs.~~

~~*Transport of Radioactive Material*~~

~~The methods used to calculate radioactive material transport are described in Section 13a2.2. The leak path factor (LPF) model terms used in this accident are provided in Table 13b.2-1. For this accident, the release of material from the guard bed is assumed to be instantaneous and is transported to the environment at an increased rate.~~

~~*Radiation Source Terms*~~

~~The initial MAR for this scenario is a portion of the iodine gas inventory evolved from target solution during normal operations. Development of the accident source term for this scenario is discussed further in Section 13a2.2.~~

~~The iodine gas inventory is produced by fission and decay of fission products and continuously evolved from the target solution and through the target solution vessel (TSV) off-gas system (TOGS) during operations. Partitioning fractions for iodine gas are used to describe the quantities of iodine in solution that move to the RPF. Removal of iodine by the TOGS zeolite beds are credited for all gases that are transported to the RPF. The MAR uses selected time intervals for the most recent purges (i.e., []^{PROP/ECI}) to account for the operational sequencing of the combined eight IUs. The MAR assumes the combined iodine gas inventory produced by eight IUs over approximately []^{PROP/ECI} of irradiation with the most recent purges of []^{PROP/ECI}. The iodine accumulates in the carbon guard bed and decays.~~

~~*Radiological Consequences*~~

~~The radioactive material is contained in the PVVS system and does not result in worker exposure. The radiological consequences of this accident scenario are determined as described in Section 13a2.2. The results of the determination are provided in Table 13b.2-2.~~

13b.2.2 LOSS OF ELECTRICAL POWER

Loss of off-site power (LOOP) was evaluated in the accident analysis as an initiating event for a number of critical equipment malfunction scenarios. A facility-wide LOOP results in automatic actuation of multiple facility engineered safety features, which act to ensure the risk associated with radiological or chemical releases is reduced to within acceptable limits. The facility-wide LOOP does not result in system or component failures within the RPF that result in unacceptable

**Table 13b.2-1 – Radiation Transport Factors
(Sheet 1 of 2)**

Accident Scenario	Radionuclide Group	LPF Model (ARF × LPF) <u>Receptor</u> <u>Activity Fraction (RAF)</u>
Maximum Hypothetical Accident	Iodine	1.0
Spill of Target Solution in the Supercell	Nobles	8.121.00E-013 Public 1.86.560E-021 Worker
	Iodine	7.101.49E-036 Public 4.201.40E-03 Worker
Spill of Eluate Solution in the Supercell	Non-Volatile	3.461.38E-057 Public 3.579.40E-065 Worker
	Nobles	8.121.04E-013 Public 1.806.96E-021 Worker
Spill of Target Solution in the RPF Pipe Trench	Iodine	7.101.88E-036 Public 4.201.77E-03 Worker
	Non-Volatile	3.461.52E-057 Public 3.571.06E-064 Worker
Spill of Target Solution from a Tank	Nobles	1.273E-014 Public 3.857.71E-032 Worker
	Iodine	1.202.49E-027 Public 3.772.26E-034 Worker
Spill of Target Solution from a Tank	Non-Volatile	3.851.11E-068 Public 9.226.71E-106 Worker
	Nobles	2.081.36E-014 Public 9.318.24E-032 Worker
Spill of Target Solution from a Tank	Iodine	1.601E-028 Public 8.981.51E-035 Worker
	Non-Volatile	5.111.18E-068 Public 2.207.18E-069 Worker

**Table 13b.2-1 – Radiation Transport Factors
(Sheet 2 of 2)**

Accident Scenario	Radionuclide Group	LPF Model (RAF x LPF) Receptor Activity Fraction (RAF)
Spill of Waste Solution in RLWI	All Nobles	4.05.66E-03 Public 6.69E+00 Worker
	Iodine	5.66E-03 Public 6.69E+00 Worker
	Non-Volatile	1.13E-06 Public 1.34E-03 Worker
PVVS Carbon Guard Bed Fire	Iodine	1.0 (5.66E-03 Public- Only) 6.70E+00 Worker
PVVS Carbon Delay Bed Fire	Nobles	4.81.50E-024 (Public- Only) 1.63E-01 Worker

Table 13b.2-2 – Radioisotope Production Facility Accident Dose Consequences

Accident Scenario	Public Dose TEDE (mrem)	Worker Dose TEDE (mrem)
Maximum Hypothetical Accident	403	No consequences
Spill of Target Solution in the Supercell	7 42	3 176
Spill of Eluate Solution in the Supercell	14 88	40 122
Spill of Target Solution in the RPF Pipe Trench	4 22	170 40
Spill of Target Solution from a Tank	5 24	398 42
Spill of Waste Solution in RLWI	13 557	861 1880
PVVS Carbon Delay Bed Fire	39 532	No- consequences40
PVVS Carbon Guard Bed Fire	81 546	No- consequences139 0

LIST OF TABLES

<u>Number</u>	<u>Title</u>
13a2.2-1	Summary of Radiation Transport Terms (Public)
13a2.2-2	Summary of Radiation Transport Terms (Worker)
13a3-1	Irradiation Facility Accident Dose Consequences
13b.2-1	Radiation Transport Factors
13b.2-2	Radioisotope Production Facility Accident Dose Consequences
13b.3-1	Inventory <u>Quantities</u> of <u>In-Process</u> Hazardous Chemicals
13b.3-2	Hazardous Chemical Source Terms and Concentration Levels

