CHAPTER 13

ACCIDENT ANALYSIS

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ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
AC	alternating current
AECL	Atomic Energy of Canada Limited
AEGLs	Acute Exposure Guideline Levels
ALOHA	Areal Locations of Hazardous Atmospheres
ARF	airborne release fraction
BENM	Best Estimate Neutronics Model
CAMS	continuous airborne monitoring system
DBA	design basis accident
DBE	design basis earthquake
DCF	dose conversion factors
DR	damage ratio
ERPGs	Emergency Response Planning Guidelines
ESFAS	engineered safety features actuation system
FCRS	facility chemical reagent system
FHA	fire hazards analysis

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
FMEA	failure modes and effects analyses
ft.	feet
gal	gallons
gpm	gallons per minute
HAZOP	hazard and operability
HVPS	high voltage power supply
IBC	intermediate bulk containers
IE	initiating event
IF	irradiation facility
IMOD	Iodine Model for Containment Codes
ISG	interim staff guidance
IU	irradiation unit
IXP	iodine and xenon purification and packaging
k_{eff}	effective neutron multiplication factor
kW	kilowatt

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
L (l)	liter
L/s (l/s)	liters per second
LABS	quality control and analytical laboratories
LFL	lower flammability limit
LIRIC	Library of Iodine Reactions in Containment
LOOP	loss of off-site power
LPF	leak path factor
LWPS	light water pool system
m/s	meters per second
MAR	material at risk
MCC	motor control center
MCNP	Monte Carlo N-Particle Transport Code
MEPS	molybdenum extraction and purification system
MIPS	molybdenum isotope packaging system
MHA	maximum hypothetical accident
Mo-99	molybdenum-99

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
N2PS	nitrogen purge system
NCS	nuclear criticality safety
NDAS	neutron driver assembly system
NPSS	normal electrical power supply system
NSC	NDAS service cell
OSHA	Occupational Safety and Health Administration
Pa	Pascal
PAC	Protective Action Criteria
PCHS	process chilled water system
PCLS	primary closed loop cooling system
pcm	percent millirho
PHA	process hazard analysis
PICS	process integrated control system
PSB	primary system boundary

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PVVS	process vessel vent system
RCA	radiologically controlled area
RDS	radioactive drain system
RF	respiratory fraction
RLWI	radioactive liquid waste immobilization
RLWS	radioactive liquid waste storage
RPCS	radioisotope process facility cooling system
RPF	radioisotope production facility
RVZ1	radiological ventilation zone 1
RVZ1e	radiological ventilation zone 1 exhaust subsystem
RVZ1r	radiological ventilation zone 1 recirculation subsystem
RVZ2	radiological ventilation zone 2
RVZ2r	radiological ventilation zone 2 recirculation subsystem

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	**Definition**
SASS	subcritical assembly support structure
SCAS	subcritical assembly system
SF_6	sulfur hexafluoride
SGS	standby generator system
SHINE	SHINE Medical Technologies
SNM	special nuclear material
SSC	system, structure, and component
TDE	total dose equivalent
TEDE	total effective dose equivalent
TEELs	Temporary Emergency Exposure Limits
TOGS	TSV off-gas system
TPS	tritium purification system
TRPS	TSV reactivity protection system
TSPS	target solution preparation system
TSSS	target solution staging system
TSV	target solution vessel

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
UPSS	uninterruptible electrical power supply system
URSS	uranium receipt and storage system
VAC/ITS	vacuum/impurity treatment subsystem
VTS	vacuum transfer system
χ/Q	atmospheric dispersion factor

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 (RPF).

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

- NUREG-1537 (USNRC, 1996) and the Interim Staff Guidance (ISG) Augmenting NUREG-1537 (USNRC, 2012a);
- Process hazard analysis (PHA) 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.

SHINE Safety Analysis (SSA) Methodology

SHINE applies a SHINE-specific, risk-based methodology similar to the guidance described in NUREG-1520, Standard Review Plan for Fuel Cycle Facilities License Applications (USNRC, 2015) in the development of the detailed accident analysis. This methodology is applied to both the IF and the RPF for consistency of the safety analysis for the entire SHINE facility.

The SSA is a systematic analysis of facility processes used to identify facility hazards associated with the processing and possession of licensed materials. The SSA has been performed for the purpose of identifying relevant hazards, potential accident sequences and consequences, equipment and specific human actions credited for safety, and programmatic administrative controls necessary to ensure the availability and reliability of safety-related SSCs. This analysis takes into consideration the facility structure, equipment, activities, personnel, processes, and administrative controls in an integrated manner to identify and analyze hazards.

Applicability

Normal operation at the SHINE facility includes IF operations as well as chemical extraction and purification operations, target solution preparation and storage activities, and waste handling and immobilization activities in the RPF.

The SSA considers all modes of operation for potential process upsets and accident sequences. The subcritical assembly system (SCAS) for each IU is analyzed for each mode of operation (i.e., Solution Removed, Startup, Irradiation, Post-Irradiation, Transfer to RPF). The associated target solution vessel (TSV) off-gas system (TOGS) operation is combined with the SCAS analysis as they are tightly coupled systems. Since the tritium purification system (TPS) services all eight IUs, it is analyzed as a continuously operating integrated system. The operating modes for the TPS include normal gas feed, recovery and purification, and TPS glovebox cleanup.

SHINE systems which operate in a batch mode are analyzed for active operation while hazardous materials are present. The molybdenum extraction and purification system (MEPS) and the iodine and xenon purification and packaging (IXP) system are either in use or idle. They are therefore analyzed for normal extraction, purification, and packaging activities. Similarly, the target solution and preparation system (TSPS) and the uranium receipt and storage system (URSS) are analyzed for normal target solution preparation activities. The radioactive liquid waste system (RLWS) and the radioactive liquid waste immobilization (RLWI) system are also analyzed for normal storage and processing of liquid wastes.

The SSA considers maintenance activities as potential initiators for accident sequences including maintenance errors, improper system restoration, impacts on operating equipment, and fires. These types of initiators were identified during the accident sequence development phase of the SSA.

Non-routine activities may include the repair or replacement of major components such as the neutron drivers or the high voltage power supplies (HVPS). Accident sequences considered in the SSA include heavy load drops on systems or components containing radiological material and inadvertent exposure to neutrons.

Periods of extended shutdown are not explicitly identified as a class of accident sequences in the SSA; however, SHINE systems are designed to achieve and maintain a safe condition for radiological materials in extended storage.

Technical specifications require limiting conditions for operation (LCO) be met during the specified conditions of applicability. When an LCO is not met, the applicable actions specified in the technical specifications are required to be completed.

General Approach

The SSA is developed based on the following major steps:

- Identification and systematic evaluation of hazards at the facility
- Comprehensive identification of potential accident/event sequences that would result in unacceptable consequences, and the expected likelihoods of those sequences
- Assessment of radiological and chemical consequences for postulated accident sequences to demonstrate compliance with acceptable limits
- Identification and description of safety-related controls (i.e., structures, systems, equipment, components, or specific actions) that are relied on to limit or prevent potential accidents or mitigate their consequences
- Identification of programmatic administrative controls that ensure the availability and reliability of identified safety systems.

The results of the SSA consist of postulated accident sequences for inclusion in this chapter. This includes a description of the accident sequences, potential consequences, controls credited to prevent or mitigate the accident sequence, and a summary of calculated dose consequences.

Hazard Identification and Evaluation

Hazard identification is performed by identifying, for each process, radiological or chemical hazards that have the potential for causing harm to the public, facility staff, or the environment. This includes physical process hazards (e.g., deflagration, fire, flooding) that could result in adverse effects on licensed materials. Radiological hazards include radiation sources from the SHINE processes (e.g., neutron driver, TSV), fission products, activation products, and tritium. Fissile material hazards are also considered for postulated criticality accidents. Chemical hazards are identified that could affect licensed materials or the safe operation of the facility. Chemical effects considered include flammable, reactive, oxidation, and chemical incompatibility effects.

The hazard identification and evaluation is performed using standard hazard evaluation methods such as Hazard and Operability Analysis (HAZOP) and Failure Modes and Effects Analysis (FMEA). The types of hazards identified for the SHINE facility are identified in Table 13a2.1-1.

Hazard evaluations are conducted to assess potential failures, causes, and consequences that provide a basis for the development of potential accident sequences. The output of the hazard evaluations are those failure-cause-consequences that have the potential for causing harm to the public, facility staff, or the environment and the possible engineered or administrative controls that may be applied for prevention or mitigation.

Process Hazard Analysis and Accident Sequence Development

The results of the hazard evaluations are used to inform the PHA and accident sequence development phase. The PHA uses the results of the hazard evaluations to develop accident sequences in alignment with the accident sequence categories described in Section 13a2.1 for the IF and Subsection 13b.1.2 for the RPF.

Accident sequence development uses the risk index methodology based on risk index values described in NUREG-1520 (USNRC, 2015). Potential accident sequences are defined based on the failures, process deviations, or external events as identified in the hazard evaluations. An initiating event is defined for each scenario that may include equipment failures, human errors, external events, or combinations of these elements.

External event-induced accident sequences are treated on a site-wide basis. The external events PHA also includes fires and flooding from causes internal to the IF and the RPF. External event initiating events that are considered include:

- external events such as earthquake, external flooding, external fires, high winds, and tornadoes;
- events that are external to the process being analyzed such as internal fires and internal flooding;
- deviations from normal process operations (credible abnormal events);
- failures of process components; and
- human errors that result in process upsets or failures.

Potential consequences are also identified for each accident sequence as one or more of the following:

- Radiological dose to the public or facility staff (i.e., control room operator)
- Chemical dose to the public or facility staff (i.e., control room operator and radiologically controlled area [RCA] worker)
- Criticality event
- No consequence of concern

The radiological consequence analysis is described in Section 13a2.2 for the IF and Section 13b.2 for the RPF. The chemical consequence analysis is described in Section 13b.3.

Accident sequences that may result in a consequence of concern are first evaluated with no engineered or administrative controls applied, referred to as an “uncontrolled accident sequence.” A total risk index number is determined based on an estimate for the likelihood of occurrence and severity of consequences. For accident sequences with unacceptable risk indices, engineered and administrative controls are applied that reduce the likelihood of occurrence and/or the severity of the consequences such that an acceptable risk level is reached. Acceptable risk levels for SHINE require that the postulated sequence is “highly unlikely” and/or the consequence severity is “low.” The final accident sequence is referred to as a “controlled accident sequence.” The credited engineered and administrative controls are identified as safety-related controls.

Risk Matrix Development

The risk matrix applied in the SSA is provided in Table 13a2.1-2. The risk matrix approach provides a method of determining the risk of various accident sequences based on a quantitative estimate of the likelihood of occurrence and the severity of the consequences. The likelihood of occurrence and the consequence severity for each uncontrolled accident sequence is estimated and corresponding categories are assigned. The risk matrix then identifies those credible accidents which have the potential to exceed the acceptable risk index values, and therefore require engineered and/or administrative controls for prevention or mitigation. The risk index values are then reassessed after application of engineered or administrative controls that result in an acceptable risk outcome.

Likelihood Evaluation

For accident sequences in categories other than inadvertent criticality, the likelihood category definitions applied in the SSA are provided in Table 13a2.1-3. The determination of the likelihood of occurrence consists of the initiating event frequency (e.g., seismic event, process component failure, human error) and may be combined with an additional component failure or human error, including any recovery times (i.e., failure duration). In most cases, the initiating events are represented by single events or single failures.

The frequency of occurrence of an initiating event for an accident sequence is represented by a failure frequency index number (FFIN). The FFINs applied in the SSA are provided in Table 13a2.1-4. The bases for determining the FFIN for an accident sequence may include evidence or the type of control.

To determine the FFIN selected for an accident sequence initiator based on the type of control, several factors are considered including:

- administrative (i.e., human error);
- type of component failure (i.e., active versus passive);
- degree of redundancy (i.e., single component, redundant component);
- design margin (e.g., design pressure versus nominal pressure); and
- other factors including degree of enhancement for administrative controls (e.g., independent verification and step sign-off).

If the accident sequence is postulated to occur only if another condition or failure is present, an additional probability of component failure or condition is included in the evaluation. The failure probability index number (FPIN) represents this as a failure on demand, or as a probability that the condition exists. This can be evaluated as a simple probability of failure on demand or approximated as the product of a failure rate and a recovery time, defined in this analysis as a duration index number (DIN). The quantitative characterization of the FPIN and DIN applied in the SSA is provided in Table 13a2.1-5 and Table 13a2.1-6, respectively.

For inadvertent criticality accident sequences, the likelihood category definitions provided in Table 13a2.1-3 are not used. Each potential criticality accident sequence is assumed to have a high consequence and controls are determined through process analysis of the fissionable material operation and application of the double contingency principle to ensure that the entire process will be subcritical under both normal and credible abnormal conditions.

Sufficient controls are identified such that at least two unlikely, independent, and concurrent changes in process conditions are required before a criticality accident is possible, as required by the double contingency principle. Qualitative analysis, based on the interpretation and expert judgment of the nuclear criticality safety staff and key stakeholders during the evaluation process, is used to determine whether changes in process conditions are unlikely and independent.

Consequence Category Definitions

The consequence category definitions applied in the SSA are provided in Table 13a2.1-7. Numerical limits for the radiological and chemical exposure effects are included in the definitions for high and intermediate consequence for the public and facility staff. The low consequence category is implicitly defined as resulting in consequences that are less than intermediate and meet the SHINE safety criteria limits defined in Section 3.1.

Safety-Related Controls

The accident sequences developed in the PHA phase identify the controls that are credited for prevention and/or mitigation of accident sequences. The types of safety-related controls that are credited for prevention and/or mitigation of accident sequences are:

- Engineered controls (active or passive), identified as safety-related SSCs; and
- Specific administrative controls (e.g., procedural controls)

Safety-related controls that are credited for prevention and/or mitigation are identified for each accident scenario in Section 13a2.2 and Section 13b.2.

Programmatic administrative controls are also implemented to assure that safety-related controls can perform their intended functions. Defense-in-depth (DID) controls may also be identified that are not credited in accident sequences but provide additional margin for risk reduction.

Incorporation into the FSAR and Technical Specifications

Accident sequences developed in the SSA inform the accident analysis and determination of consequences of the limiting accident scenarios described in Section 13a2.2 for the IF and Section 13b.2 for the RPF.

The safety-related SSCs that are required to be operable to meet the assumptions underlying the SSA are included within Section 3.0 of the technical specifications, Limiting Conditions for Operation and Surveillance Requirements.

Section 4.0 of the technical specifications, Design Features, includes design features that are identified in the SSA. These are aspects of the facility design and other physical conditions (e.g., distance to the site boundary, building free volume) that are inputs or assumptions in the radiological dose calculations that support the SSA dose consequence analysis.

The SSA also identifies the programmatic administrative controls that are required to be implemented to ensure that safety-related SSCs will be capable of performing their intended functions. Section 5.0 of the technical specifications, Administrative Controls, includes the programmatic administrative controls identified in the SSA (e.g., maintenance of safety-related SSCs) and requires that those programs are established, implemented, and maintained. Section 5.0 additionally requires the development and use of procedures that implement the specific administrative controls identified in the SSA. Section 5.0 also includes discussion of the configuration management program, which 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 intended functions. The configuration management program also lists SSA-identified controls not otherwise included in Sections 3.0, 4.0, or 5.0 of the technical specifications that will be maintained under the configuration management program and will not be modified as described in the technical specifications without prior NRC approval.

13a2.1 ACCIDENT-INITIATING EVENTS AND SCENARIOS

The 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 maximum hypothetical accident (MHA) is also defined to result in the bounding radiological consequences for the SHINE facility.

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

- MHA (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 (PSB) (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 PHA process and are therefore considered within each event category.

13a2.1.1 MAXIMUM HYPOTHETICAL ACCIDENT

The guidance in NUREG-1537 (USNRC, 1996) 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 TSV 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.

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 PSB by causing increased power density, temperature, and pressure.

The 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 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 [ ]$^{\mathrm{PROP/ECI}}$ 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 [ ]$^{\mathrm{PROP/ECI}}$ 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.

<u>Scenario 1 – Increase in the Target Solution Density During Operations</u>

The TOGS regulates the pressure in the PSB. During irradiation operations, pressurization of the target solution could occur if there is a malfunction in TOGS.

A system pressurization could also occur following a deflagration or detonation in the PSB due to hydrogen accumulation during or following irradiation operations. The causes of this event are described in Subsection 13a2.1.9. Related reactivity effects are considered in this section.

Increased pressure in the TSV would cause the target solution to be compressed as void space decreases. This would cause an increase in reactivity in the SHINE system. With a fixed neutron source, the reactivity increase during irradiation operations leads to a power increase.

Pressure transients are described in Subsection 4a2.6.1.4 and limiting pressure transient analysis results are discussed in Subsection 4a2.6.3. Peak power during the pressure transients is calculated as less than 240 kW.