13b.3 ANALYSES OF ACCIDENTS WITH HAZARDOUS CHEMICALS

~~The potential hazards of the chemicals proposed to be used at the SHINE facility have been evaluated. The analysis has been performed for hazardous chemicals within the facility that interact with or are produced from licensed materials. The analysis also includes other toxic and reactive hazardous chemicals that are present in the SHINE facility but do not directly interact with licensed materials. Therefore, the analysis is bounding for all hazardous chemicals that interact with or are produced from licensed materials. Safety related or administrative controls have been developed only for those systems or processes where the hazardous chemical is produced from or otherwise associated with licensed materials. Consequence or chemical dose modeling are evaluated using dispersion models and/or computer codes that conform to the methodologies described in NUREG/CR 6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook (USNRC, 1998).~~

~~Quantitative exposure standards are selected to meet acceptable limits for public and worker health and safety. The quantitative acceptance limits are taken from the Protective Action Criteria (PAC) values (USDOE, 2018), which correspond to the Acute Exposure Guideline Levels (AEGs), Emergency Response Planning Guidelines (ERPGs), or Temporary Emergency Exposure Limits (TEELs) values for such chemicals. Two exceptions are applied to rhodium chloride and uranyl peroxide, which do not have published PAC values. For these chemicals, acceptance values are assigned based on Occupational Safety and Health Administration (OSHA) occupational exposure limits.~~

SHINE has evaluated the potential hazards of chemicals within the main production facility. These include chemicals that are licensed materials or have licensed materials as precursor compounds, or substances that physically or chemically interact with licensed materials and that are toxic, explosive, flammable, corrosive, or reactive to the extent that they endanger life or health. These include substances that are comingled with licensed material or are produced by a reaction with licensed material. These do not include substances prior to process addition to licensed materials or after process separation from licensed materials. The analysis is therefore bounding for all hazardous chemicals produced from or comingled with licensed materials.

The hazardous chemical consequence assessment is performed to demonstrate that potential consequences meet the SHINE Safety Criteria, as defined in Section 3.1, for the public and workers (i.e., a radiologically controlled area [RCA] worker and a control room operator). The inventory of in-process hazardous chemicals used at the SHINE facility, compiled by process location and quantity, is provided in Table 13b.3-1.

Chemical Process Descriptions

The chemical processes used in the SHINE facility are described in Sections 4b.3, 4b.4, 9a2.2, and 9b.7.

Chemical Accidents Description and Source Term Determination

~~The only chemical accident scenario with the potential to exceed established chemical exposure guidelines for workers is a seismic event resulting in the failure or overturning of the uranium receipt and storage system (URSS) uranium oxide storage rack, causing multiple storage can failures. The uranium storage racks are seismically qualified to maintain their structure and position during a seismic event, which prevents the potential chemical exposure.~~

~~The failure of a single can during transfer or handling operations does not result in chemical dose consequences being exceeded.~~ For each of the hazardous chemicals identified in

Table 13b.3-1, a release scenario is postulated. Each postulated scenario defines the material at risk (MAR) as the largest quantity present in a single vessel or process location. The MAR may therefore be less than the maximum quantities identified in Table 13b.3-1 (e.g., the total waste stream may be subdivided into multiple tanks). The chemical source term is then evaluated using the following methodology.

The formula for determining the source term (ST), the amount of hazardous material made airborne and respirable, of each chemical release is given by the following formula:

$$ST = MAR \times ARF \times RF \times DR \times LPF$$

Where:

- MAR is the material at risk, the quantity of material potentially affected;
- ARF is the airborne release fraction;
- RF is the respiratory fraction;
- DR is the damage ratio, the portion of the MAR affected by the release scenario (conservatively assumed to be 1.0 for all scenarios); and
- LPF is the leak path factor, the proportion of airborne material that leaks out of a building or enclosure. A leak path factor of 0.1 is applied for scenarios that occur in confinements (i.e., supercell, gloveboxes, subgrade vaults) to model the confinement barrier for the spill locations. This represents a 10 percent vol/hr leak rate from confinements. This conservatively bounds leak rates determined through more detailed analyses in the radiological dose analyses for these confinements.

Estimation of the source term falls into two categories:

- 1) Non-volatile chemicals (e.g., solids, liquids with low vapor pressures), and
- 2) Volatile chemicals (i.e., liquids with vapor pressures in excess of 10 Torr at 100°F).

For non-volatile chemicals, the MAR is taken to be the largest quantity of the chemical present in a single vessel or process location. Values for the ARF and RF are taken from the guidance in NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook (USNRC, 1998).

For volatile liquids, the MAR x ARF x RF product is replaced by the total mass released as calculated by the ALOHA (Areal Locations of Hazardous Atmospheres) computer code, Version 5.4.7.

To account for uncertainty in the MAR quantities, a multiplier of 1.2 is applied to the calculated source term.

The MAR and source terms for each chemical release scenario are presented in Table 13b.3-2.