If pressurization is sustained, the event is terminated by the TRPS high time-averaged neutron flux trip, which de-energizes the neutron driver and opens the TSV dump valves. This trip results in a rapid reduction in the power generation in the TSV due to the loss of the neutron source, followed by the reactivity decrease from draining the target solution. The high time-averaged neutron flux trip prevents damage to the PSB. No damage to the PSB occurs and there are no radiological consequences.

Scenario 2 – Target Solution Temperature Reduction During Fill/Startup

The TSV is cooled by the PCLS. The PCLS is a closed loop that circulates cooling water [ ]PROP/ECI past the TSV walls to remove heat generated in the target solution during normal irradiation. The light water pool provides passive cooling of the TSV dump tank to remove heat generated during shutdown operations. The light water pool contains no dedicated cooling, and is cooled by contact with the PCLS-cooled components.

An excessive cooldown could occur if the PCLS malfunctions and overcools the target solution in the TSV, adding positive reactivity due to the negative temperature coefficient. An overcooling event is prevented by the TRPS trip on low PCLS temperature.

During fill/startup, the limiting scenario occurs when the TSV has been filled in Mode 1 to normal startup k_{eff} values. Then, the system is transitioned to Mode 2, and prior to accelerator operations, a failure of the PCLS occurs resulting in temperature decreasing in the TSV. The PCLS temperature decreases from 77°F to 59°F (25°C to 15°C) and results in a maximum reactivity insertion of [ ]PROP/ECI. This corresponds to a minimum volume of 4.2 percent, which is less than the minimum volume margin to critical used during fill. The event is discussed in Subsection 4a2.6.3. The reactivity increase is small and the system remains subcritical. No damage to the PSB occurs and there are no radiological consequences.

Greater temperature changes are prevented by the TRPS IU Cell Safety Actuation on high and low PCLS temperatures.

Scenario 3 – Target Solution Temperature Reduction During Irradiation

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 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, [ ]$^{\text{PROP/ECI}}$. 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 HVPS redundant breakers on a driver dropout signal after [ ]$^{\text{PROP/ECI}}$ 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 [ ]$^{\text{PROP/ECI}}$ from loss of void and up to [ ]$^{\text{PROP/ECI}}$ in [ ]$^{\text{PROP/ECI}}$ 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 resulting peak power density and return to equilibrium from the event is described in Subsection 4a2.6.3.3.

The TRPS Driver Dropout signal safely prevents power generation levels that would exceed target solution operating limits or damage the PSB.

Further analysis is provided in Subsection 13a2.2.2.

Scenario 5 – Moderator Addition Due to Cooling System Malfunction

The PCLS is a closed loop that circulates cooling water [ ]$^{\text{PROP/ECI}}$ past the TSV and neutron multiplier walls to remove heat generated in the TSV and neutron multiplier during normal irradiation and shutdown operations. If there were a breach between the TSV and PCLS or light water pool, cooling water could be added to the target solution. Moderator addition could also occur due to failure of a TOGS condenser demister unit or recombiner condenser unit, leading to radioisotope process facility cooling system (RPCS) water ingress into the TSV.

Water ingress into the TSV dilutes the target solution. A dilution event such as this would lower the overall reactivity of the target solution due to the high hydrogen to uranium ratio in the target solution (target solution is over-moderated).

If the break were to occur near the surface of the target solution or in TOGS, it is possible that the water could fill the upper space of the TSV between the solution level and the overflow lines. This would create a reflector. The maximum potential reactivity effect of a reflector forming in this manner has been evaluated assuming no mixing. In Mode 1, the limiting event occurs after the TSV has already been filled to normal startup k_{eff} values. The reactivity insertion is not significant enough to drive the system to criticality. Excess neutron flux levels during Mode 1 are prevented by the TRPS IU Cell Safety Actuation on high source range flux, resulting in a dump of the target solution to the TSV dump tank.

In Mode 2, the reactivity increase is prevented from resulting in excessive power generation by the IU Cell Safety Actuation on high time-averaged neutron flux. In a TSV overflow condition, excess water and target solution drain to the TSV dump tank via overflow lines. The TRPS initiates an IU Cell Safety Actuation and IU Cell Nitrogen Purge on dump tank low-high level in Mode 2. The water ingress could affect the proper functioning of TOGS by flooding sweep gas flow paths, but the nitrogen purge ensures that hydrogen gas concentrations remain within acceptable limits. No damage to the PSB occurs and there are no radiological consequences.

<u>Scenario 6 – Additional Target Solution Injection during Fill/Startup and Irradiation Operations</u>

During Mode 2 operations, target solution injection from the target solution hold tank is not credible due to its isolation from the TSV using redundant isolation valves, and the fact that the TSV is located higher than the target solution hold tank, thus preventing an accidental gravity-driven transfer of target solution to the TSV during operation. Target solution in the TSV fill lift tank is drained back to the hold tank following the fill process. No damage to the PSB occurs and there are no radiological consequences.

During fill/startup operations in Mode 1, excess fissile material is prevented from being added by several controls. These controls are described in two primary groups: (1) physical design and chemistry controls that prevent excess fissile material in the TSV, and (2) prevention of operator errors during the fill process that lead to excess fissile material. The limiting scenario is described following the controls.

Physical Design and Chemistry Controls

The first control is in the preparation of the target solution itself, where uranium enrichment is independently verified by SHINE and concentration in the target solution is controlled to within required accuracy levels. These controls ensure that the fissile material per volume of target solution is prepared within design calculation parameters.

The uranium concentration of target solution is verified within acceptable range after preparation of a new batch and after making adjustments to an existing batch, prior to transferring the batch to the TSV. No mechanisms, other than target solution adjustment, have been identified that would change fissile material per volume of target solution outside the bounds of what has been determined to be safe.

The physical placement of the TSV above the target solution hold tank and TSV dump tank prevents inadvertent draining of fissile material from these tanks to the TSV during the fill/startup operations.

Another control is the inherent limitations on fill rate. This limitation is due to the limited gravity-driven head of the fill process combined with the high hydraulic resistance of the fill path.

Prevention of Operator Errors During the Fill Process

During the fill process, the operators use fill procedures following the 1/M measurement method and containing hold points at certain volume levels to verify expected system behavior. The fill procedures limit the size of the solution addition steps the operators can use to one-half of the volume to predicted critical. This reduces fill increments as k_{eff} increases until the desired subcritical multiplication is reached. These procedural controls are fundamentally similar to reactor startup processes that routinely and safely start up reactors.

In addition to the procedural controls, the TRPS stops inadvertent target solution injection during the fill upon detection of high source range count rates in Mode 1. The TRPS initiates an IU Cell Safety Actuation to close the target solution fill valves and opens the TSV dump valves upon detection of high count rates.

Although not credited, a manual trip by the operators also causes the TSV solution to transfer to the TSV dump tank should an unsafe condition arise.

Furthermore, fill valve sequence controls in TRPS ensure proper neutron flux stabilization occurs between solution addition steps. Above pre-determined neutron flux levels in Mode 1, the TRPS limits the time the fill valve can be open so that rate of target solution addition is controlled. This nonsafety-related DID control allows the delayed neutron precursor population time to reach steady state. Subsection 7.4.4.1 provides a detailed description of the TRPS Fill Stop function.

The analyzed event is an inadvertent target solution injection after the system has already been filled to normal startup k_{eff} values. This could be caused by operator errors or malfunction of the process integrated control system (PICS). An injection of target solution occurs at the maximum rate allowed by the fill valve sequencing. The TRPS trips the system on high source range flux, draining the target solution faster than it can be filled. Reactivity is rapidly decreased in the SCAS, terminating the event.

Transient analysis of this event is presented in Subsection 4a2.6.3. No damage to the PSB occurs and there are no radiological consequences.

A solution addition event not crediting the operation of the TRPS Fill Stop function has also been analyzed. The resulting power increase is small. No damage to the PSB occurs and there are no radiological consequences.

Scenario 7 – Realistic, Adverse Geometry Changes

Because of the liquid nature of the target solution, the variability of TSV core geometry is considered.

The TSV, subcritical assembly support structure (SASS), TSV dump tank, piping, and associated dump valves are of robust construction and are seismically-qualified. In addition, the PSB and SASS are designed to withstand the pressures resulting from the maximum credible deflagration, and significant geometry changes are prevented during that event.

Vibration is considered in Subsection 4a2.7.3 and discussed as having no significant reactivity effect.

Consideration is given to the potential change in target solution spatial density caused by the formation and movement of voids. This can cause an insertion of reactivity event as voids form and collapse, but does not lead to uncontrolled/undamped power oscillations (see Subsection 4a2.6.1.4).

Sloshing of the target solution due to seismic acceleration is the limiting event for geometry changes. This event has been analyzed by assuming a range of sloshing amplitudes, for the minimum core volume and the nominal core. The effect generally results in negative reactivity effects in the TSV due to the geometry of the core. The sloshing distributes the core away from a more compact form, increasing neutron leakage. The event does not result in significant increases in power or challenging the safety limits of the system. No controls are needed to mitigate sloshing of the target solution. No damage to the PSB occurs and there are no radiological consequences.

Solution redistribution from vibrations is expected to be minimal and is bounded by sloshing.

<u>Scenario 8 – Reactivity Insertion Due to Moderator Lumping Effects</u>

The PCLS is a closed loop that circulates cooling water [ ]PROP/ECI past the TSV walls to remove heat generated in the TSV during normal irradiation operations. The cooling system design and operating characteristics preclude significant reactivity effects due to moderator changes in the subcritical assembly during operation. The PCLS is operated far from boiling conditions, and there is no scenario where boiling occurs in the PCLS. The PCLS passes through straight-through vertical cooling channels, which largely mitigate collection of voids and moderator lumping. Voids simply exit the top of the cooling channels. PCLS contains an air separator to remove entrained air.

Void formation within [ ]PROP/ECI changes the moderation profile in the TSV. A calculation of the expected reactivity changes due to voiding out the PCLS from nominal coolant temperature and density to a fully-voided cooling system was performed. Voids were assumed to occur in the cooling channels around the TSV [ ]PROP/ECI. The calculation was performed at cold (Mode 1) startup conditions and hot (Mode 2) irradiation conditions.

Results of the calculation show that for the PCLS, there is a positive insertion of reactivity with uniform voiding in the PCLS. The analysis shows that for a uniform voiding of 20 percent, reactivity changed by approximately [ ]PROP/ECI in Mode 1 and [ ]PROP/ECI in Mode 2. For a voiding of 100 percent, the reactivity changed by approximately [ ]PROP/ECI in Mode 1 and [ ]PROP/ECI in Mode 2. The reactivity impact from 5 percent voiding is very small (i.e., approximately [ ]PROP/ECI).

Given the inherent design of the TSV cooling channels to avoid accumulation of void and the air-water separator in the PCLS, there is no significant effect from moderator lumping. Additional design features to prevent cooling channel voiding are discussed in Subsection 5a2.2.2. No damage to the PSB occurs and there are no radiological consequences.

Scenario 9 – Inadvertent Introduction of Other Materials into the Target Solution Vessel

The chemical control of the target solution is performed during the preparation and adjustment of the solution in the RPF. Once the target solution is prepared for use in the TSV, there are no additional chemical control additives in the IF.

No significant pH changes are expected during irradiation due to the stability of sulfuric acid under irradiation.

While other materials are not normally added to the TSV, process upsets that could lead to inadvertent introductions were evaluated. The inadvertent introduction of other materials into the TSV could come from: (1) sources external to the PSB, or (2) sources internal to the PSB itself.

Material Entering the TSV from Sources External to the PSB

The TSV fill lines are isolated once the TSV is filled and ready for irradiation operations. There is no need to add any chemicals to control the chemistry of the target solution during the irradiation cycle.

The only systems that significantly interact with the SCAS during irradiation operations are the TOGS, NDAS, light water pool, and PCLS. The TOGS can adjust pressure and oxygen concentrations through gas removal and additions to the PSB. These gas space changes have no effect on reactivity beyond PSB pressure change reactivity effects, which are discussed in Section 4a2.6. The PCLS and light water pool are unable to add material to the TSV, except for water ingress scenarios, which are described in Subsection 13a2.1.2.2, Scenario 5.

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 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 leading to excess evaporation. The other identified scenario requiring control was failure of TOGS to return condensate to the TSV.

TOGS pressure control could fail during irradiation operations and cause lower pressure (higher vacuum) in the TSV, which could increase solution evaporation and/or cause boiling. This could result in increased uranium concentrations and a reactivity increase.

TOGS condensate return lines could clog, leading to increased holdup of condensate in TOGS or diversion of condensate to the TSV dump tank. Reduction of condensate return would lead to increased target solution uranium concentration and a reactivity increase.

The pressure control failure scenario is prevented through redundant TOGS vacuum relief valves that prevent excess vacuum in the PSB. Redundant relief valves protect the PSB from damage and results in no radiological consequences. The reduction in condensate return scenario is prevented through the TRPS IU Cell Safety Actuation on high power range neutron flux.

Scenario 11 – Failure to Control Temperature during 1/M Measurement at Startup

Postulation that a failure in the PCLS occurs during the startup process results in high target solution temperature and errors in the 1/M measurements during the fill process. These errors could be non-conservative and lead to an increase in reactivity during the fill process. This scenario is prevented through the TRPS IU Cell Safety Actuation on high source range neutron flux or high PCLS temperature. Following the TRPS trip, the target solution dumps to the TSV dump tank, decreasing reactivity and resulting in safe shutdown of the TSV.

Because each of these events has preventative measures in place, there are no radiological consequences.

13a2.1.2.3 Accident Consequences

No releases are expected to occur as a result of insertion of excess reactivity events described above. However, additional discussion associated with the most limiting scenario (Scenario 4 – High Reactivity and Power Due to High Neutron Production at Cold Conditions) is provided in Subsection 13a2.2.2.

13a2.1.3 REDUCTION IN COOLING

This subsection discusses the reduction in cooling in the SCAS. The following components were evaluated:

- The neutron multiplier
- The TSV containing uranyl sulfate solution
- The TSV dump tank containing uranyl sulfate solution

These components are cooled by the PCLS during irradiation operations to maintain a target solution average temperature of not more than 176°F (80°C) at 125 kW of heat generation in the TSV. PCLS rejects heat to the RPCS, which in turn is cooled by the process chilled water system (PCHS). Because the PCLS, RPCS, and PCHS cooling pumps are driven by off-site power, a loss of coolant flow occurs due to power failure and could occur due to failure of a pump, inadvertent valve closure, or a pipe break.

If cooling loop circulation flow is lost, the target solution is dumped to the TSV dump tank. The light water pool removes decay heat from the TSV dump tank by passively absorbing the heat in its approximately 14,900 gallons (56,400 L) water volume.

13a2.1.3.1 Identification of Causes, Initial Conditions, and Assumptions

The SCAS is cooled by the PCLS and the light water pool. The PCLS is a closed loop that circulates cooling water through [ ]PROP/ECI PCLS cooling water also flows around the TSV and neutron multiplier walls to remove heat generated in the target solution and neutron multiplier during normal irradiation and shutdown operations. Section 5a2.2 specifies that the PCLS is designed to remove 580,000 Btu/hr (170 kW).

The light water pool provides a large heat capacity for passively rejecting heat from the TSV dump tank during shutdown operations.

There are several failures that can result in the reduction in cooling in the SCAS:

- Loss of normal power
- Loss of or reduced cooling of PCLS due to:
  - Flow blockage
  - Pump malfunction
  - Operator error
  - Pipe break

- Valve closure
- Loss of RPCS or PCHS

These failures create two possible scenarios for reduction in cooling evaluation:

1) Loss of normal power resulting in loss of PCLS and de-energized neutron driver.
2) Loss of PCLS cooling due to blockage, malfunction, or operator error (neutron driver remains operating).

Scenario 1 – Loss of Normal Power

This scenario results in a loss of coolant flow in the PCLS cooling loop. The neutron driver does not function without off-site power, and therefore, the irradiation process is stopped. No further heat is generated in the target solution with the exception of power from delayed neutrons and decay heat.

Scenario 2 – Loss of PCLS Cooling

This scenario assumes a loss of PCLS flow without a LOOP, resulting in continued operation of the neutron driver. Full power continues to generate heat.

13a2.1.3.2 General Scenario Descriptions

Subsection 13a2.1.3.1 identifies two accident scenarios requiring evaluation of the temperature response of the light water pool and the target solution.

Scenario 1 – Loss of Normal Power

The loss of PCLS cooling flow is a result of a loss of normal power. The loss of neutron driver power terminates the neutron source production and reduces heat generated in the target solution prior to draining of the solution.

A TRPS signal initiates a Driver Dropout on low PCLS flow or high temperature, which opens the NDAS HVPS breakers, preventing a restart of neutron production. The TSV temperature increase prior to the TRPS dump of the target solution to the TSV dump tank introduces negative reactivity. The loss of PCLS flow also results in an IU Cell Safety Actuation after 180 seconds.

Redundant TSV dump valves open due to the TRPS actuation, draining the target solution to the TSV dump tank located in the light water pool. The light water pool is the heat sink for removal of decay heat from the target solution in the TSV dump tank. Thermal analysis of the TSV has been performed and shown that the target solution does not reach boiling conditions during this event, and no damage to the PSB occurs. The TSV dump tank is designed to maintain the target solution subcritical.