Chemical Accident Consequences

A hazardous chemical consequence assessment was performed to demonstrate that potential consequences are within acceptable limits. This assessment determines if the release of

hazardous chemicals from the SHINE facility could lead to exceeding the Protective Action Criteria (PAC) values. ~~The inventory of chemicals used at the SHINE facility is presented in Table 13b.3-1.~~

A consequence analysis for the public and nearest residence was performed using the ~~ALOHA (Areal Locations of Hazardous Atmospheres)~~ PAVAN (An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations) computer code (USDOE ~~NRC, 2013~~ NRC, 1982). The ~~material at risk (MAR) present for each chemical is identified in Table 13b.3-2. The MAR is assumed to be the largest quantity of material that can be present for a single release event. In most cases, this is equal to one half of the chemical inventory listed in Table 13b.3-1, but limited by the capacity of a single storage container (i.e., a single container spill). One exception is the release of sulfur hexafluoride, in which the inventory from all eight neutron driver assembly system (NDAS) pressure vessels is assumed to be released as a result of a seismic event.~~ chemical exposure to the public and nearest residence is then calculated using the 95th percentile atmospheric dispersion factors (χ/Q) calculated using the PAVAN computer code.

~~To model the chemical exposure to the facility workers, the evaporation rates for liquids or directly released material for spilled solids from the ALOHA calculations are used to determine the amount of each chemical released into the facility atmosphere. The evaporation rate is determined by setting the assumed wind speed to the minimum value allowed in ALOHA, which simulates the indoor air movement. The puddle area used for evaporation is modeled by using the room dimensions that the chemical is stored in. The resulting concentration of a chemical release within the facility is calculated as a homogenous mixture within the radioisotope production facility (RPF) volume.~~

~~To model the chemical exposure to the members of the public, the evaporation rates from the chemical puddle are calculated in the same way as above for the worker dose. The evaporated chemical is then dispersed using a 4.2 meters per second (m/s) mean wind speed, which is based on meteorological data from the Southern Wisconsin Regional Airport. The resulting chemical concentration for the site boundary and the nearest residence is determined. The summary of the chemical concentration calculation is provided in Table 13b.3-2. Note that two additional chemicals (hydrogen fluoride and sulfur dioxide) are included in Table 13b.3-2. These are decomposition byproducts of the sulfur hexafluoride gas used in the NDAS and are not normally stored on site.~~

~~The chemical concentrations for workers and the nearest residence are below the PAC-1 values.~~ source term is used to determine the amount of each chemical released into the facility atmosphere. For the RCA worker, the total source term released into the facility is assumed to be well mixed within the building free volume (i.e., irradiation facility [IF] or radioisotope production facility [RPF]) to determine a chemical concentration. For the control room operator, the same concentration is assumed to be released from the facility roll-up door and is transported to the facility ventilation intake that services the control room. ALOHA is then used to calculate the indoor concentration at the location of the ventilation intake louver that services the control room.

Quantitative exposure standards are selected to meet acceptable limits for public and worker health and safety. The quantitative acceptance limits are taken from the PAC values (USDOE, 2018), which correspond to the Acute Exposure Guideline Levels (AEGs), Emergency Response Planning Guidelines (ERPGs), or Temporary Emergency Exposure Limits (TEELs) values for such chemicals. Exceptions are applied to rhodium chloride, uranyl sulfate, and uranyl

peroxide, which do not have published PAC values. For these chemicals, acceptance limits were developed using guidance from DOE-HDBK-1046-2016, Temporary Emergency Exposure Limits for Chemicals: Method and Practice (USDOE, 2016).

Three chemical accident scenarios are identified that have the potential to exceed established chemical exposure acceptance limits for workers if safety-related controls are not applied:

- Sulfuric acid: A spill from a subgrade liquid waste collection tank may potentially exceed the control room chemical consequence limit. The subgrade vault is credited as a safety-related control to limit the source term to maintain the peak control room concentration to less than the PAC-2 limit.
- Uranium oxide: A seismic event resulting in the failure or overturning of the uranium receipt and storage system (URSS) uranium oxide storage rack, causing multiple storage can failures. The uranium storage racks are seismically qualified to maintain their structure and position during a seismic event, which prevents the potential chemical exposure. The failure of a single can during transfer or handling operations does not result in chemical dose consequences which exceed acceptance limits.
- Uranium oxide: A spill of uranium oxide powder in the URSS glovebox or target solution preparation system (TSPS) glovebox causes a quantity of the powder to become airborne. The gloveboxes are seismically qualified to maintain their low leakage boundary during a seismic event, which limits the chemical exposure to workers to within acceptable limits.

The acceptance limits established for chemical consequence are that the PAC-1 limit shall not be exceeded for members of the public, and that PAC-2 limits shall not be exceeded for workers. The results in Table 13b.3-2 show that no chemical consequence exceeds PAC-1 limits at the site boundary or the nearest residence, and no chemical consequence exceeds PAC-2 limits for the worker.

Chemical Process Safety Controls

The components credited for prevention of the chemical dose consequences are:

- URSS uranium storage racks are seismically qualified to maintain their structure and position during seismic events.

The components credited for mitigation of the chemical dose consequences are:

- Confinement barriers (i.e., supercell, gloveboxes, subgrade vaults) are credited for those chemical spill scenarios that occur within a confinement structure as identified in Table 13b.3-2.

**Table 13b.3-1 – ~~Inventory~~Quantities of ~~In-Process~~ Hazardous Chemicals
 (Sheet 1 of 3)**

Chemical Name	Location	Inventory (kg)	Storage-Container-Type-and-Capacity
Alpha-Benzoin Oxime	Chemical Storage <u>Subgrade Waste Tanks</u>	44.48E-01 <u>1.00E-02</u>	Two 250-gram (g) containers
	<u>Supercell</u>	<u>3.28E-03</u>	
	<u>RLWI Tank</u>		
Ammonia (28 wt.%) um Hydroxide	Chemical Storage <u>Supercell</u>	499 <u>2.17E-02</u>	(Two 1000-liter [I] intermediate bulk containers [IBC]) Liquid
Ammonium Nitrate	<u>Subgrade Waste Tanks</u>	<u>1.81E+01</u>	
	<u>Supercell</u>	<u>4.96E-02</u>	
	<u>RLWI Tanks</u>	<u>1.32E-01</u>	
[] ^{PROP/ECI}	Chemical Storage <u>TSPS Room</u>	406.20E-01 <u>2.79E+00</u>	Two 5 kilogram (kg) containers
	<u>Target Solution Storage</u>	<u>1.86E-01</u>	
	<u>Subgrade Waste Tanks</u>	<u>3.10E-01</u>	
	<u>Supercell</u>	<u>2.70E-03</u>	
	<u>RLWI Tank</u>		
Hydrochloric Acid (38 wt.%)	Chemical Storage <u>Subgrade Waste Tanks</u>	41.92E-02 <u>1.00E-02</u>	Two one-gallon containers
	<u>Supercell</u>	<u>1.41E-04</u>	
	<u>RLWI Tank</u>		
Hydrogen Peroxide (30 wt.%)	Chemical Storage <u>TSPS Room</u>	668 <u>2.66E+00</u>	Liquid (Two 1000 IBC)
	<u>Supercell</u>	<u>1.00E-02</u>	

**Table 13b.3-1 – ~~Inventory~~Quantities of ~~In-Process~~ Hazardous Chemicals
(Sheet 2 of 3)**

Chemical Name	Location	Inventory (kg)	Storage-Container-Type-and-Capacity
Mineral Oil	Chemical Storage	360	Liquid (Two 55-gallon drums)
Molybdenum Trioxide	Chemical Storage	4	Two one-gallon containers
Nitric Acid (70 wt.% in chemical storage)	Chemical Storage	1979	Liquid (Two 1000-HBC)
	<u>Subgrade Waste</u>	<u>1.79E+01</u>	
	<u>Tanks</u>	<u>9.49E-01</u>	
	<u>Supercell</u>	<u>1.31E-01</u>	
Potassium Hexachlororuthenate	Chemical Storage	47.47E-03	Two 250-g containers
	<u>Subgrade Waste</u>	<u>1.00E-02</u>	
	<u>Tanks</u>	<u>5.46E-05</u>	
	<u>Supercell</u>	<u>5.46E-05</u>	
Potassium Permanganate	Chemical Storage	44.73E-01	Two 250-g containers
	<u>Subgrade Waste</u>	<u>1.00E-02</u>	
	<u>Tanks</u>	<u>3.46E-03</u>	
	<u>Supercell</u>	<u>3.46E-03</u>	
Rhodium Chloride	Chemical Storage	48.96E-03	Two 250-g containers
	<u>Subgrade Waste</u>	<u>1.00E-02</u>	
	<u>Tanks</u>	<u>6.55E-05</u>	
	<u>Supercell</u>	<u>6.55E-05</u>	
Silver Nitrate	Chemical Storage <u>Supercell</u>	<u>1.00E-02</u>	Two 250-g containers
Sodium Hydroxide (50.5 wt.%)	Chemical Storage <u>Supercell</u>	4544 <u>5.17E-01</u>	Two 1000-HBC

**Table 13b.3-1 – ~~Inventory~~Quantities of ~~In-Process~~ Hazardous Chemicals
 (Sheet 3 of 3)**

Chemical Name	Location	Inventory (kg)	Storage-Container-Type-and-Capacity
Sodium Iodide	Chemical Storage Supercell	1.00E-02	Two 250-g bottles
Sodium Sulfite (98 wt. %)	Chemical Storage Subgrade Waste Tanks	23.12E+00	Two 250-g bottles
	Supercell	1.00E-01	
	RLWI Tank	2.28E-02	
Sulfuric Acid	Chemical Storage TSPS Room	3599.67E+00	Liquid (Two 1000-HBC)
	Target Solution Storage	5.46E+01	
	Subgrade Waste Tanks	5.62E+02	
	Supercell	6.07E+00	
	RLWI Tank	4.49E+00	
Sulfur Hexafluoride	Irradiation unit cells (8)	1440	Pressurized Gas (NDAS pressure vessel)
Uranium Metal	Uranium Storage URSS Room	6206.20E+02	Solid pieces (Storage container maximum size: 7.8 kg)
Uranium Oxide	Uranium Storage URSS Room	7327.31E+02	Solid powder (Storage container maximum size: 5.04 kg)
	TSPS Room	8.60E+00	
Uranyl Peroxide	Target Solution Preparation TSPS Room	431.15E+01	Intermediate compound in uranyl sulfate dissolution tanks
Uranyl Sulfate	In-Process TSPS Room	10581.91E+02	Liquid (Target solution prep tank maximum []^{PROP/ECL})
	Target Solution Storage	8.58E+02	
	Subgrade Waste Tanks	7.26E+01	
	Supercell	9.54E+01	
	RLWI Tank	1.05E+00	