Scenario 2 – Loss of PCLS Cooling

A loss of PCLS cooling with continued operation of the neutron driver is assumed. Failures that can result in a loss of PCLS cooling are described in Subsection 13a2.1.3.1.

A low PCLS flow or high temperature signal initiates a Driver Dropout, which causes the NDAS HVPS breakers to open. This terminates the irradiation process by the accelerator.

After a 180 second delay, an IU Cell Safety Actuation is initiated, opening the redundant TSV dump valves and draining the target solution to the TSV dump tank. The light water pool is the heat sink for removal of decay heat from the target solution in the TSV dump tank. Thermal analysis of the TSV has been performed and shown that the target solution does not reach boiling conditions during this event, and no damage to the PSB occurs. The TSV dump tank is designed to maintain the target solution subcritical.

13a2.1.3.3 Accident Consequences

The accident consequences associated with reduction in cooling events are evaluated further in Subsection 13a2.2.3.

13a2.1.4 MISHANDLING OR MALFUNCTION OF TARGET SOLUTION

The TSV uses a liquid target solution that generates fission products that are contained by the PSB. The PSB consists of the TSV, the TSV dump tank, the TOGS, and associated connected piping and piping components. The accidents involving the mishandling or malfunction of the target solution within the IF, including a failure of the PSB, are analyzed here. The accidents involving mishandling or malfunction of target solution within the RPF are analyzed in Subsection 13b.2.4.

Within the boundaries of the IF, the target solution is contained in the TSV, the TSV dump tank, and associated connected piping and piping components.

Insertion of excess reactivity and inadvertent criticality events involving the target solution are discussed in Subsection 13a2.1.2.

13a2.1.4.1 Identification of Causes, Initial Conditions, and Assumptions

Mishandling and malfunction of target solution events fall broadly into two categories. The first is spills or leaks that cause target solution to migrate into unintended locations. The second category is changes in the physical or chemical form of the target solution that results in adverse effects. Within this category, three specific initiating events are considered: (1) failure to control pH of the solution, (2) failure to control solution temperature, and (3) failure to control solution pressure. These two categories of initiating events were used in the hazard evaluation process to inform the selection of appropriate initiating events for the SHINE system.

Several potential initiating events were evaluated, including:

- Spills or leakage from the TSV and process tanks
- Excessive cooling of target solution
- Precipitation of the target solution
- Failures of valves, piping, or tanks
- Failure to control pressure which initiates target solution boiling and impacts target solution concentration
- Operator errors or equipment failures resulting in inadvertently overflowing tanks or misdirecting flow

Failure to control pH of the target solution in the IF results in potential excessive corrosion and pressure boundary failure events, as described in this subsection, and potential for precipitation events as described in Subsection 13a2.1.2. Failure to control temperature or pressure of target solution are also described in Subsection 13a2.1.2.

Events involving the failure to control pH during solution preparation or adjustment are discussed in Chapter 13b.

The initial conditions and assumptions associated with mishandling or malfunction of target solution include:

- The PSB does not contain significant sources of pressure. Leakage between the PSB and the light water pool will normally flow from the pool to the PSB should a break occur.
- The primary confinement boundary isolates the PSB from the rest of the facility by robust walls, ceilings, and floors.
- Penetrations for piping, ducts and electrical cables, and shield plugs are sealed to limit the release of radioactive materials from the facility. Integrated leak rate from the primary confinement boundary (see Subsection 6a2.2.1) is less than that assumed in the dose analysis.
- The primary confinement is cooled by a recirculating air ventilation system. The primary confinement is ventilated to RVZ1e through the PCLS expansion tank.
- IF tanks and piping that have the potential to contain fissile material, except the TSV, are designed with passive measures that prevent an inadvertent criticality with the most reactive uranium concentration.
- Drains that lead from the pipe trenches and tank vaults are designed with a geometry that prevents an inadvertent criticality of the leaked target solution.

13a2.1.4.2 General Scenario Descriptions

There are several types of scenarios that are identified as mishandling or malfunction of the target solution within the IF: (1) failure of the PSB below the level of the light water pool, (2) failure of the TSV-to-PCLS pressure boundary resulting in in-leakage to the TSV, (3) failure of the RPCS-to-PSB interface, (4) failure of the TSV-to-PCLS pressure boundary resulting in target solution leakage to the PCLS, (5) failure in the TOGS causes high pressure in the TSV during fill mode, and (6) target solution leakage into a valve pit. Each of these scenarios and their potential causes is discussed below:

<u>Scenario 1a – Failure of the PSB Below the Level of the Light Water Pool</u>

A failure of a PSB component below the water line of the light water pool may be caused by excessive corrosion of PSB components. This failure results in water in-leakage to the primary system from the light water pool. The water in-leakage fills the dump tank, TSV, and TOGS with a mixture of target solution and pool water. Potential consequences of the flooding of the PSB include:

- An inadvertent criticality within the TOGS or,
- A deflagration of hydrogen gas in the TSV, TSV dump tank, or TOGS headspace due to insufficient sweep gas flow

A criticality in the TOGS could occur from target solution intrusion into the TOGS system. Criticality in the TOGS is prevented by the favorable geometry of the TOGS components, as described in Section 4a2.8.

Consequences related to hydrogen deflagrations are discussed in Subsection 13a2.1.9.

DID protections in place to prevent a failure of the PSB below the level of the light water pool are:

- control of solution pH through target solution sampling in the TSPS and target solution hold tank;
- a 30-year corrosion allowance in the PSB component design; and
- chemistry monitoring of the PCLS to limit corrosion (see Section 5a2.5).

Scenario 1b – Failure of the PSB Resulting in Target Solution Leakage into the Light Water Pool

A failure in the PSB below the light water pool surface may also result in target solution leakage from the primary system to the light water pool. The target solution mixes with the pool water and noble gases, volatile fission products, and particulates evolve into the IU cell gas space. Some of the radionuclides would then leak through the primary confinement boundary into the building and then into the environment. The consequences of leakage of target solution into the light water pool are mitigated by the primary confinement boundary, which keeps the doses within acceptable levels. The dose consequences of this accident scenario are analyzed in Subsection 13a2.2.4.

Scenario 2 – Failure of the TSV-to-PCLS Pressure Boundary Resulting in In-Leakage to the TSV

A failure of the PSB between the TSV and the PCLS may be caused by excessive corrosion of PSB components. This failure generally results in water in-leakage to the primary system from the PCLS. The water in-leakage fills the TSV dump tank, TSV, and TOGS with a mixture of target solution and PCLS water. Potential consequences of the flooding of the PSB include an inadvertent criticality within TOGS or deflagration of hydrogen gas in the TSV headspace or TOGS due to insufficient sweep gas flow. Criticality in the TOGS is prevented by the favorable geometry of the TOGS components, as discussed in Section 4a2.8.

Consequences related to hydrogen deflagrations are described in Subsection 13a2.2.9.

DID protections are present to help prevent a failure of the PSB between the TSV and PCLS, which include:

- control of solution pH through target solution sampling in the target solution hold tank;
- a 30-year corrosion allowance in the PSB component design;
- chemistry monitoring of the PCLS to limit corrosion (see Section 5a2.5).

Scenario 3 – Failure of the RPCS-to-PSB Interface

A failure of the RPCS pressure boundary in the TOGS may be caused by excessive corrosion of the PSB in a TOGS condenser. This failure results in water in-leakage to the primary system from the RPCS. The water in-leakage fills the TSV dump tank, TSV, and TOGS with a mixture of target solution and RPCS water. Potential consequences of the flooding of the PSB include an inadvertent criticality in TOGS or deflagration of hydrogen gas in the TSV headspace or TOGS due to insufficient sweep gas flow. Criticality in the TOGS is prevented by the favorable geometry of the TOGS components, as discussed in Section 4a2.8.

Consequences related to hydrogen deflagrations are discussed in Subsection 13a2.1.9.

Scenario 4 – Failure of the TSV-to-PCLS Pressure Boundary Resulting in Target Solution Leakage to the PCLS

Leakage from the primary system into the PCLS due to a failure of the PSB between the TSV and the PCLS is an additional concern. This failure results in: (1) a potential release of target solution into the primary cooling room with potential for higher dose to workers or the public, or (2) a criticality accident in PCLS equipment. Normally the PCLS is at higher pressure than the TSV, so water will flow from the PCLS into the TSV. However, once pressure equilibrium is established, target solution could leak into the PCLS. The protections in place to prevent and mitigate a failure of the PSB between the TSV and PCLS are PCLS isolation supply and return valves, radiation detection on the RVZ1e exhaust from the PCLS expansion tank, and redundant isolation dampers on the RVZ1e exhaust from the PCLS expansion tank. Target solution leakage into the PCLS will result in radioactive gases entering the PCLS expansion tank, flowing past the RVZ1e IU cell radiation monitors in the RVZ1e exhaust duct, and initiating an IU Cell Safety Actuation including isolation of the PCLS isolation valves and the RVZ1e exhaust duct.

As DID, the failure of the pressure boundary may first result in in-leakage and overflow into the TSV dump tank, which is detected with the level detection in the TSV dump tank. The TRPS then closes the PCLS isolation valves and RVZ1e isolation dampers, stopping potential transfer of target solution to the PCLS and reducing the source of water that could enter the PSB, and isolating the ventilation exhaust from the IU cell.

Additional DID 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 protections in place for this scenario are the configuration of the TSV fill line to prevent significant volume of target solution from backflowing from the TSV into the VTS lift tank and a check valve in the VTS uranium vacuum header. 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. If target solution were to enter the VTS header, the check valve would prevent the target solution from reaching non-favorable-geometry components in the VTS.

DID 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 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, (2) valve pit shielding and confinement for fission products that could result from leakage, reducing potential dose to workers and the public, (3) RDS sump tank liquid detection sensors detect fluid in-leakage, stopping in-process solution transfers within the facility, and (4) radiation monitors on the RVZ1 and RVZ2 building exhausts isolate building ventilation supply and exhaust dampers on high radiation, reducing potential dose to workers and the public.

Because this piping is potentially located in either the IF or the RPF, this event and associated dose consequences is further analyzed in Chapter 13b.

13a2.1.4.3 Accident Consequences

The release of target solution from the PSB to the light water pool or connected process systems results in potential radiological exposure to workers and the public. The accident consequences associated with the mishandling or malfunction of target solution are evaluated further in Subsection 13a2.2.4.

13a2.1.5 LOSS OF OFF-SITE POWER

A LOOP can occur for a variety of reasons related to the reliability and operation of the transmission system, stress during peak grid load conditions, severe weather effects from high wind, tornado, or ice and snowstorms, a seismic event, or equipment failure in the supplying substation. It may also be a result of failure or malfunction of the facility normal electrical power supply system (NPSS) such as the facility transformers or switchgear. This may result in a partial or complete LOOP to the facility.

Partial electrical power may also be lost resulting in partial system losses. System or equipment failures due to partial losses of electrical power within the facility are discussed under other accident analysis sections (e.g., Subsection 13a2.1.3). The partial loss of power scenarios are considered and bounded by the accident scenarios described in this section.

For the purposes of this discussion, it is assumed that a complete loss of off-site alternating current (AC) power occurs from causes that are external to the SHINE facility or common cause failures in the NPSS. Consequences of a complete LOOP to the facility are presented in Subsection 13a2.2.5.

13a2.1.5.1 Identification of Causes, Initial Conditions, and Assumptions

The electrical power systems that support the SHINE facility are described in detail in Chapter 8.

The standby generator system (SGS) is a commercial grade natural gas generator that is used for nonsafety functions at the SHINE facility. It is available as a normal back-up power supply for selected asset protection loads as discussed in Chapter 8 but is not credited as an emergency power source.

The uninterruptible electrical power supply system (UPSS) provides two divisions of safety-related emergency power to the SHINE facility. The facility equipment that is served by the UPSS is described in Section 8a2.2. This system is capable of delivering required emergency power for the required duration during normal and abnormal operation.

A LOOP may occur during any combination of operating modes within the IF and the RPF. Some potential causes of a LOOP are:

- Degradation (reliability) of the transmission system;
- Electrical grid stress during peak load conditions;
- Severe weather effects from high wind, tornado, ice or snowstorms;
- Seismic event;
- Equipment failure in the supplying substation; or
- Failure or malfunction of facility transformers or switchgear.

Loss of all off-site power bounds partial loss of power scenarios within the facility. Partial loss scenarios include: (1) a complete loss of one division of power, and (2) a loss of one individual bus or motor control center (MCC). The effect of partial loss of power scenarios are limited to those systems or processes affected whereas a total loss of electrical power affects all systems and processes.

The initial conditions and assumptions are summarized below:

- Eight TSVs are conservatively assumed to be in irradiation operations mode, with the maximum source term and decay heat levels.
- Irradiation power is assumed to be 137.5 kW, 10 percent above maximum operating power.
- Bulk target solution temperature in the TSV is at the limit of 176°F (80°C).

- Initial light water pool temperature is assumed to be 95°F (35°C).
- Complete loss of PCLS flow at time of initiating event.
- Light water pool level of not less than [ ]PROP/ECI below finished floor (water depth of approximately [ ]PROP/ECI), which provides sufficient passive heat sink to remove decay and residual heat from the target solution.
- Hydrogen concentration in TSV and TOGS is maintained within operating limits prior to the event.
- UPSS is available providing sufficient battery capacity for essential loads for their required runtime as provided in Table 8a2.2-1.
- Resupply of N2PS occurs within three days following a LOOP.

13a2.1.5.2 General Scenario Description

As noted in Subsection 13a2.1.5.1, the worst-case scenario is a complete LOOP. Although the interruption of off-site power may be relatively brief, it is assumed for this analysis that off-site power remains unavailable for an extended period of time. This could potentially occur if the LOOP is due to severe weather or a seismic event that damages substation equipment or associated transmission lines.

The sequence of events for a long-term LOOP is as follows:

- The UPSS automatically maintains power to the 125 VDC UPSS buses A and B, supplying power to the equipment identified in Section 8a2.2.
- Each NDAS HVPS de-energizes and the associated irradiation processes stop.
- The TSV dump valves open, draining the uranyl sulfate solution in the operating TSVs to their respective TSV dump tanks, as designed.
- The PCLS loses power to its pumps. Forced convection cooling ceases and heat is removed by natural convection to the light water pool.
- Hydrogen generation continues to occur due to radiolysis from the decay of fission products. The TOGS blowers and recombiner heaters operate on UPSS power for at least five minutes. The blowers continue forced flow through the TOGS recombiners for a short period of time as hydrogen production levels decrease and bubbles leave the target solution.
- The N2PS begins passively injecting nitrogen gas into the PSB. Nitrogen gas is injected in the eight SCAS systems via a connection to the dump tank. The gas purges the PSB 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.
- Eight NDAS contain maximum tritium inventory of [ ]PROP/ECI of tritium gas.
- The facility structure is designed to withstand credible external events including seismic events, severe weather effects, tornado generated missiles, and impact from aircraft.

In addition, seismic events assume that:

- SSCs, including their foundations and supports, that are required to perform their safety function(s) in the event of a design basis earthquake (DBE) are classified as Seismic Category I.
- SSCs that are co-located with a Seismic Category I SSC and that are required to maintain their structural integrity in the event of a DBE to prevent unacceptable interactions are classified as Seismic Category II.
- Seismic Category II SSCs are not required to remain functional in the event of a DBE.

For further details of seismic design criteria refer to Section 3.4.

13a2.1.6.2 General Scenario Descriptions

The following discusses the external event scenarios which impact the IF or RPF:

Seismic Events Affecting the IF and RPF

Scenario 1 – Seismic Event causing TOGS Failure

A seismic event may cause the failure of the TOGS. A failure of TOGS in one or more IUs could result in hydrogen deflagrations. Potential consequences of TOGS failure include radiological dose.

To prevent a TOGS failure from an earthquake the TOGS is seismically qualified. The UPSS provides the TOGS with emergency power if normal facility power is lost. The TOGS functions for a short time after the IU cell shutdown until the N2PS can purge TOGS and lower the concentration of hydrogen, reducing the possibility of hydrogen deflagration. Based on this discussion, the TOGS does not fail during a seismic event and no further analysis is required.

Scenario 2 – Seismic Event causing PCLS Failure

A seismic event may cause the failure of the PCLS. A failure of PCLS in one or more IUs could result in reduction in or excessive cooling, reactivity insertion, and potential criticality. Potential consequences of PCLS failure include radiological dose.

To prevent these conditions, redundant high power range neutron flux signals initiate a TRPS actuation that opens the redundant TSV dump valves. A TRPS actuation is also initiated on a high PCLS cooling water temperature or low PCLS cooling water flow. Reduction in cooling is discussed in Subsection 13a2.1.3. Reactivity insertions due to excessive cooling are discussed in Subsection 13a2.1.2. Based on this discussion, the loss of PCLS does not result in a radiological release and no further analysis is required.

Scenario 3 – Seismic Event Causing Multiple NDAS Failures

A seismic event may cause the failure of one or more NDAS units. A failure such as a NDAS vacuum boundary failure in one or more IUs results in a release of tritium in one or more IU cells. Potential consequences of multiple NDAS failures include radiological dose. The dose analysis conservatively assumes the simultaneous failure of all eight NDAS to bound the maximum allowable operating state in the IF.

To mitigate the impact of such failures of the NDAS vacuum boundary, TPS target chamber supply pressure and TPS target chamber exhaust pressure instrumentation and ventilation isolation mechanisms are used to confine released tritium. Accident consequences of this event are discussed in Subsection 13a2.2.6.

A seismic event may also cause the failure of TPS components located in the TPS glovebox. The radiological consequences of a failure of the TPS components due to a seismic event is bounded by the TPS failure due to deflagration, as described in Subsection 13a2.2.12.2.

Scenario 4 – Seismic Event Causing a Single NDAS Failure

A failure of the NDAS in a single IU cell is discussed in Subsection 13a2.1.12.

Scenario 5 – Seismic Event Causing NDAS Tritium Feed Fault

A seismic event may cause the failure of the NDAS tritium feed. A NDAS tritium feed failure results in tritium prematurely entering the NDAS which causes higher power density and potential uranium precipitation. Potential consequences of NDAS tritium feed failure include radiological dose.

To prevent these conditions, redundant high time-averaged power range neutron flux signals initiate a TRPS actuation that opens the redundant TSV dump valves. In addition, the TPS is provided with a passive design feature to limit the flow rate of tritium into the NDAS target chamber. In the event of NDAS tritium feed failure, the primary confinement boundary is used to contain such incidents. Excess reactivity insertions due to high neutron production at cold conditions are discussed in Subsection 13a2.1.2.2, Scenario 4. Based on this discussion, the tritium feed failure does not result in a radiological release and no further analysis is required.

Scenario 6 – Seismic Event Causing Light Water Pool Liner Failure

A seismic event may cause the failure or leak in the light water pool liner. A failure or leak in the light water pool liner could result in a loss of cooling water inventory and result in target solution heat up. Potential consequences of the light water pool liner failure include radiological dose.

To prevent a loss of pool cooling water, the light water pool liner is seismically-qualified, and penetrations through the liner are located above the minimum pool water height to limit out-leakage below this level. Piping penetrations into the light water pool with the potential for siphoning below the minimum acceptable water level contain anti-siphon devices or other means to prevent inadvertent loss of pool water. Because of the limited volume of water available to leak, anti-siphon design features, and the design leak rate of the penetration, no further analysis is required.

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 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 PSB 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 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.

To reduce the possibility of tube failure from a tornado or high wind event, the N2PS nitrogen tube bank is located in a reinforced concrete vault protecting the cylinders from tornado missile impact. No further analysis for this event is required.

Scenario 11 – Heavy Snow or Ice due to Severe Winter Weather

Heavy snow or ice accumulation due to severe weather may cause damage to the main production facility structure or systems including a loss of the normal building ventilation path or a loss of the safety-related PVVS effluent release path, which can then lead to a deflagration in the facility. Severe weather may also disrupt the nitrogen gas resupply following a N2PS activation.

The facility structure is designed to withstand heavy snow and ice loading to prevent damage. The exhaust point for the safety-related PVVS effluent path is designed to be above the design snow accumulation level, and the ventilation system air intakes are above the potential snow drift height. The N2PS system is supplied with enough nitrogen for three days of operation which is adequate to allow a resupply of the nitrogen tanks. No chemical or radiological consequences result from severe weather accident scenarios.

Transportation Accidents

The main production facility is designed to withstand credible aircraft impacts and transportation accidents, as discussed in Subsection 3.4.5.

External Flooding Affecting the IF and RPF

The main production facility was evaluated for external flood events. The results of the evaluation show that external flood events do not have an impact on the IF or RPF, as described in Subsection 2.4.2.

External Fires from Natural Sources

The main production facility was evaluated for the potential for external fires from natural causes. The results of the evaluation show that external fires from natural sources do not have an impact on the main production facility as described in Subsection 2.2.3.

13a2.1.6.3 Accident Consequences

The failure of eight NDAS pressure boundaries as a result of a seismic event (Scenario 3) results in potential radiological exposure to workers and the public. The primary confinement boundaries are credited to mitigate the consequences of this failure. The accident consequences associated with this external event are discussed further in Subsection 13a2.2.6.

13a2.1.7 MISHANDLING OR MALFUNCTION OF EQUIPMENT

The waste gases from irradiation of the target solution are of two major types: the hydrogen and oxygen produced by radiolysis of water in the target solution, and radioactive fission product gases. Mishandling or malfunction of equipment within the IU or TOGS cells has the potential to cause leakage of these gases. Specifically, a failure of the TOGS portion of the PSB could allow escape of fission product gases or hydrogen into the primary confinement boundary and the RCA. Analysis of this event and other potential mishandling or malfunction of equipment events, excluding detonation or deflagration of hydrogen within TOGS and other exothermic chemical reactions within the PSB, are included in this section.

- Events involving the mishandling or malfunction of target solution are discussed in Subsection 13a2.1.4.
- The detonation or deflagration of hydrogen within the TOGS or otherwise affecting the PSB is analyzed in Subsection 13a2.1.9.
- Other unintended exothermic chemical reactions within the PSB are analyzed in Subsection 13a2.1.10.
- The loss of vessels and line failures for systems within the RPF are analyzed in Subsection 13b.2.4.

The neutron driver and TPS system failures within the IU cell could similarly result in releases of tritium and hydrogen.

- Events related to the neutron driver are analyzed in Subsection 13a2.1.6 and Subsection 13a2.1.12.
- Events related to the TPS are analyzed in Subsection 13a2.1.12.

13a2.1.7.1 Identification of Causes, Initial Conditions, and Assumptions

A failure of the PSB resulting in a release of fission product gases may be caused by excessive corrosion of PSB components.

The initial conditions and assumptions associated with mishandling or malfunction of equipment include:

- Fission product gases (e.g., Kr, Xe, and halogens) produced during irradiation operations are retained within TOGS until the target solution batch irradiation cycle is completed.
- As the TOGS circulates sweep gas during the irradiation cycle, a portion of the iodine is removed by the zeolite beds, and hydrogen and oxygen are recombined by the catalytic recombiners, but no other gases are removed or purged.
- Each IU is operated on an irradiation cycle of 30 days with a minimum [ ]PROP/ECI residence for the target solution in the TSV dump tank following irradiation.
- The material-at-risk for these events is conservatively taken as the inventory at shutdown, at the end of the irradiation cycle, after [ ]PROP/ECI, with the safety-based assumptions listed in Table 11.1-1.
- The IUs are operated independently, so that an event on one TOGS does not affect another TOGS or IU cell.
- The TOGS cells are isolated from the rest of the facility by robust walls, ceiling, and floor. The physical separation of individual TOGS prevents malfunctions in one TOGS from affecting the others.
- Penetrations for piping, ducts, and electrical cables are sealed to limit the release of radioactive materials from the confinement boundary.
- The TOGS cells are cooled by a recirculating air ventilation system and are isolated from all other facility ventilation systems. A single ventilation connection from the PCLS expansion tank to the RVZ1e subsystem is provided for hydrogen gas removal from the cooling systems, and is isolated on a high radiation signal in the ventilation duct.

13a2.1.7.2 General Scenario Descriptions

There are three scenarios identified as mishandling or malfunctions of equipment. Each of these scenarios and their potential causes is discussed below:

<u>Scenario 1 – Failure of the TOGS Pressure Boundary Resulting in Release of Off-Gas into the TOGS Cell</u>

A failure of TOGS pressure boundary downstream of a TOGS blower may be caused by corrosion of a TOGS component. This failure results in a release into the TOGS cell of fission product gases and hydrogen normally managed by the TOGS, resulting in increased worker and public dose. Consequences of a release of fission product gases into the TOGS cell are discussed in Subsection 13a2.2.7. The protections in place to mitigate the consequences of a release of fission product gases are maintenance, inspection, and testing of the PSB, and the primary confinement boundary, which is described in Section 6a2.2.

<u>Scenario 2 – Failure of the TOGS Vacuum Tank</u>

The TOGS vacuum tank is in the light water pool below the water line. A failure of the vacuum tank that results in flooding the TOGS vacuum tank results in a loss of vacuum and flooding in the TOGS system on subsequent opening of the vacuum makeup valve. This scenario is considered in Subsection 13a2.1.4, which discusses failures of the PSB below the level of the light water pool.

Scenario 3 – TOGS Vacuum Makeup Valve Fails Open

The TOGS vacuum makeup valve inadvertently opens because of a failure of the valve controller or due to operator error. A subsequent opening of the VTS to vacuum tank isolation valve results in an elevated release of fission product gases and hydrogen to the VTS header and PVVS via the vacuum tank. The elevated release of fission product gases and hydrogen does not exceed the PVVS capability to remove the gases. The release is mitigated by the PVVS carbon delay beds and the PVVS guard beds. The mitigation controls reduce the release to the level of normal relief valve operation. Therefore, no further analysis is required.

13a2.1.7.3 Accident Consequences

The release of off-gas from the PSB to the IU or TOGS cell results in potential radiological exposure to workers or the public. The primary confinement boundary is credited to mitigate the consequences of a release of target solution in the IU or TOGS cell. The accident consequences associated with the mishandling or malfunction of equipment are evaluated as discussed above.

13a2.1.8 LARGE UNDAMPED POWER OSCILLATIONS

As recommended by the ISG Augmenting NUREG-1537 (USNRC, 2012a), the TSV is evaluated for large undamped power oscillations as a potential event that could occur during irradiation operation due to reactivity variations in the target solution that lead to fluctuations in the neutron multiplication (k_{eff}) within the irradiated target solution. Large undamped power oscillations are power oscillations that grow over time due to positive reactivity feedback effects and challenge the design limits of the subcritical assembly.

The TSV experiences power oscillations with reactivity variations within the target solution. Neutron driver output can oscillate and lead to power variations as well, but driver output oscillation amplitudes are limited due to the physical limitations of accelerator design. Coupled plant response as a result of transients has been analyzed. Power oscillations that occur are self-limiting as a result of the inherent design and safety characteristics associated with the TSV, operating parameters, and plant response to transients.

13a2.1.8.1 Identification of Causes, Initial Conditions, and Assumptions

Power oscillations may occur in the TSV as a result of normal anticipated reactivity variations within the target solution or neutron driver source output variations. The causes or scenarios for power oscillations include:

- TSV TOGS failure results in variations in TSV gas pressure.
- Variations in the neutron driver voltage, current, tritium concentration, or other parameter results in variations in the fusion neutron production rate.
- Failure in the PCLS temperature control loop or RPCS supply results in temperature oscillations.

The initial conditions and assumptions for this scenario are as follows:

- Negative temperature and void coefficients are within license limits discussed in Subsection 4a2.6.3.
- The TRPS neutron flux setpoints are within technical specification limits.

Perturbations from the TOGS can result in pressure changes in the TSV. During startup, irradiation, and shutdown operations, TOGS regulates gas pressures in the PSB to maintain pressures within the acceptable range. Increased gas pressures in the TSV reduce void fraction, leading to positive reactivity addition. Transient analysis of a complete void collapse is presented in Subsection 4a2.6.3.3. Excessive TSV power oscillations from TOGS pressure oscillations are prevented by redundant TRPS high neutron flux IU Cell Safety Actuation signals.

The neutron driver has variability in neutron production rates due to normal variations in beam current and focusing, voltages, and tritium gas concentrations. These variations lead to corresponding variations in fission power in the SCAS. An evaluation of the neutron driver induced transient is presented in Subsection 4a2.6.1.4. The TRPS high wide range neutron flux IU Cell Safety Actuation signal prevents excessive TSV power oscillations that challenge design limits should the neutron driver return to full power rapidly following a reduced power transient.

PCLS provides cooling water to the TSV, and therefore, temperature variations in the PCLS directly lead to TSV temperature variations. PCLS provides constant cooling water inlet temperature to the TSV within ranges described in Section 5a2.2. The target solution temperature ranges are provided in Section 4a2.2. Subsection 4a2.6.1 evaluates PCLS temperature variations and effects on the TSV. The TRPS high neutron flux IU Cell Safety Actuation signals prevent excessive TSV power oscillations from PCLS temperature variations. There are no large, undamped power oscillations that result from PCLS operation.

Power density limits for thermal-hydraulic and chemical stability of the target solution are described in Subsection 4a2.6.3. In addition, Subsection 4a2.6.3 discusses the limiting core configuration, which is the core configuration with the highest power density.

As discussed in Subsection 4a2.6.1.4 the large negative temperature and void coefficients result in a stable TSV with self-limiting power oscillations under analyzed reactivity variations.

13a2.1.8.2 General Scenario Description

As noted in Subsection 13a2.1.8.1, power oscillations may occur in the TSV during normal operation as a result of target solution reactivity or neutron driver source output variations. Because of the TSV and interfacing system design and operating parameters, the reactivity variations are small at operating power, resulting in a very stable TSV with self-limiting power oscillations.

Large power oscillations that could potentially challenge design limits are prevented by TRPS setpoints on high neutron flux. No operator actions are required to damp power oscillations. When a TRPS high neutron flux setpoint is exceeded, the neutron driver is automatically de-energized, the TSV dump tank valves automatically open, and the target solution is dumped (by force of gravity) into the favorable geometry TSV dump tank. No analyzed power oscillation scenario results in damage to the PSB.

13a2.1.8.3 Accident Consequences

Additional discussion associated with large undamped power oscillations is provided in Subsection 13a2.2.8.

13a2.1.9 DETONATION AND DEFLAGRATION IN THE PRIMARY SYSTEM BOUNDARY

This subsection discusses the effects of a hydrogen deflagration or detonation in the PSB. Irradiation of the target solution produces significant quantities of hydrogen and oxygen and small quantities of fission products. The TOGS is the primary control for mitigating hazards associated with the evolved gases. Functional requirements for the TOGS include maintaining the concentration of hydrogen to less than the lower flammability limit (LFL), recombining the hydrogen and oxygen, and returning the recombined water back to the TSV. The TOGS functions largely as a closed loop during the irradiation process, with gas additions and removals as needed to maintain proper functioning. TOGS is purged as needed to the PVVS via the VTS. Chapter 6 includes a discussion of the facility combustible gas management systems.

13a2.1.9.1 Identification of Causes, Initial Conditions, and Assumptions

The formation and release of hydrogen due to radiolytic decomposition is an inherent result of irradiation of water. Several potential scenarios that could result in the accumulation of hydrogen and potential deflagration or detonation were evaluated. A deflagration or detonation accident could occur if the TOGS fails, which could allow hydrogen to accumulate in the TSV headspace, dump tank, or TOGS piping. Potential failures that have been identified include a loss of power, failure of the TOGS blowers, blockage or restriction in the TOGS flow path, and water leakage into the PSB that results in reduced sweep gas flow. Hydrogen could also accumulate if there is degraded performance of the TOGS, such as reduced volumetric flow rate due to a partially-obstructed demister or reduced recombiner effectiveness.

Upon loss of TOGS function, hydrogen concentrations in the TSV headspace and TOGS rise. After the neutron driver is shut down on loss of TOGS flow, the voids in the target solution collapse and release hydrogen from the solution. This effect combined with continued radiolysis from delayed fission product decay further increases hydrogen concentration. Hydrogen concentration may increase above the LFL, which may cause a deflagration, but remains lower than the concentrations needed to produce a hydrogen detonation.

The initial conditions and assumptions associated with a deflagration of hydrogen gas are:

- A hydrogen deflagration produces a maximum overpressure condition of 65 pounds per square inch absolute (psia). Discussion of the design basis for the TOGS is found in Section 4a2.8.
- Each TSV is serviced by a dedicated and independent TOGS. In this section it is assumed a single TOGS fails, allowing hydrogen to accumulate in the TSV and TSV dump tank. Multiple TOGS failures resulting from a loss of normal power event are addressed in Subsection 13a2.1.5.
- The target solution is at steady-state conditions at 110 percent of the licensed power limit when the TOGS failure occurs. This is conservative since it implies the maximum hydrogen generation rate in the target solution.

13a2.1.9.2 General Scenario Descriptions

A deflagration could occur if the TOGS were to fail during the irradiation process. Irradiation of the target solution generates significant quantities of hydrogen and oxygen. The LFL for hydrogen in the headspace is rapidly reached if the TOGS fails and the neutron driver continues to operate.

Scenario 1 – TOGS Failure Resulting in Hydrogen Deflagration

A TOGS failure may occur due to flow blockage or failure of the TOGS blowers, loss of sweep gas flow to the TSV dump tank or overfilling the TSV that causes a reduction of available headspace and sweep gas flow. The loss of TOGS functionality allows the hydrogen gas concentration to increase in the headspace in the TSV and/or TSV dump tank. The hydrogen gas may ignite and cause a deflagration in the PSB. Failures of the PSB in the TOGS cell not related to deflagration or detonation are considered in Subsection 13a2.1.7.

Overfills of the TSV due to operator error are prevented by the redundant overflow lines to the TSV dump tank and redundant TSV dump tank level sensors which initiate a dump of the TSV. In addition, the criteria to transition from Mode 0 to Mode 1 include the TOGS mainstream flow being at or above the low flow limit to ensure that TOGS is functioning prior to fill operations.