**Table 13b.3-2 – Hazardous Chemical Source Terms and Concentration Levels
 (Sheet 1 of 3)**

Hazardous Chemical (Release Mechanism/Location)	MAR (kg)	Airborne Fraction	Source Term (kg)	PAC-1 ^(a) (l)	PAC-2 ^(a) (l)	PAC-3 ^(a) (l)	Control Room Operator/ RCA Worker Concentration (l)	Site Boundary Concentration (230 m) ^(a) (l)	Nearest Residence (788 m) ^(a) (l)
Alpha-Benzoin Oxime (Subgrade Waste Tanks)	0.250.0688	2E-3	5.00E-41.38	0.49- mg/m ³	5.4- mg/m ³	32- mg/m ³	0.0279- mg/m ³ 1.30E-03/ 7.68E-05	Negligible1.30E- 05	Negligible8.50E- 07
Ammonia (28-wt.%)um Hydroxide (Supercell)	249.50.1 ^(b)	1.0	249.52490	30-ppm13	160-ppm140	1100-ppm840	0.553- ppm1.53E+00/ 1.39E-01	27.9-ppm2.89E- 02	2.30-ppm1.89E- 03
Ammonium Nitrate (Subgrade Waste Tanks)	2.77		55.37	6.7	73	440	5.23E-02/3.09E-03	5.22E-04	3.42E-05
[] ^{PROP/ECL} (TSPS Room)	50.744	2E-3	0.010.76 ^(c) / 74.4 ^(c)	[] ^{PROP/ECL}	[] ^{PROP/ECL}	[] ^{PROP/ECL}	0.558- mg/m ³ 1.26E-02/ 3.15E-03	0.0706- mg/m ³ 3.57E-05	Negligible2.34E- 06
Hydrochloric Acid (38-wt.%)Supercell)	20.038 ^(b)	1.0	21380	1.8-ppm2.7	22-ppm33	100-ppm150	7.58E-3-ppm7.14E- 01/7.71E-02	0.155- ppm1.90E-02	0.0208- ppm1.24E-03
Hydrogen Peroxide (30-wt.%)TSPS Room)	334.03.2	1.0	334.01380	10-ppm14	50-ppm70	100-ppm140	4.05E-3-ppm1.69E- 01/7.71E-02	0.0367- ppm2.24E-03	Negligible1.47E- 04
Mineral Oil	180	1.0	180	140- mg/m ³	1500- mg/m ³	8000- mg/m ³	Negligible	Negligible	Negligible
Molybdenum Trioxide	4	2E-3	0.002	2.3- mg/m ³	43- mg/m ³	260- mg/m ³	0.112- mg/m ³	0.0141- mg/m ³	Negligible
Nitric Acid (70-wt.% in- chemical- storageSubgrade Waste Tank)	989.52.7 ^(d)	1.0	989.54820	0.16-ppm41	24-ppm62	92-ppm240	0.0157- ppm2.55E+00/ 2.69E-01	0.0769- ppm7.91E-03	0.00858- ppm5.19E-04
Potassium Hexachloro-ruthenate ^(b) (Supercell)	0.250.012	2E-3	5.00E-40.24	None- Identified0.5	None-Identified2	None- Identified20	N/A2.27E-04/1.34E- 05	N/A2.26E-06	N/A1.48E-07