In-leakage of water from the light water pool, PCLS, or RPCS into the PSB may cause loss of headspace. In these events, the N2PS purges sweep gas through the PSB to prevent damage to the PSB from excessive accumulation of hydrogen.

Nitrogen purging of the PSB is actuated by TRPS on signals that indicate loss of TOGS ability to properly maintain PSB hydrogen concentrations: low-high and high-high dump tank level, low TOGS oxygen concentration, low TOGS mainstream loop flow, low TOGS dump tank flow, high temperatures in the TOGS condenser demister, and ESFAS loss of electrical power. Nitrogen purging ensures that hydrogen concentrations remain below the level that could result in damage to the PSB. The operation of the N2PS is described further in Chapter 6, and the design of the system is described in Section 9b.6.

The pressure safety limit of the PSB is greater than the maximum credible deflagration pressure and does not fail due to a deflagration within the PSB.

Scenario 2 – PCLS Radiolysis Resulting in Hydrogen Deflagration

Under normal conditions, hydrogen gas generated in the PCLS is ventilated to the facility ventilation system (RVZ1e). A failure of the ventilation system may result in increased hydrogen gas concentration in the PCLS expansion tank. The hydrogen may ignite and cause a deflagration or detonation in the PCLS expansion tank, resulting in a release of radioactive material if the PSB is damaged.

A flame arrestor on the PCLS expansion tank that vents to the primary confinement atmosphere prevents potential ignition sources from causing a deflagration in the PCLS expansion tank. In the event that a release of radioactive material did occur, then the release is mitigated by the primary confinement boundary. Radiation detection instruments on the RVZ1e duct generates an IU Cell Safety Actuation and close redundant isolation valves to RVZ1e. The potential exposures from this event are bounded by the release of target solution to the IU cell, which is discussed in Subsection 13a2.2.4.

13a2.1.9.3 Accident Consequences

Because detonations and deflagrations in the PSB do not result in the failure of the PSB, there are no radiological consequences associated with these accident scenarios. Further discussion is provided in Subsection 13a2.2.9.

Analysis of PSB failures below the light water pool is provided in Subsection 13a2.2.4.

13a2.1.10 UNINTENDED EXOTHERMIC CHEMICAL REACTIONS OTHER THAN DETONATION

Unintended exothermic chemical reactions other than detonation have been evaluated as potential initiating events as part of the accident analysis within the IF. This subsection examines safety aspects of exothermic chemical reactions that challenge the PSB integrity in the IF, other than hydrogen deflagrations or detonations. Hydrogen deflagrations and detonations are addressed in Subsection 13a2.1.9.

13a2.1.10.1 Identification of Causes, Initial Conditions, and Assumptions

The scenario evaluated in this subsection is the uranium metal-water reaction in the neutron multiplier.

<u>Scenario 1 – Uranium Metal-Water Reaction in the Neutron Multiplier Assembly</u>

For the uranium metal-water reaction, the IU is operating at normal irradiation conditions. The neutron multiplier, as manufactured, is [ ]$^{PROP/ECI}$. The PCLS provides cooling to the TSV and the neutron multiplier and transfers gases produced from radiolysis to the expansion tank.

The neutron multiplier radionuclide inventory is developed assuming 30 years of continuous operation at 137.5 kW.

The uranium metal-water reaction in the neutron multiplier may be caused by an event which breaches the neutron multiplier cladding allowing water to come into direct contact with the uranium metal. Possible causes include corrosion of the cladding, uranium metal-cladding interaction due to radiation-induced growth, or other mechanical damage incurred during maintenance. The breach may occur at any time during the lifecycle of the neutron multiplier allowing water intrusion over an extended period of time.

13a2.1.10.2 General Scenario Descriptions

<u>Scenario 1 – Uranium Metal-Water Reaction in the Neutron Multiplier Assembly</u>

A small breach of the neutron multiplier cladding allows PCLS water into the cladding [ ]$^{PROP/ECI}$. The water intrusion results in an exothermic uranium metal-water reaction in the neutron multiplier assembly. The reaction generates hydrogen gas inside the neutron multiplier cladding shell [ ]$^{PROP/ECI}$. An accumulation of hydrogen gas could result in a deflagration under certain conditions. These conditions include sufficient oxygen concentration, an ignition source, or autoignition temperatures being reached. In this scenario, the hydrogen produced mixes with [

]PROP/ECI

inhibits a potential deflagration. Therefore, a hydrogen deflagration in the neutron multiplier from this event is considered unlikely.

Hydrogen gas that migrates into the PCLS stream from the neutron multiplier leak accumulates in the expansion tank, which is vented to the RVZ1e. Therefore, a hydrogen deflagration in the PCLS from uranium metal-water reactions is also unlikely.

13a2.1.10.3 Accident Consequences

The accident consequences associated with unintended exothermic chemical reactions are discussed further in Subsection 13a2.2.10. The accident consequences associated with a tritium release from the TPS glovebox are discussed further in Subsection 13a2.2.12.

13a2.1.11 SYSTEM INTERACTION EVENTS

This subsection discusses the effects of system interactions on the systems which contain radionuclide material. System interactions have the potential to cause damage that may lead to the release of these materials.

Three categories of system interactions between systems located within the IF and the RPF are considered in this analysis. These are: (1) functional interactions, (2) spatial interactions, and (3) human-intervention interactions.

Functional Interactions

Functional interactions are interactions between systems or subsystems that result from a common interface. A functional interaction exists if the operation of one system can affect the performance of another system or subsystem. An adverse functional interaction exists when the operation and/or performance of an (initiating) system adversely affects the operation and/or performance of an SSC as it performs its safety-related function. Functional interaction events 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 PSB.

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:

<u>Loss of Off-Site Power</u>

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

<u>Reduction of cooling</u>

- 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
  - MEPS process coolers
  - 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
  - RLWI 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. The spatial interactions considered include the effects of internal fires, internal flooding, chemical releases, and other dynamic failure effects.

Human-Intervention Interactions

Human-intervention interactions are adverse system interactions caused by human errors in the RPF which can cause adverse system performance in the subcritical assembly during irradiation operations. Human errors are identified as potential causes for other accident sequences and are not explicitly identified in this section. For example, human interactions or errors considered as potential causes for accident sequences include:

- Failure to operate equipment when required
- Inappropriate operation of equipment
- Maintenance error affecting operating equipment
- Testing error affecting operating equipment

Human errors downstream in the RPF processes that are related to mixing or transfer of target solution are considered in Subsection 13b.2.5.

13a2.1.11.1 Identification of Causes, Initial Conditions, and Assumptions

The identification of causes of system interaction events are provided in the subsections in Chapter 13 as referenced below. There are no unique initial conditions or assumptions associated with system interaction events.

13a2.1.11.2 General Scenario Descriptions

The following section discusses the system interactions that can occur at the main production facility. System interactions that are already analyzed in other parts of Chapter 13 are referenced to those subsections and not evaluated in this subsection. System interactions that are not described in other subsections are discussed below.

Functional Interactions

Loss of Off-Site Power

LOOP events are described in Subsection 13a2.1.5.

Reduction of Cooling

Events that could cause a reduction of cooling include PCHS or RPCS failure, LOOP, or external events.

- Reduction in cooling due to PCHS or RPCS failure is described in Subsection 13a2.1.3.
- Reduction in cooling following a LOOP is described in Subsection 13a2.1.5.
- Reduction in cooling due to external events is described in Subsection 13a2.1.6.

Loss of Ventilation

A loss of ventilation could be caused by equipment failure, a LOOP, or external events.

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 IU. 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 IU. 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 RCA exhaust filter room in the mezzanine area. Causes of fires include a catastrophic failure of the HVPS and maintenance activities including hot work.

The consequences of the fire scenarios are the potential release of radioactive materials, including tritium. The release of tritium is evaluated in Subsection 13a2.1.12.

Radioactive materials accumulated in the exhaust filter trains can also be released in the event of a fire. However the exhaust filter trains are monitored and alarmed for buildup and replaced. Therefore, a significant release of radioactive material is not expected to occur.

Additional effects of fire damage on other facility systems include potential loss of TOGS, PCLS, and ventilation system functions. Loss of the TOGS is described in Subsection 13a2.1.4 and Subsection 13a2.1.9. Loss of PCLS is described in Subsection 13a2.1.3 and Subsection 13a2.1.5. Loss of ventilation systems is described in Subsection 13a2.1.11.2.

The protections in place to prevent or mitigate the effects of a fire in the IF include the protection features described above (i.e., detection, rated barriers, manual suppression). Strict administrative controls on combustible materials and maintenance activities, including hot work are also in place. For a fire involving the HVPS, a catchment pan to contain oil leakage or spray limits the potential spread of oil reducing the potential for fire spread from the HVPS. Therefore, a release of radioactive material is not expected to occur.

TPS piping failures resulting in deflagration are discussed in Subsection 13a2.1.12.3.

Hydrogen deflagration within the PSB is discussed in Subsection 13a2.1.9.

Fires caused by external events are discussed in Subsection 13a2.1.6.

Exothermic Chemical Reaction Scenarios

Exothermic chemical reaction scenarios are discussed in Subsection 13a2.1.10.

Internal Flooding

Potential internal flooding scenarios in the IF have been evaluated.

There is no potential for widespread internal flooding within the IF. The primary sources of internal flooding are cooling water systems (e.g., PCLS) located in the IF with limited volume and pressure. The primary consequence of a leak in these systems is a loss of cooling to components served by the system. Localized water leaks or spray are contained to the room in which the system resides and would not result in widespread flooding.

One flooding scenario unique to the IF is a leak in a light water pool that serves the IU. A leak in the light water pool liner may result in leakage of water into the pipe trench and subgrade vaults introducing moderator around pipes and tanks containing uranyl sulfate solution. The nuclear criticality analyses for the trench and vaults assumes bounding moderation conditions which includes full reflection. Therefore there is no consequence as a result of this scenario.

A complete drainage of a light water pool due to a large break would also result in a loss of residual heat removal capability from the SCAS. The light water pool liner is designed to remain intact during normal operation as well as during DBE and DBA events. Penetrations through the light water pool liner are above the minimum water level.

Flooding caused by external events is discussed in Subsection 13a2.1.6.

Dynamic Effects

Process systems in the main production facility operate at low temperatures (i.e., generally less than 200°F [93°C], except for the TOGS hydrogen recombination components) and low pressures (i.e., less than 100 psig [689 kPa gauge]), which are not subject to dynamic effects as are found in high energy systems. As needed, safety-related systems are protected from the dynamic effects related to equipment failure and external events. No consequences result from dynamic effects interactions in the main production facility.

Human Intervention Interactions

Human interventions can cause adverse system interactions because of the single common control room at the main production facility. Operators are able to control multiple systems within the IF and the RPF from the control room. Operator errors may occur including performing control operations on the wrong system, failing to perform required actions, or performing actions out of sequence.

Maintenance is performed as a normal scheduled activity and as a response to emergent equipment problems. Maintenance may occur during all modes of operation, including while irradiation or processing activities are in progress. Errors that occur during maintenance activities can cause failures in operating systems such as support systems. Maintenance errors may be detected upon return to service through post-maintenance testing. However, undetected errors may result in system failures at some later point in time.

Human intervention interactions as accident scenario initiating events are described in other sections in this chapter as applicable and are not evaluated further in this section.

13a2.1.11.3 Accident Consequences

The system interactions described in the preceding sections do not result in radiological consequences. Accident consequences resulting from system interactions that are referenced to other subsections in Chapter 13 are evaluated in those subsections.

Further discussion regarding system interaction events described in this section is provided in Subsection 13a2.2.11.

13a2.1.12 FACILITY-SPECIFIC EVENTS

Several accident scenarios that are unique to the IF and have the potential for inadvertent radiation exposure to workers or members of the public were evaluated. Facility-specific accident scenarios are associated with the NDAS, the TPS, and potential IF damage resulting from heavy load drops.

13a2.1.12.1 Identification of Causes, Initial Conditions, and Assumptions

General scenario descriptions for events involving the NDAS, TPS, and heavy load drop include causes of each scenario.

For accident scenarios involving the NDAS, the following initial conditions and assumptions apply:

- The NDAS contains the bounding inventory of tritium gas for full power.
- The NDAS pressure vessel contains the maximum inventory of sulfur hexafluoride (SF_6) gas.
- The primary confinement boundary for an affected IU cell is operable, including the RVZ1e IU cell radiation monitors and isolation valves.

For accident scenarios involving the TPS, the following initial conditions and assumptions apply:

- The TPS glovebox confinement is operable, including the confinement isolation valves.
- The glovebox atmosphere is inerted with helium.
- Automatic isolation valves are installed in the system to isolate sections of the system to minimize system release.
- Leakage of tritium from the glovebox enclosure or the external piping is detected by the continuous airborne monitoring system (CAMS) or other leakage detection systems to provide alarms for facility personnel evacuation.
- The TPS-NDAS interface lines contain the maximum inventory of tritium gas.

For accident scenarios involving heavy load drops, the following initial conditions and assumptions apply:

- An IU cell is in maintenance with the IU cell shielding plug removed and the TSV and NDAS empty, or
- An IU cell is in service with IU cell shielding plug in place.

13a2.1.12.2 General Scenario Descriptions

Neutron Driver Assembly System Event Descriptions

There are four scenarios that are specific to the operation of the NDAS in the IF. These scenarios are: (1) inadvertent exposure to neutrons within the IU, (2) inadvertent exposure to neutrons in the NDAS service cell (NSC), (3) catastrophic failure of the NDAS, and (4) an NDAS vacuum boundary failure.

<u>NDAS Scenario 1 – Inadvertent Exposure to Neutrons within the IU</u>

Inadvertent exposure to neutrons may be caused by operation of a neutron driver while personnel are in the IU cell, such as during maintenance or assembly/disassembly activities, inadvertent access to an IU cell during irradiation operations, or failure to properly install IU cell shielding following access. An operator error which results in the neutron driver becoming energized with nearby personnel or without adequate shielding results in a significant neutron dose to workers. Operator error is the most likely cause of inadvertent exposure to neutrons within the IU.

Protections in place to prevent the inadvertent operation of a neutron driver in the IU cell are the lockout/tagout of the HVPS, opening of electrical breakers for the HVPS, and a two-key interlock for the NDAS control system. Also, the NDAS operating procedures require evacuation of deuterium and tritium from the NDAS and isolation of the deuterium and tritium supplies to the IU while in maintenance. Proper installation of IU cell shielding following IU cell access is verified before operation of the neutron driver. The accelerator cannot produce significant neutron-producing reactions without deuterium or tritium. The inadvertent exposure to neutrons within the IU is not credible due to the administrative controls and protections in place.

<u>NDAS Scenario 2 – Inadvertent Exposure to Neutrons within the NSC</u>

Inadvertent exposure to neutrons may be caused by operation of a neutron driver while personnel are in the NSC, such as during maintenance or assembly/disassembly activities. An operator error which results in the neutron driver becoming energized with nearby personnel or without adequate shielding results in a potential for significant neutron dose to workers. Operator error is the most likely cause of inadvertent exposure to neutrons within the NSC.

Protections in place to prevent the inadvertent operation of a neutron driver in the NSC are NSC operating procedures, which include independent confirmation of room clearance prior to testing, pre-job briefs, and postings; and a two-key interlock on the NDAS control system. Operating procedures also require control of deuterium gas during testing, to ensure deuterium gas supplies are only available when use is planned. The inadvertent exposure to neutrons within the NSC is not credible due to the administrative controls and protections in place.

NDAS Scenario 3 – Catastrophic Failure of the NDAS

Catastrophic failure of the NDAS may be caused by a failure of a ceramic component inside the neutron driver. A leak or failure of the ceramic results in a loss of NDAS vacuum inside the SF_6 pressure vessel and subsequent overpressure of the NDAS vacuum boundary causing failure. Failure of the vacuum boundary results in a leak of tritium and SF_6 gas to the IU cell, which causes IU cell pressurization. Pressurization of the IU cell can cause increased leakage rates between the IU cell and the IF, which results in higher dose to workers and the public due to the release of tritium.

The accident scenario is mitigated by the primary confinement boundary pressure sensors in the TPS target chamber supply lines and TPS target chamber exhaust lines and isolation valves on RVZ1e from the PCLS expansion tank. The primary confinement boundary is described in detail in Chapter 6. Multiple catastrophic failures of NDAS units are described in Subsection 13a2.1.6.

NDAS Scenario 4 – NDAS Vacuum Boundary Failure

Release of tritium from the NDAS vacuum boundary may be caused by a weld or vacuum seal failure or improper maintenance. A failure of the NDAS vacuum boundary results in a leak of tritium into the IU cell, causing higher dose to workers and to members of the public. The accident scenario is mitigated by the primary confinement boundary, which is described in detail in Chapter 6.

Tritium Purification System Event Descriptions

There are five scenarios that are specific to the operation of the TPS in the IF. These scenarios are: (1) TPS piping failure due to deflagration, (2) release of tritium into the IF due to glovebox deflagration, (3) release of tritium to the facility stack, (4) excessive release of tritium from the tritium storage bed, and (5) release of tritium into the IF due to TPS-NDAS interface line mechanical damage.

TPS Scenario 1 – TPS Piping Failure due to Deflagration

Improper system restoration following maintenance allowing air intrusion into TPS piping or by air in-leakage from the NDAS may result in a deflagration within the TPS piping that causes a piping failure and a release of tritium gas into the TPS glovebox. The release of tritium gas into the TPS glovebox results in higher dose to workers and to members of the public.

The release of tritium is confined within the tritium confinement boundary, including the TPS glovebox and secondary enclosure cleanup subsystem. The tritium confinement boundary is described in detail in Section 6a2.2. Isolation of the TPS room ventilation is also credited for mitigation.

TPS Scenario 2 – Release of Tritium into the IF due to Glovebox Deflagration

Leakage of TPS piping may lead to TPS glovebox failure caused by deflagration that causes the tritium confinement boundary to fail. TPS piping leakage may be the result of improper restoration to operating conditions from maintenance or of liquid nitrogen ingress into the gaseous nitrogen lines which causes embrittlement and failure of the TPS piping. Failure of the tritium confinement boundary releases tritium into the TPS room and results in higher dose to workers and to members of the public.

The TPS gloveboxes are designed such that the minimum size prevents the possibility of reaching the LFL for the quantity of available hydrogen. The TPS gloveboxes are also inerted with helium which prevents the presence of oxygen. Based on the glovebox design and inert atmosphere, a deflagration in a glovebox is not considered credible and is not analyzed further.

TPS Scenario 3 – Release of Tritium to the Facility Stack

A release of tritium directly to the facility stack may be caused by improper restoration to operating conditions from maintenance which results in a leak of tritium into a glovebox and a concurrent misalignment of the VAC/ITS valves following maintenance. A release of tritium to the facility stack results in higher worker and public dose.

The protection in place to mitigate a release of tritium to the facility stack is the tritium monitor on the TPS glovebox pressure control and VAC/ITS process vent exhaust to RVZ1e, which causes an isolation of the glovebox as part of the tritium confinement boundary.

TPS Scenario 4 – Excessive Release of Tritium from the Tritium Storage Bed

Excessive release of tritium from the tritium storage bed may be caused by failure of the storage bed heater control resulting in excessive heat input. Failure of the heater results in an excessive quantity of tritium added to the TPS system, resulting in overpressurization and release of tritium to a TPS glovebox. The tritium release is confined within the tritium confinement boundary, which is described in detail in Section 6a2.2.

The protection in place to mitigate a release of tritium to the facility stack is the tritium monitor on the TPS gloveboxes pressure control and VAC/ITS process vent exhaust, which causes an isolation of the glovebox as part of the tritium confinement boundary.

TPS Scenario 5 – Release of Tritium into the IF due to TPS-NDAS Interface Line Mechanical Damage

A release of tritium directly to the IF may be caused by mechanical damage to the TPS-NDAS interface lines which results in a leak of tritium to the IF. A release of tritium to the IF results in higher dose to workers and to members of the public. The TPS-NDAS interface lines are in subgrade penetrations which reduces the likelihood of mechanical damage that results in a tritium leak and are protected from mechanical impact between the subgrade penetration and the TPS gloveboxes.

The majority of the length of the TPS-NDAS interface lines are routed in subgrade sleeves and are therefore protected from mechanical damage from external impacts. A small length of the TPS-NDAS interface lines from the point at which they emerge from the subgrade sleeves in the TPS room to the TPS glovebox isolation valves is protected from mechanical damage by external guards. Therefore, TPS-NDAS interface line mechanical damage is not credible.

Heavy Load Drop Scenario Descriptions

With respect to the SHINE facility, a heavy load is defined as a load that, if dropped, may cause radiological consequences that challenge the accident dose criteria described in Section 13a2.2. There are three scenarios that are specific to heavy load drops in the IF. These scenarios are (1) a heavy load drop into an open IU cell, (2) a heavy load drop onto an in-service IU cell, and (3) a heavy load drop onto TPS equipment.

<u>Heavy Load Drop Scenario 1 – Heavy Load Drop into an Open IU Cell</u>

A crane mechanical failure or operator error during a lift may result in a heavy load drop into an open IU cell. The heavy load can damage the SCAS components 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 into an open IU cell is not credible.

<u>Heavy Load Drop Scenario 2 – Heavy Load Drop onto an In-Service IU Cell.</u>

A crane mechanical failure or operator error during a lift may result in a heavy load drop onto an in-service IU cell. The heavy load can damage the IU cell plug which results in damage to SCAS components 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 into an in-service IU cell is not credible.

<u>Heavy Load Drop Scenario 3 – Heavy Load Drop onto TPS Equipment</u>

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 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.

Table 13a2.1-1 – Hazard Types

Hazard Type	Hazards
Radiological	Fission products (in solution, aerosol, and off-gas), decay products, activation products, tritium, neutron, gamma
Fissile	Uranium oxide, uranium metal, uranyl sulfate (target solution), uranyl peroxide, uranium salts
Chemical - Toxic	Uranium, SF_6 gas, SF_6 decomposition products, fission and decay products
Chemical - Flammable/Explosive	Hydrogen gas, oxygen gas, uranium metal
Chemical - Reactivity	Sulfuric acid, nitric acid, NaOH
Chemical - Oxidizer	Oxygen gas, hydrogen peroxide
Chemical - Incompatibility	Acids and bases
Chemical - Asphyxiant	Nitrogen gas, SF_6 gas, clean agent for fire protection
Deflagration/Detonation	Hydrogen gas, oxygen gas
High voltage	Accelerator high voltage power supply
High pressure	Compressed gas cylinders (nitrogen, oxygen, helium), SF_6 gas
High temperature	Accelerator ion beam, process heaters, hydrogen recombiners
Low temperature	Liquid nitrogen
Kinetic energy	Ventilation and process steam blowers & fans
Potential energy	Pressurized gas cylinders (nitrogen, oxygen, helium), SF_6 pressure vessel
Internal fire	Initiators (electrical equipment, maintenance), combustible materials, hydrogen gas
Internal flooding	Process equipment, fire protection, cooling water systems
External events	Seismic, tornado, tornado generated missiles, severe weather, flooding (possible maximum precipitation), external fire, aircraft impact, industrial and transportation events (toxic gas, explosion)

Table 13a2.1-2 – Risk Matrix

Severity of Consequences	Likelihood of Occurrence		
	Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Consequence Category 3 High (3)	Acceptable 3	Unacceptable 6	Unacceptable 9
Consequence Category 2 Intermediate (2)	Acceptable 2	Unacceptable 4	Unacceptable 6
Consequence Category 1 Low (1)	Acceptable 1	Acceptable 2	Acceptable 3

Table 13a2.1-3 – Likelihood Category Definitions

Likelihood Category	Likelihood Index (T)	Event Frequency Limit	Risk Index Limits
Highly Unlikely	1	Less than 10^{-5} per event, per year	T ≤ -5
Unlikely	2	Between 10^{-4} and 10^{-5} per event, per year	-5 < T ≤ -4
Not Unlikely	3	More than 10^{-4} per event, per year	-4 < T

Table 13a2.1-4 – Failure Frequency Index Numbers

Failure Frequency Index Number (FFIN)	Based on Evidence	Based on Type of Control	Comments
-6	External event with freq. < 10^{-6}/yr	N/A	If initiating event, no controls needed.
-5	Initiating event with freq. < 10^{-5}/yr	N/A	For passive safe-by-design components or systems; failure is considered highly unlikely for robust passive engineered controls: 1. Whose dimensions fall within established single parameter limits or that can be shown by calculation to be subcritical including the use of the approved subcritical margin, 2. That have no credible failure mechanisms that could disrupt the credited design characteristics, and 3. Whose design characteristics are controlled so that the only potential means to effect a change that might result in a failure to function would be to implement a design change
-4	No failures in 30 years for hundreds of similar controls in industry	1. Exceptionally robust passive engineered control (PEC), 2. Two independent active engineered control (AECs), PECs, or enhanced specific administrative control (SAC)	Rarely can be justified by evidence. Further, most types of single control have been observed to fail.
-3	No failures in 30 years for tens of similar controls in industry	A single control with redundant parts, each a PEC or AEC	None
-2	No failure of this type in the facility in 30 years	A single PEC	None
-1	A few failures may occur during facility lifetime	1. A single AEC 2. Enhanced SAC 3. Redundant SAC	None
0	Failure occurs every 1 to 3 years	A single SAC	None
1	Several occurrences per year	Frequent event, inadequate control	Not for controls, just initialing events.
2	Occurs every week or more often	Very frequent event, inadequate control	Not for controls, just initialing events.

Table 13a2.1-5 – Failure Probability Index Numbers

Failure Probability Index Number (FPIN)	Probability of Failure on Demand	Based on Type of Control	Comments
-6	10^{-6}	N/A	If initiating event, no control needed.
-4 or -5	10^{-4} - 10^{-5}	1. Passive engineered control (PEC) with high design margin. 2. Inherently safe process. 3. Two redundant controls more robust than a simple AEC, PEC, or enhanced SAC.	Can rarely be justified by evidence. Most types of single controls have been observed to fail.
-3 or -4	10^{-3} - 10^{-4}	1. Single PEC 2. Single AEC with high availability	None
-2 or -3	10^{-2} - 10^{-3}	1. Single AEC 2. Enhanced SAC 3. SAC for routine planned operations	None
1- or -2	10^{-1} - 10^{-2}	A SAC that must be performed in response to a rare unplanned demand.	None

Table 13a2.1-6 – Duration Index Numbers

Duration Index Number (DIN)	Average Failure Duration	Duration in Years	Comments
1	> 3 years	10	None
0	1 year	1	None
-1	1 month	0.1	Formal monitoring to justify indices < -1
-2	A few days	0.01	None
-3	8 hours	10^{-3}	None
-4	1 hour	10^{-4}	None
-5	5 minutes	10^{-5}	None

Table 13a2.1-7 – Consequence Category Definitions

Consequence Category	Facility Staff	Off-Site Public
High Consequence 3	RD > 100 rem CD > PAC-3	RD > 25 rem 30 milligrams sol U intake CD > PAC-2
Intermediate Consequence 2	5 rem < RD ≤ 100 rem PAC-2 < CD < PAC-3	1 rem < RD ≤ 25 rem PAC-1 < CD ≤ PAC-2
Low Consequence 1	Accidents with lower radiological and chemical exposures than those above	Accidents with lower radiological and chemical exposures than those above

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 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 Method

The transport of radioactive material for the accident analysis was quantitatively evaluated using a 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 (Pa-s/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:

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

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 *IAj* is the integrated activity in the control room at time bin *j*, *Oj* 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:

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

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 *IAj* and *Oj* 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 (χ/Q)j replaces the control room occupancy factor.

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

The $RAF_{p,}$ j 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 are consistent with the methodology as described in Section 3.2.5.2 of NUREG/CR-6410 (USNRC, 1998). However, the combined RAF term described in the previous section accounts for the leak path factor, airborne release fraction, and atmospheric dispersion factors. The cumulative RAF values represent the time-dependent leakage of radionuclides. The RAF values are calculated for the leakage from the source volume to the building for the worker dose (duration of the event), and the source volume to the environment for the public dose (duration of the event).

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 factor (DCF) values for submersion for each radionuclide. Inhalation doses are calculated based on the respirable fraction, DCF for inhalation, and breathing rate. The DCF values used in the analysis are taken from 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 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. A ground release was used as the release point.

Releases into the IF or RPF control volumes are assumed to be detected by the radiation monitors on the radiological ventilation zone 1 (RVZ1) and radiological ventilation zone 2 (RVZ2) building exhausts, isolating building ventilation supply and exhaust dampers on high radiation, reducing potential dose to workers and the public.

<u>Conservatism</u>

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 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.

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 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 MAXIMUM HYPOTHETICAL ACCIDENT

As described in Subsection 13a2.1.1, the postulated maximum hypothetical accident (MHA) for the SHINE facility is a failure of the 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 13a2.2.7.

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 feedback inherently reduces reactivity and power. However, during some transients the power increases to a level higher than the steady state maximum power level of 125 kW.

The scenario that was found most limiting is the high power due to high neutron production and high reactivity at cold conditions (Scenario 4 described in Subsection 13a2.1.2.2). This limiting scenario adds substantial reactivity to the operating system and results in the highest calculated peak power.

13a2.2.2.3 Sequence of Events

The limiting sequence of events is as follows:

1. The TSV is operating at steady state at the licensed power limit of 125 kW, with an operating k_{eff} of approximately [ ]$^{PROP/ECI}$.
2. The accelerator ceases to produce neutrons because of an upset, dropping source neutron production to 0 percent. The TRPS detects the loss of neutron production and begins the [ ]$^{PROP/ECI}$ delay prior to initiating a Driver Dropout.
3. The primary closed loop cooling system (PCLS) continues to function and cool the target solution in the TSV.
4. As the system reduces in power, void leaves the solution, causing a reactivity increase of up to [ ]$^{PROP/ECI}$. The k_{eff} after void loss is approximately [ ]$^{PROP/ECI}$.

5. At [ ]PROP/ECI after the loss of neutron production, target solution cooling results in a temperature decrease of less than 7°C. This results in a reactivity increase of up to [ ]PROP/ECI. The system remains subcritical.
6. The accelerator output is restored at [ ]PROP/ECI just prior to the TRPS Driver Dropout initiating. Power increases to a level that is greater than the steady state power before the upset occurred, with a peak power calculated at [ ]PROP/ECI. This power level would result in a TRPS IU Cell Safety Actuation on high wide range neutron flux.

<u>Safety Controls</u>

For this most limiting scenario, the safety controls that prevent consequences of an insertion of excess reactivity event and ensure that damage to the PSB does not occur are:

- Low power range neutron flux
- TRPS Driver Dropout, resulting in redundant neutron driver assembly system (NDAS) high voltage power supply (HVPS) breakers opening
- Redundant HVPS breakers on neutron driver
- High wide range neutron flux
- TRPS IU Cell Safety Actuation on high wide range neutron flux

Additional safety controls that prevent consequences of other scenarios described in Subsection 13a2.1.2.2 include:

- TRPS IU Cell Safety Actuation on the following parameters:
  - High time-averaged neutron flux
  - High wide range neutron flux
  - High source range neutron flux
  - Low PCLS temperature
  - High PCLS temperature
  - Low PCLS flow
- TRPS IU Cell Nitrogen Purge on the following parameters:
  - Low-high TSV dump tank level
  - High-high TSV dump tank level

13a2.2.2.4 Damage to Equipment

The TRPS is designed to end the event and place the target solution in a safe shutdown condition without the need for operator action. The TRPS prevents challenges to the integrity of the PSB. No equipment damage results from the postulated insertion of excess reactivity event.

13a2.2.2.5 Radiation Source Terms

Because the postulated insertion of excess reactivity events do not exceed any design limits or cause damage to the PSB, there is no radiation source term.

13a2.2.2.6 Radiological Consequences

Because the insertion of excess reactivity events do not exceed any design limits or cause damage to the PSB, there are no radiological consequences to workers or the public.

13a2.2.3 REDUCTION IN COOLING

This section discusses the analysis and determination of consequences for the reduction in cooling event.

13a2.2.3.1 Initial Conditions

Initial conditions of the loss of or reduced PCLS flow event are:

- The TSV is operating normally at steady-state in Mode 2 (irradiation).
- 137.5 kW thermal power generated within the target solution.
- PCLS is providing greater than the minimum flow rate of [ ]PROP/ECI.
- PCLS supply temperature is less than the maximum supply temperature of 77°F (25°C).
- The TSV was filled to the minimum cold fill volume of [ ]PROP/ECI, which maximizes power density.
- Average target solution temperature of up to 175°F (80°C).
- Target solution decay heat based on end of life target solution conditions, which generates the highest decay heat.

13a2.2.3.2 Initiating Events

Subsection 13a2.1.3 describes two possible scenarios for the reduction in cooling accident: (1) loss of normal power, and (2) loss of PCLS flow. Of these two, the loss of or reduced PCLS flow is the limiting event, because in this scenario accelerator operation could continue with reduced cooling flow. In the loss of normal power event, the accelerator is unable to function without continuous power, greatly reducing heat production. Target solution is drained to the TSV dump tank after an allowable delay, and decay heat is removed by the light water pool.

The scenario postulates a reduction in PCLS flow without a loss of off-site power (LOOP). The neutron driver continues to operate. There are numerous possible causes that could result in loss of total flow or flow reduction in the PCLS, including failure of the power supply to the pump, pump shaft lockup or failure, or operator error. Loss of PCLS cooling could also result because of flow path isolation due to inadvertent valve closure, loss of heat sink, or PCLS leakage due to a piping or component failure.

13a2.2.3.3 Sequence of Events

The scenario starts with the TSV in Mode 2, operating normally at full power. PCLS cooling flow is reduced, resulting in increased TSV temperature. Depending on the failure, PCLS supply temperature may also increase. The neutron driver is de-energized by the TRPS Driver Dropout on low PCLS flow. For loss of PCLS cooling capability events due to high temperatures, the TRPS Driver Dropout would also actuate on high PCLS temperature.