**Table 13b.3-2 – Hazardous Chemical Source Terms and Concentration Levels
(Sheet 2 of 3)**

Hazardous Chemical (Release Mechanism/Location)	MAR (kg)	Airborne Fraction	Source Term (kg)	PAC-1 ^(a) (mg/m ³)	PAC-2 ^(a) (mg/m ³)	PAC-3 ^(a) (mg/m ³)	Control Room Operator/ RCA Worker Concentration (mg/m ³)	Site Boundary Concentration (230 m) ^(a) (mg/m ³)	Nearest Residence (788 m) ^(a) (mg/m ³)
Potassium Permanganate (Subgrade Waste Tanks)	0.250.0727	2E-3	5.00E-41.45	8.6 mg/m ³	14 mg/m ³	150 mg/m ³	0.0279- mg/m ³ 1.37E-03/ 8.12E-05	Negligible1.37E-05	Negligible8.99E-07
Rhodium Chloride ^{(e)(e)} (Supercell)	0.250.012	2E-3	5.00E-40.24	0.1 mg/m ³ 1.68	None-Identified18.5	None-Identified110	0.0279- mg/m ³ 2.27E-04/ 1.34E-05	Negligible2.26E-06	Negligible1.48E-07
Silver Nitrate (Supercell)	0.250.012	2E-3	5.00E-40.24	0.0475- mg/m ³	0.9 mg/m ³	5.4 mg/m ³	0.0279- mg/m ³ 2.27E-04/ 1.34E-05	Negligible2.26E-06	Negligible1.48E-07
Sodium Hydroxide (50.5 wt.% Supercell)	770.50.620	4E-7/hr	3.082E-4 kg/hr12.4	0.5 mg/m ³	5 mg/m ³	50 mg/m ³	0.00287- mg/m ³ 1.17E-02/ 6.93E-04	Negligible1.17E-04	Negligible7.67E-06
Sodium Iodide (Supercell)	0.250.012	2E-3	5.00E-40.24	13 mg/m ³	140 mg/m ³	860 mg/m ³	0.0279- mg/m ³ 2.27E-04/ 1.34E-05	Negligible2.26E-06	Negligible1.48E-07
Sodium Sulfite (Subgrade Waste Tanks)	0.250.478	2E-3	5.00E-49.55	11 mg/m ³	120 mg/m ³	710 mg/m ³	0.0279- mg/m ³ 9.02E-03/ 5.33E-04	Negligible9.01E-05	Negligible5.91E-06
Sulfuric Acid (Subgrade Waste Tanks)	1799.578.0	1.0	1799.51560	0.20 mg/m ³	8.7 mg/m ³	160 mg/m ³	4.48E-4 mg/m ³ 1.47E+00/ 8.71E-02	Negligible1.47E-02	Negligible9.65E-04
Sulfur Hexafluoride	1440	1.0	168.48	3000 ppm	33000 ppm	200000 ppm	9.41 ppm	3.31 ppm	0.368 ppm
Hydrogen Fluoride ^(d)	0.3105	1.0	0.3105	1 ppm	24 ppm	44 ppm	0.0173 ppm	0.0445 ppm	Negligible
Sulfur Dioxide ^(d)	1.45E-3	1.0	1.45E-3	0.2 ppm	0.75 ppm	30 ppm	8.10E-5 ppm	Negligible	Negligible
Uranium Metal ^{(e)(f)} (URSS Room)	7.8	0	Negligible0	0.6 mg/m ³	5 mg/m ³	30 mg/m ³	Negligible0.00E+00/ 0.00E+00	Negligible0.00E+00	Negligible0.00E+00
Uranium Oxide (URSS Room)	5.0440.0	2E-3	0.042400	0.74 mg/m ³ 68	10 mg/m ³	530 mg/m ³	0.558- mg/m ³ 2.27E+00/ 9.99E+00	0.0706- mg/m ³ 2.26E-02	Negligible1.48E-03

**Table 13b.3-2 – Hazardous Chemical Source Terms and Concentration Levels
(Sheet 3 of 3)**

Hazardous Chemical (Release Mechanism/Location)	MAR (kg)	Airborne Fraction	Source Term (kg)	PAC-1 ^(a) (mg/m ³)	PAC-2 ^(a) (mg/m ³)	PAC-3 ^(a) (mg/m ³)	Control Room Operator/RCA Worker Concentration (mg/m ³)	Site Boundary Concentration (230 m) ^(a) (mg/m ³)	Nearest Residence (788 m) ^(a) (mg/m ³)
Uranyl Peroxide ^{(f)(g)} (TSPS Room)	436.84	4E-7/hr	1.72E-5/1368	1.04E-3 mg/m ³ 0.94	None- Identified10.4	None- Identified62	1.60E-4 mg/m ³ 1.29E+00/ 5.70E+00	Negligible1.29E- 02	Negligible8.46E- 04
Uranyl Sulfate ^{(f)(g)} (TSPS Room)	1016.8191. 2	4E-7/hr	4.0672E-4 kg/ hr235 ^(c) / 19120 ^(c)	0.78- mg/m ³ 0.92	4.3- mg/m ³ 10.2	26- mg/m ³ 61	3.79E-3 mg/m ³ 3.91E+00/ 1.07E+00	Negligible1.11E- 02	Negligible7.25E- 04

- a. ~~Consequence is taken as the outside concentration calculated by ALOHA. Negligible values are those determined by the software to not have significant concentration at the point selected. Protective Action Criteria (PAC) values are based on the U.S. Department of Energy's Protective Action Criteria Database (USDOE, 2018), unless otherwise specified.~~
- b. ~~Based on the absence of PAC levels and significant health effects, further evaluation of potassium hexachlororuthenate was not required. PAC values were not identified for potassium hexachlororuthenate in the PAC Database (USDOE, 2018a). PAC values for chlorine are provided, as chlorine has the most restrictive PAC values of the compound's constituents. MAR increased to the minimum mass that ALOHA can model for a puddle release.~~
- c. ~~PAC limits were not identified for rhodium chloride. An acceptance value of 0.1 mg/m³ is assigned based on acceptable OSHA occupational exposure limits from the MSDS for uranyl rhodium chloride. The first source term value listed is for a two-minute release, while the second source term value corresponds to a full tank release. For each receptor, the source term value which yields the most conservative result is used.~~
- d. ~~Hydrogen fluoride and sulfur dioxide are chemical byproducts and are not normally stored on site. Based on largest capacity subgrade waste tank.~~
- e. ~~Uranium metal is stored as solid pieces. Therefore, there is no hazard from dropping solid metal pieces. PAC values were not identified for rhodium chloride in the PAC Database (USDOE, 2018). PAC values were developed from toxicity information found on the safety data sheet using the methodology from DOE-HDBK-1046-2016 (USDOE, 2016).~~
- f. ~~PAC limits were not identified for uranyl peroxide. An acceptance value of 1.04E-3 mg/m³ is assigned based on acceptable OSHA occupational exposure limits from the MSDS for uranyl peroxide. Uranium metal is stored as solid pieces. Therefore, there is no hazard from dropping solid metal pieces.~~
- g. ~~PAC limits were not identified for uranyl sulfate; therefore, values for a similar material uranyl fluoride are used. This is acceptance since soluble uranium compounds pose similar health hazards. PAC values were not identified for uranyl peroxide or uranyl sulfate in the PAC Database (USDOE, 2018). For uranium compounds, ACGIH STEL is 0.6 mg/m³, which is multiplied by a compound adjustment factor based on the methodology from DOE-HDBK-1046-2016 (USDOE, 2016) to obtain the TEEL-1 (PAC-1) value. PAC-2 and PAC-3 values were calculated based on the methodology from DOE-HDBK-1046-2016.~~