The low PCLS flow trip is a minimum of [ ]PROP/ECI, and the PCLS supply temperature is a maximum of 77°F (25°C).

The TRPS Driver Dropout opens the NDAS HVPS breakers, terminating neutron production. After a 180 second delay, the TRPS initiates an IU Cell Safety Actuation on loss of PCLS flow (or high PCLS temperature). The TSV dump valves open and the target solution is dumped to the

TSV dump tank. The light water pool provides passive cooling to the TSV dump tank for the removal of decay heat from the target solution.

Safety Controls

The following safety controls prevent a reduction in cooling event and ensure that the target solution in the TSV does not boil:

- TRPS Driver Dropout on loss of PCLS flow and high PCLS temperature
- TRPS IU Cell Safety Actuation on low PCLS flow rate and high PCLS temperature
- Light water pool
- Verification of light water pool level prior to entering Mode 1
- NDAS HVPS trip breakers
- Redundant TSV dump valves

13a2.2.3.4 Damage to Equipment

The TRPS is designed to end the event and place the target solution in a safe shutdown condition without the need for operator action. The TRPS also prevents challenges to the integrity of the PSB. No equipment damage results from the postulated reduction in cooling event.

13a2.2.3.5 Radiation Source Terms

Analyses show that if the PCLS supply temperature exceeds the operating limit of 77°F (25°C) or the PCLS flow rate is below the operating limit of [ ]PROP/ECI, TRPS indicates an IU Cell Safety Actuation, the target solution is transferred to the TSV dump tank where it is passively cooled by the light water pool, and there is no boiling in the TSV or in the TSV dump tank.

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 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 via the RVZ1e IU cell radiation monitors 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.

<u>Safety Controls</u>

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

- Primary confinement boundary
- Ventilation radiation monitors

- RVZ1e IU cell isolation mechanisms
- Holdup volume in the RVZ1e
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

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 RAF 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.

Iodine is partitioned by assuming that the iodine present is fully dissolved into the target solution prior to the initiating event and none is present in the gas space of the TSV. Once the event occurs, the iodine is dissolved in the water and partitioned according to the pH of the pool, the temperature of the pool, and the pool and gas volumes. Deposition of iodine on the walls of the IU cell due to the temperature difference of the warm gas and the cold walls is also credited as a removal mechanism. As iodine is deposited on the cell walls, more iodine is evolved from the light water pool and into the gas space. The iodine partitioning determines the transport of volatile iodine out of the pool.

Some radionuclides deposited in the light water pool are released to the gas space as an aerosol when radiolytically-generated hydrogen gas bubbles burst at the pool surface. Radiolysis becomes a long-term source of both non-volatiles and iodine.

Once the MAR is released into the gas space of the IU cell there are several paths through which leakage could occur. The primary leak path will be around the IU cell plug perimeter. Potential leak paths are modeled as a single leakage junction to the IF.

13a2.2.4.6 Radiological Consequences

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

13a2.2.5 LOSS OF OFF-SITE POWER

13a2.2.5.1 Initial Conditions

Facility power is being supplied from off-site by the electric utility. The initial conditions for the LOOP event are further described in Subsection 13a2.1.5.1.

13a2.2.5.2 Initiating Event

The initiating event is a LOOP resulting in the loss of the normal electrical power supply system (NPSS) function.

13a2.2.5.3 Sequence of Events

The sequence of events for an extended LOOP is described in Subsection 13a2.1.5.2.

The facility combustible gas management system described in Chapter 6 is designed to function following a LOOP to disperse the combustible gases that are generated by radiolysis. This system removes combustible gases through the PVVS carbon beds and filters to the environment, through the PVVS alternate vent path.

<u>Safety Controls</u>

The safety controls credited for prevention of accidents resulting from a LOOP event are:

- Uninterruptible electrical power supply system (UPSS)
- NDAS HVPS breakers
- TSV dump valves
- Light water pool
- Verification of light water pool level prior to entering Mode 1
- TOGS
- N2PS
- Nitrogen in the N2PS is refilled every 72 hours while in use
- PVVS alternate vent path
- PVVS carbon guard and carbon delay beds

13a2.2.5.4 Damage to Equipment

The LOOP event does not result in any damage to equipment.

The safety-related functions of the equipment supplied by the UPSS are uninterrupted; therefore, the safety-related functions continue to be performed. Irradiation processes stop, and the target solution is drained from operating TSVs to their respective TSV dump tank. Decay heat is removed by natural convection to the light water pool. The combustible gas management system eliminates the risk of a release of radioactive material due to deflagration.

13a2.2.5.5 Radiation Source Terms

Because the postulated LOOP event does not result in the loss of safety-functions of the equipment supplied by the UPSS, there is no radiological source term for this accident sequence.

13a2.2.5.6 Radiological Consequences

Because the postulated LOOP event does not result in the loss of safety-functions of the equipment supplied by the UPSS, there are no radiological consequences for this accident sequence.

13a2.2.6 EXTERNAL EVENTS

The facility structure is designed to withstand credible external events as described in Subsection 13a2.1.6. Most of the analyzed accidents involving credible external events are prevented by the facility structure or seismic qualification of affected SSCs. The only postulated accident scenario resulting in a radiological release involving an external seismic event is a tritium release from simultaneous failure of multiple NDAS units. The simultaneous release of tritium from all eight operating neutron driver assemblies is analyzed as a DBA. This scenario is described in Subsection 13a2.1.6 as Scenario 3. The consequences of this accident are analyzed in this subsection.

13a2.2.6.1 Initial Conditions

The initial conditions for external events are described in Subsection 13a2.1.6.1.

13a2.2.6.2 Initiating Event

A seismic event is the initiating event for a tritium release into multiple IU cells. All NDAS accelerators experience vacuum boundary component failures and cause a pressurized release of tritium and SF_6 gas into the eight IU cells simultaneously as a result of the seismic event.

The initial accident conditions for each IU cell are the same to those accident conditions involving the NDAS of a single IU cell, as described in Subsection 13a2.1.12.1.

13a2.2.6.3 Sequence of Events

The accident sequence proceeds as follows:

1. The initiating event is a seismic event that causes the simultaneous vacuum boundary component failure in all eight NDAS units, instantaneously releasing tritium and SF_6 gas into the IU cells.
2. The IU cells become slightly pressurized due to the mass of released SF_6 gas.
3. Some tritium is transported into the IF through penetrations in the confinement boundary.
4. Detection of high TPS target chamber supply pressure or high TPS target chamber exhaust pressure actuates the primary confinement boundary isolation valves and irradiation unit trips 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. Tritium migrates to the IF through the IU cell plugs and is released to the environment.
6. Detection of high TPS target chamber supply pressure or high TPS target chamber exhaust pressure 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 within 10 minutes upon actuation of the radiation alarms.

Radiation transport is driven primarily by barometric breathing between the IU cell and the IF.

The safety-related SSCs in the IU cell do not fail during a seismic event, but the NDAS and its internal components are not safety-related and cannot be relied upon to remain intact following a design basis earthquake.

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

Safety Controls

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

- Primary confinement boundary
- TPS Train Isolation on high TPS target chamber supply pressure or high TPS target chamber exhaust pressure
- Ventilation isolation mechanisms
- Holdup volume in the RVZ1e
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

It is assumed that the primary confinement is intact and performs a mitigation function with respect to radionuclide transport from the IU cells 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 systems.

13a2.2.6.4 Damage to Equipment

Failure of the NDAS vacuum boundary does not cause subsequent damage to equipment. While the NDAS vacuum boundary integrity is not seismically qualified to maintain integrity, the NDAS is designed to maintain structural integrity during and following a design basis earthquake.

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 between the cells and the IF is not impacted by the accident sequence.

13a2.2.6.5 Radiation Source Terms

The initial MAR for this scenario is a total of [ ]PROP/ECI of tritium from all of the neutron driver assemblies.

The accident source term development is discussed in Section 13a2.2. The RAF 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.

13a2.2.6.6 Radiological Consequences

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

13a2.2.7 MISHANDLING OR MALFUNCTION OF EQUIPMENT

The bounding scenario analyzed for mishandling or malfunction of equipment events is a loss of the PSB integrity which results in a release of off-gas into the TOGS cell. This scenario is described in Subsection 13a2.1.7.2 as Scenario 1.

13a2.2.7.1 Initial Conditions

Initial accident conditions are described in Subsection 13a2.1.7.1.

13a2.2.7.2 Initiating Event

The accident sequence is initiated by a failure of the PSB in the TOGS within the TOGS cell. The cause of the initiating event is discussed in Subsection 13a2.1.7.

13a2.2.7.3 Sequence of Events

The accident sequence proceeds as follows:

1. A failure of the 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 Section 6a2.2.
3. Some radioactive material is transported into the IF through penetrations in the confinement boundary.
4. The radioactive material is then dispersed throughout the IF and exits to the environment through building penetrations.
5. Detection of high radiation in the RVZ1e ventilation from the IU cell via the RVZ1e IU cell radiation monitors actuates ventilation dampers and minimizes the transport of radioactive material to the environment. The assumed response time for RVZ1e ventilation is 20 seconds from detection of high airborne radiation. A sufficient time delay is provided by the holdup volume in RVZ1e to prevent radioactive gases from exiting through RVZ1e prior to isolation.

6. 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. The N2PS actuates.
7. The main facility ventilation system (i.e., RVZ2) is isolated by the ESFAS within 30 seconds of detectable accident conditions. Leakage to the environment continues through unfiltered leakage pathways.
8. Personal dosimeters, local radiation alarms, and alarms in the facility control room notify facility personnel of radiation leakage.
9. Facility personnel evacuate the immediate area within 10 minutes upon actuation of the radiation area monitor alarms.

A portion of the gaseous iodine is adsorbed onto the cell walls, while the majority of the available MAR is transported to the IF through pressure-driven flow caused by the N2PS and leakage through the primary confinement boundary. Transport to the environment occurs through penetrations in the RCA boundary.

Safety Controls

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

- Primary confinement boundary
- RVZ1e IU cell radiation monitors
- Ventilation isolation mechanisms
- Holdup volume in the RVZ1e
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

It is assumed that the primary confinement boundary is intact and performs a mitigation function with respect to radionuclide transport from the TOGS 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 systems.

13a2.2.7.4 Damage to Equipment

The TOGS zeolite bed may continue to function following a failure of the TOGS but is not credited for source term reduction following the initiating event. Similarly, under normal operating conditions, the recirculating ventilation in the TOGS cell provides filtration which may reduce dose consequences but is not credited to remain functional under accident conditions. These assumed failures are conservative.

Leakage of the TOGS pressure boundary does not cause subsequent damage to equipment credited for safety.

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 RAF 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.
4. The TSV dump tank valves automatically open and the target solution is dumped by force of gravity into the subcritical dump tank with favorable geometry, ending the event sequence.

Safety Controls

- The design and safety characteristics of the TSV to resist undamped power oscillations.
- IU Cell Safety Actuation initiated by TRPS
- TRPS high neutron flux trips
- TSV dump valves and TSV dump tank

13a2.2.8.4 Damage to Equipment

No damage to equipment occurs because power oscillations in the TSV are self-limiting and do not become large undamped power oscillations. The TRPS high neutron flux limits halt power oscillations before they challenge design limits.

13a2.2.8.5 Radiation Source Terms

Because power oscillations in the TSV are self-limiting and because the TRPS acts to prevents power levels that challenge design limits, there is no damage to the PSB, and therefore no radiation source term.

13a2.2.8.6 Radiological Consequences

Because large undamped power oscillations are shown to not occur and large power oscillations that challenge design limits are halted before they occur, there are no radiological consequences to workers or the public.

13a2.2.9 DETONATION AND DEFLAGRATION IN THE PRIMARY SYSTEM BOUNDARY

The release of hydrogen and oxygen by radiolysis from the target solution during and after irradiation may lead to high concentrations of hydrogen, which may then result in detonation or deflagration within the PSB. Normally, the TOGS provides ventilation of the headspace above the TSV to maintain hydrogen concentrations below the lower flammability limit (LFL). A failure of the TOGS to perform its design function may result in conditions that could lead to a hydrogen detonation or deflagration, as described in Subsection 13a2.1.9.

13a2.2.9.1 Initial Conditions

Hydrogen concentration in the TSV and TOGS prior to the initiating event is assumed to be at three percent. Additional initial conditions are described in Subsection 13a2.1.9.1.

13a2.2.9.2 Initiating Event

Potential initiating events are discussed in Subsection 13a2.1.9.1.

13a2.2.9.3 Sequence of Events

The accident sequence proceeds as follows:

1. A failure causes a single TOGS blower to fail, resulting in a complete loss of flow through the affected train and total loss of TSV dump tank flow.
2. The other TOGS blower continues to operate normally.

3. TRPS detects the loss of flow and executes an IU Cell Safety Actuation and IU Cell Nitrogen Purge.
4. The IU Cell Safety Actuation opens the TSV dump valves and NDAS HVPS breakers, terminating the irradiation process.
5. Hydrogen generation in the TSV and TSV dump tank continues due to radiolysis caused by delayed fission and decay radiation. Hydrogen evolution from solution occurs at an increased rate as solution voids collapse.
6. Within four seconds, N2PS is at full flow to the dump tank. Hydrogen and other gases are vented to PVVS through the combustible gas management system exhaust point. Gases pass through the PVVS carbon guard and carbon delay beds before being exhausted from the building at the safety-related exhaust point.
7. The remaining TOGS blower continues operation for a minimum of five minutes.
8. The combined action of the remaining TOGS blower and N2PS maintains the peak hydrogen concentration less than 7.7 percent; therefore, the peak pressure will not exceed the design pressure of the PSB if a deflagration occurs, and no radiological materials will be released. Detonations cannot occur because this peak concentration is less than the lower detonation limit.
9. As delayed fission and decay of short-lived radionuclides decline, the production and evolution of hydrogen declines following shutdown and draining of the TSV to the TSV dump tank. The N2PS continues to provide sweep gas diluting and removing any remaining hydrogen.

Safety Controls

The safety controls credited for prevention of accidents which may cause detonation or deflagration in the PSB are:

- TOGS capable of maintaining hydrogen concentration within design limits, assuming the worst case single active failure following IU trip (see Subsection 4a2.8.6)
- TOGS low-flow trips (TRPS function)
- TOGS oxygen sensor which detect incipient degradation or failure
- TOGS demister high temperature trips (TRPS function), which detect incipient degradation or failure
- N2PS
- Nitrogen in the N2PS is refilled every 72 hours while in use
- TSV fill line isolation valves mode-permissive interlock
- TSV overflow lines to the TSV dump tank
- TSV dump tank level sensors (TRPS function)
- TSV dump tank low flow sensors (TRPS function)
- TSV target solution dump on dump tank level sensors (TRPS function)
- PCLS expansion tank flame arrestor
- Radiation detection in RVZ1e exit from PCLS expansion tank
- Isolation valves in RVZ1e exit from PCLS expansion tank

13a2.2.9.4 Damage to Equipment

If hydrogen deflagration occurs at the peak calculated concentration of 7.7 percent, the PSB remains intact. Damage to other primary system components internal to TOGS in the affected train may occur; however, such damage will not result in any release of radiological material.

13a2.2.9.5 Radiological Source Terms

Because the PSB remains intact, there is no radiological source term for this accident sequence.

13a2.2.9.6 Radiological Consequences

Because the PSB remains intact, there are no radiological consequences for this accident sequence.

13a2.2.10 UNINTENDED EXOTHERMIC CHEMICAL REACTIONS OTHER THAN DETONATION

As discussed in Subsection 13a2.1.10, the potential for an unintended exothermic chemical reaction within the IF is unlikely. Therefore, there is no radiological consequence to the workers or the public.

Accident scenario consequences associated with the release of tritium gas are discussed in Subsection 13a2.2.12.

13a2.2.10.1 Initial Conditions

Initial accident conditions are described in Subsection 13a2.1.10.1.

13a2.2.10.2 Initiating Event

Scenario 1 – Uranium Metal-Water Reaction in the Neutron Multiplier Assembly

The initiating event is a small breach of the neutron multiplier cladding, allowing PCLS water into the cladding [ ]PROP/ECI.

The initiating events associated with unintended exothermic chemical reactions other than detonation are further discussed in Subsection 13a2.1.10.1.

13a2.2.10.3 Sequence of Events

Scenario 1 – Uranium Metal-Water Reaction in the Neutron Multiplier Assembly

The accident sequence proceeds as follows:

1. A small breach of the neutron multiplier cladding occurs, allowing PCLS water to enter the cladding [ ]PROP/ECI.
2. The water intrusion results in an exothermic uranium metal-water reaction, generating hydrogen.
3. The presence of [ ]PROP/ECI inhibits a potential deflagration.
4. Small amounts of hydrogen gas migrate into the PCLS and travel to the PCLS expansion tank, along with hydrogen normally generated in PCLS itself via radiolysis. The expansion tank is vented to RVZ1e to prevent hydrogen accumulation in that tank.