13b.4 REFERENCES

[USDOE, 2016. Temporary Emergency Exposure Limits for Chemicals: Methods and Practice, DOE-HDBK-1046-2016, U.S. Department of Energy, December 2016.](#)

[USDOE, 2018. Chemicals of Concern and Associated Chemical Information, Protective Action Criteria \(PAC\) Tables Rev. 29a, U.S. Department of Energy, June 2018.](#)

[USNRC, 1982. PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations, NUREG/CR-2858, U.S. Nuclear Regulatory Commission, November 1982.](#)

[USNRC, 1998. Nuclear Fuel Cycle Facility Accident Analysis Handbook, NUREG/CR-6410, U.S. Nuclear Regulatory Commission, March 1998.](#)

USNRC, 2012. Final Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors, U.S. Nuclear Regulatory Commission, October 17, 2012.

~~**USNRC, 1998.** Nuclear Fuel Cycle Facility Accident Analysis Handbook, NUREG/CR-6410, U.S. Nuclear Regulatory Commission, March 1998.~~

~~**USDOC, 2013.** Areal Locations of Hazardous Atmospheres (ALOHA) Technical Documentation, NOAA Technical Memorandum NOS-OR&R-43, U.S. Department of Commerce, November 2013.~~

~~**USDOE, 2018.** Chemicals of Concern and Associated Chemical Information, Protective Action Criteria (PAC) Tables Rev. 29a, U.S. Department of Energy, June 2018.~~

**ENCLOSURE 3
ATTACHMENT 2**

SHINE MEDICAL TECHNOLOGIES, LLC

**SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION
SUPPLEMENT NO. 5 AND RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

**TECHNICAL SPECIFICATIONS CHANGES
PUBLIC VERSION
(MARK-UP)**

- e. The radiation protection plan: at least once per calendar year (interval between audits not to exceed 15 months);
 - f. The QAPD: at least once every other calendar year (interval between audits not to exceed 30 months); ~~and~~
 - ~~g.~~ g. The physical security plan: at least once every other calendar year (interval between audits not to exceed 30 months); and
 - ~~g.h.~~ g.h. The nuclear criticality safety program: at least once every third calendar year (interval between audits not to exceed 36 months).
2. Deficiencies identified during the audit will be entered into the Corrective Action Program. Deficiencies uncovered that affect nuclear safety shall immediately be reported to Level 1 management. A written report of the findings of the audit shall be submitted to Level 1 management and the review and audit committee members within three months after the audit has been completed.

5.3 Radiation Safety

The RPM shall be responsible for the implementation of the radiation protection program. The requirements of the radiation protection program are established by 10 CFR Part 20. The program shall use the guidelines of ANSI/ANS 15.11-1993, Radiation Protection at Research Reactor Facilities. Furthermore, SHINE is committed to ensuring that radiation exposures are ALARA and in maintaining and effective ALARA Program.

The radiation protection department is independent of facility operations. This independence ensures that the radiation protection department maintains its objectivity and is focused only on implementing sound radiation protection principals necessary to achieve occupational doses and doses to members of the public that are ALARA.

Radiation protection staff maintain the ability to raise safety issues with the review and audit committee or executive management.

5.4 Procedures

1. Procedures for the operation and use of the SHINE Facility provide appropriate direction to ensure that the facility is operated normally within its design basis, and in compliance with technical specifications. These procedures are written, reviewed, approved by appropriate management, as well as controlled and monitored to ensure that the content is technically correct, and the wording and format are clear and concise.
2. The process required to make changes to procedures, including substantive and minor permanent changes, and temporary deviations to accommodate special or unusual circumstances during operation shall be in compliance with ANSI/ANS 15.1-2007.
3. SHINE shall prepare, review, and approve written procedures for the following basic topics:
 - a. startup, operation, and shutdown of the IU;
 - b. target solution fill, draining, and movement within the SHINE Facility;

5.5 Programs

The following programs shall be established, implemented, and maintained.

5.5.1 Nuclear Safety Program

The SHINE nuclear safety program documents and describes the methods used to minimize the probability and consequences of accidents resulting in radiological or chemical release. The program applies a graded approach to the design and management of processes to assure plant safety through risk reduction and satisfaction of SHINE's performance goals. The safety program accomplishes these goals through development and maintenance of the accident analysis, identification of safety-related structures, systems, and components (SSCs) credited for accident mitigation, and establishment of programmatic administrative controls to ensure reliability of the credited SSCs.

5.5.2 Training and Qualification

The SHINE training and qualification programs include initial and requalification training programs for Licensed Operators, which were developed to conform to the requirements of 10 CFR Part 55, as it pertains to non-power facilities, following the guidance contained in ANSI/ANS 15.4-2016, Selection and Training of Personnel for Research Reactors.

5.5.3 Radiation Protection

The SHINE radiation protection program is provided to protect the radiological health and safety of workers and the public. The program meets the requirements of 10 CFR 20, Subpart B, and is consistent with the guidance provided in ANSI/ANS 15.11-2016, Radiation Protection at Research Reactor Facilities, and Regulatory Guide 8.2, Revision 1, Administrative Practices in Radiation Surveys and Monitoring. In addition, SHINE has established this program to maintain occupational radiation exposures and releases to the environment ALARA.

5.5.4 Configuration Management

The SHINE configuration management program provides oversight and control of design information, safety information, and records of modifications that might impact the ability of safety-related SSCs to perform their functions. The configuration management program is applied to all safety-related SSCs [and is used to evaluate each change to the SHINE Facility for the potential to affect safety-related SSCs.](#) The configuration management program is used to maintain consistency among the design requirements, the physical configuration and the facility documentation, and ensures changes are made in accordance with 10 CFR 50.59, 10 CFR 70.72, and the administrative controls and reviews specified by this program.

Table 5.5.4 lists controls derived from the accident analysis not otherwise included in Sections 3, 4, or 5 of the technical specifications. SHINE maintains these controls under the configuration management program and will not modify the characteristics of the items listed in Table 5.5.4 without prior NRC approval.

Category	Characteristic
Irradiation Unit (continued)	The fill rate of target solution into the TSV is not more than [] ^{PROP/ECI} . This rate is set and verified during Startup Testing using a throttle valve that is subsequently locked in place.
	The overflow lines in the TSV are located at a height sufficient to protect the TSV headspace to allow proper operation of the TOGS.
	The TSV dump tank is designed with $k_{eff} < 0.9594$ for the most reactive uranium concentration for the prevention of criticality.
	The TOGS physical design ensures k_{eff} is < 0.9594 for the most reactive uranium concentration for the prevention of criticality.
	[] ^{PROP/ECI}
Coolant Systems	The PCLS expansion tank is provided with a flame arrester at the tank vent inlet to prevent ignition of hydrogen accumulated in the expansion tank from ignition sources in the primary Confinement boundary.
Isotope Production Systems	The TSSS process pipes are seismically qualified.
	The MEPS [] ^{PROP/ECI} is seismically qualified.
	The design of the MEPS upper three-way valve prevents reverse flow from the target solution return header to the eluate tank.
	The MEPS extraction feed pump discharge lines have overpressure protection.
	Vaults, trenches, valve pits, and hot cells where high concentration uranium-bearing solutions may be present are equipped with drains to the favorable geometry RDS sump tanks. The RDS drain in the supercell additionally provides over pressure protection via a relief path to the RDS.
Confinement	Confinement boundaries within the facility are provided to limit the release of effluents from the enclosure to the external environment through controlled or defined pathways.
	The holdup volume in RVZ1e from the PCLS expansion tank, between radiation detectors and isolation devices, provides a time delay for isolation of IU cell gases exiting the confinement boundary.

5.8.3 Additional Event Reporting Requirements

Events which meet the reporting requirements of 10 CFR 70.50; 10 CFR 70.52; or paragraphs (a)(1) through (a)(3), (b)(3), or (c) of 10 CFR Part 70, Appendix A shall be reported to the NRC as prescribed in the applicable regulation.

~~5.8.3~~ 5.8.4 Startup Report

SHINE shall conduct startup testing in accordance with the Startup Testing Program, as described in FSAR Section 12.11. Following completion of startup testing, SHINE will submit a Startup Report to the NRC Document Control Desk that identifies the startup tests performed.

The Startup Report shall be submitted within 6 months of the completion of all startup testing activities.

5.9 Records

5.9.1 Lifetime Records

The following records are to be retained for the lifetime of the SHINE Facility:

1. Gaseous and liquid radioactive effluents released to the environs;
2. Offsite environment-monitoring surveys required by the technical specifications;
3. Radiation exposure for all monitored personnel; and
4. Updated drawings of the SHINE Facility.

Applicable annual reports, if they contain all of the required information, may be used as records in this section.

5.9.2 Five Year Records

The following records are to be maintained for a period of at least five years or for the life of the component involved if less than five years:

1. Normal SHINE Facility operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);
2. Principal maintenance operations;
3. Reportable occurrences;
4. Surveillance activities required by the technical specifications;
5. Facility radiation and contamination surveys where required by applicable regulations;
6. Radioactive material inventories, receipts, and shipments;
7. Approved changes in operating procedures; and
8. Records of meeting and audit reports of the review and audit group.

5.9.3 Records to be retained for at least one certification cycle

Records of retraining and requalification of operations personnel who are Licensed pursuant to 10 CFR Part 55 shall be maintained at all times while the individual is employed or until License is renewed.