5. Small amounts of fission products from the multiplier migrate into the PCLS water. The presence of fission products in excess of normal operating levels is detected via the RVZ1e IU cell radiation monitors installed in the exhaust of the PCLS expansion tank.

Safety Controls

- The design of the neutron multiplier to inhibit deflagration is a safety control (including [ ][PROP/ECI]).

The sequences of events associated with unintended exothermic chemical reactions other than detonation are further discussed in Subsection 13a2.1.10.2.

13a2.2.10.4 Damage to Equipment

As discussed in Subsection 13a2.1.10, no damage beyond the initiating events is anticipated to occur as a result of unintended chemical reactions other than detonation.

13a2.2.10.5 Radiation Source Terms

Scenario 1 – Uranium Metal-Water Reaction in the Neutron Multiplier Assembly

Because a gross failure of the multiplier cladding is unlikely based on its design and a deflagration due to small leaks in the cladding is unlikely (as described in Subsection 13a2.1.10.2), a uranium metal-water reaction in the neutron multiplier assembly does not result in consequences to the worker or the public.

13a2.2.10.6 Radiological Consequences

Scenario 1 – Uranium metal-water reaction in the neutron multiplier assembly

Because a gross failure of the multiplier cladding is unlikely based on its design and a deflagration due to small leaks in the cladding is unlikely, there are no radiological consequences to the worker or the public from this event sequence.

13a2.2.11 SYSTEM INTERACTION EVENTS

As discussed in Subsection 13a2.1.11, no releases are expected to occur as a result of system interaction events. There are no consequences to the workers or the public from system interaction events, as discussed below. Accident consequences resulting from system interactions that are referenced to other subsections in Chapter 13 are evaluated in those subsections.

13a2.2.11.1 Initial Conditions

There are no unique initial conditions associated with system interaction events.

13a2.2.11.2 Initiating Event

Potential causes for system interaction events are described in Subsection 13a2.1.11.

13a2.2.11.3 Sequence of Events

Functional Interactions

<u>Loss of Off-Site Power</u>

LOOP events are described in Subsection 13a2.2.5.

<u>Reduction of Cooling</u>

Reduction of cooling events are described in Subsection 13a2.2.3, Subsection 13a2.2.5, and Subsection 13a2.2.6.

<u>Loss of Ventilation</u>

Postulated loss of ventilation scenarios do not result in radiological consequences based on the preventive controls described in Subsection 13a2.1.11.

Additional loss of ventilation scenarios are described in Subsection 13a2.2.5 and Subsection 13a2.2.6.

<u>Safety Controls</u>

The safety controls credited for prevention of accidents which may cause radiological consequences from a loss of ventilation are:

- Redundant and diverse TOGS instrumentation (e.g., low flow) that initiates a TRPS signal
- Redundant low and high PCLS temperature trip that initiates a TRPS signal

Spatial Interactions

<u>Fires</u>

Postulated fire scenarios in the IF are prevented by fire protection features identified in the fire hazards analysis (FHA), as described in Subsection 13a2.1.11.

Additional fire scenarios are discussed in Subsection 13a2.2.6 and Subsection 13a2.2.9.

<u>Safety Controls</u>

The safety controls credited for prevention of accidents which may cause radiological consequences from fires are:

- Fire protection measures: low combustible loading, fire detection, manual fire-fighting capabilities, and rated fire barriers to limit the potential for fire initiation and spread
- 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 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.

13a2.2.12.1 Tritium Release into an IU Cell

The release of tritium from an operating neutron driver assembly is analyzed as a DBA. The bounding scenario is described in Subsection 13a2.1.12.2 as NDAS Scenario 3, and the dose consequences are analyzed below.

13a2.2.12.1.1 Initial Conditions

Initial conditions for facility-specific events are described in Subsection 13a2.1.12.1.

13a2.2.12.1.2 Initiating Event

An internal NDAS vacuum boundary component fails and causes a pressurized release of tritium and SF_6 gas into the IU cell. Potential causes of the initiating event are discussed in Subsection 13a2.1.12.2.

13a2.2.12.1.3 Sequence of Events

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

1. The initiating event is a vacuum boundary component failure in the NDAS, which instantaneously releases tritium and SF_6 gas into the IU cell.
2. The IU cell becomes slightly pressurized due to the mass of released SF_6 gas.
3. Tritium is transported into the IF through penetrations in the confinement boundary.
4. Detection of high TPS target chamber supply pressure or high TPS target chamber exhaust pressure actuates the primary confinement boundary isolation valves and an irradiation unit 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. Tritium migrates to the IF through penetrations in the primary confinement boundary and is released to the environment.
6. Detection of high TPS target chamber supply pressure or high TPS target chamber exhaust pressure 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 within 10 minutes upon actuation of the radiation area monitor alarms.

Radiation transport is primarily driven by barometric breathing between the IU cell and the IF.

<u>Safety Controls</u>

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
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Additional safety controls credited for prevention of consequences of other NDAS scenarios described in Subsection 13a2.1.12.2 are:

- NDAS operating procedures require evacuation of deuterium and tritium from the NDAS and isolation of the deuterium and tritium supplies to the IU while in maintenance
- Proper installation of IU cell shielding following IU cell access is verified before operation of the neutron driver
- NDAS service cell (NSC) procedures prevent inadvertent operation of the NDAS while personnel are in the NSC

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 RAF 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 Glovebox

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 two TPS trains and instantaneously releases the tritium inventory into 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 high TPS confinement A/B/C 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 high TPS confinement A/B/C tritium monitors.
6. The radioactive material is then dispersed throughout the IF and exits the facility to the environment through building penetrations.
7. Detection of high TPS confinement A/B/C tritium actuates ventilation dampers between the RCA and the environment and minimizes the transport of radioactive material to the environment.
8. Personal dosimeters, local radiation alarms, and alarms in the facility control room notify facility personnel of radiation leakage.
9. 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 each TPS glovebox and the TPS room is constant. After the TPS room ventilation is isolated, radiation transport is driven by air exchange between 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.

<u>Safety Controls</u>

The safety controls credited for mitigation of this accident are:

- TPS room ventilation isolations
- Glovebox pressure control and VAC/ITS ventilation isolations
- TPS confinement A/B/C tritium monitors
- Tritium confinement boundary, as described in Section 6a2.2
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms
- Tritium release event recovery actions are completed within 10 days

In addition, TPS glovebox deflagration is prevented by:

- TPS glovebox gas space inerted with helium
- TPS 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 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 RAF 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.

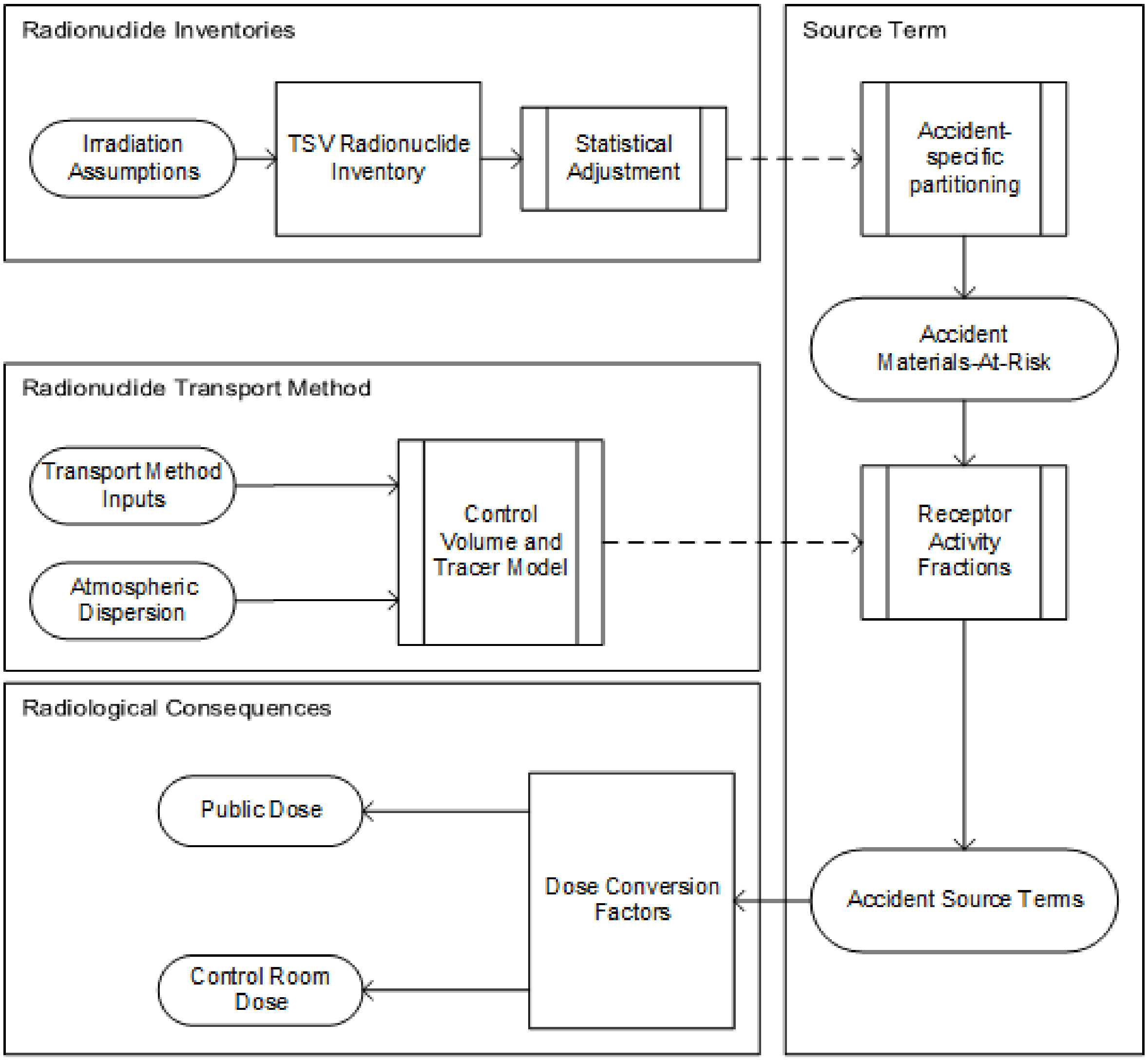
Table 13a2.2-1 – Summary of Radiation Transport Terms (Public)

Accident Category	Receptor Activity Fraction (RAF)				
	Nobles (30-day)	**Iodine (30-day)**	**Non-volatiles (30-day)**	**Tritium (10-day)**	**Tritium (30-day)**
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	1.29E-03	1.26E-04	9.88E-10	N/A	N/A
External Events (Subsection 13a2.2.6)	N/A	N/A	N/A	N/A	4.07E-04
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	1.41E-03	3.69E-04	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	4.07E-04
• Tritium Release into the Tritium Purification System Glovebox	N/A	N/A	N/A	1.78E-04	N/A

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

Accident Category	Receptor Activity Fraction (RAF) (30 days)			
	Nobles	**Iodine**	**Non-volatiles**	**Tritium**
Mishandling or Malfunction of Target Solution (Subsection 13a2.2.4)	8.41E-01	1.06E-01	6.49E-07	N/A
External Events (Subsection 13a2.2.6)	N/A	N/A	N/A	2.87E-01
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	9.92E-01	3.23E-01	0	N/A
Facility-Specific Events (Subsection 13a2.2.12)				
• Tritium Release into an IU Cell	N/A	N/A	N/A	2.87E-01
• Tritium Release into the Tritium Purification System Glovebox	N/A	N/A	N/A	1.08E-01 (10 days)

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 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)**
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)		
• Primary system boundary leak into an IU cell	440	771
Loss of Off-Site Power (LOOP) (Subsection 13a2.2.5)	No consequences	
External Events (Subsection 13a2.2.6)	292	588
Mishandling or Malfunction of Equipment (Subsection 13a2.2.7)	727	1940
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)		
• Tritium Release into an IU Cell	37	74
• Tritium Release into the Tritium Purification System Glove Box	798	1380

13a4 REFERENCES

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13b RADIOISOTOPE PRODUCTION FACILITY ACCIDENT ANALYSES

13b.1 RADIOISOTOPE PRODUCTION FACILITY ACCIDENT ANALYSIS METHODOLOGY

The accident analysis process for the radioisotope production facility (RPF) was conducted using the same methodology as the accident analysis in the irradiation facility (IF), described in Section 13a2.1. The radiological consequences were evaluated using the same methodology described in Section 13a2.2 for the IF.

13b.1.1 PROCESSES CONDUCTED OUTSIDE THE IRRADIATION FACILITY

The production of molybdenum-99 (Mo-99) and other fission products occurs in the IF. After the irradiation of the target solution is completed, the solution is transferred from the IF to the RPF and processed for radioisotope extraction and purification. Other processes occurring within the RPF include target solution processes for reuse, waste handling, and product packaging. These processes that occur within the RPF are evaluated via hazard identification and a process hazard analysis (PHA). The hazard identification process includes a review of potential radiological hazards, chemical hazards, and other facility hazards that might be present.

Process that are conducted in the RPF fall into the following categories:

- Operations with special nuclear material (SNM)
  - Irradiated target solution processed for radioisotope extraction
  - Irradiated target solution processed for reuse or for waste disposal
  - Operations with unirradiated SNM
- Radiochemical operations
- Operations with hazardous chemicals

The operations involving SNM include the uranium receipt and storage system (URSS), target solution preparation system (TSPS), the molybdenum extraction and purification system (MEPS), the iodine and xenon purification and packaging (IXP) system, the quality control and analytical testing laboratories (LABS), the target solution staging system (TSSS), the vacuum transfer system (VTS), the radioactive liquid waste storage (RLWS) system, the radioactive liquid waste immobilization (RLWI) system, and the radioactive drain system (RDS). The operations that do not involve SNM but pose a radiological or chemical hazard from radiochemical operations and operations with hazardous chemicals include the molybdenum isotope packaging system (MIPS), the process vessel vent system (PVVS), and the facility chemical reagent system (FCRS). Other systems in the RPF that do not have direct radiological or chemical hazards are evaluated for impact on the systems listed above.

The URSS receives, thermally oxidizes (if needed), repackages, and stores low-enriched uranium prior to target solution preparation in the TSPS. The URSS is classified both as an operation with unirradiated SNM and as an operation with hazardous chemicals. Because of the presence of uranium, the URSS poses a criticality, radiological, and chemical hazard.

The TSPS prepares low-enriched uranyl sulfate solution, which, once qualified for use, is referred to as target solution. The TSPS is classified both as an operation with unirradiated SNM and as an operation with hazardous chemicals. Because of the presence of uranium, the TSPS poses a criticality, radiological, and chemical hazard.

The MEPS separates the molybdenum from the irradiated target solution and purifies the resulting product. The extraction portion of the MEPS is an operation involving irradiated target solution processed for radioisotope extraction and contains significant quantities of uranium. Because of the presence of uranium, the MEPS extraction process is analyzed for criticality hazards. In addition, the extraction process involves radiological and chemical exposure hazards. The purification portion of MEPS, as well as isotope packaging operations in MIPS, are considered radiochemical operations, but pose a lesser hazard than extraction operations, because these processes are physically separated from the extraction operations and involve smaller quantities of radioactive material.

The IXP system separates iodine from acidic solutions and purifies the resulting product. The separation operations handle irradiated target solution processed for radioisotope extraction and contain significant quantities of uranium. Because of the presence of uranium, the IXP process has the potential for a criticality. In addition, the IXP has radiological and chemical exposure hazards.

The LABS are used to analyze samples of target solution, radioisotope products, and other process fluids. The operations in the LABS involve small amounts of SNM, radiochemicals, and hazardous chemicals. Because of the presence of uranium, the LABS are analyzed for criticality hazards. In addition, the LABS involve radiological and chemical exposure hazards.

The TSSS receives both irradiated target solution from the radioisotope extraction processes and unirradiated target solution from the TSPS. The TSSS allows for the target solution to be sampled prior to reuse or disposal, and stages target solution for transfer to the IF or the waste system. The system is categorized as irradiated target solution processed for reuse or waste disposal. Because of the presence of uranium, the TSSS has the potential for a criticality as well as radiological and chemical exposure hazards.

The VTS serves as the transfer system for irradiated target solution between RPF tanks and for transfers between the RPF and the IF. The system also provides the capability to sample tank contents in the TSSS and the RLWS. The system performs operations involving irradiated target solution processed for reuse or waste disposal. Because of the presence of uranium, the VTS has the potential for a criticality as well as radiological and chemical exposure hazards.

The RLWS serves as a waste system for solutions resulting from the processing of licensed material, and target solution batches or portions thereof that will no longer be used in facility processes. The RLWS involves operations with irradiated target solution processed for waste disposal. Because of the presence of uranium, the RLWS has the potential for a criticality. In addition, this process has radiological and chemical exposure hazards.

The RLWI serves as a waste immobilization system for solutions received from the RLWS. The RLWI involves operations with irradiated target solution processed for waste disposal. Because of the presence of uranium, the RLWI has the potential for a criticality. In addition, this process has radiological and chemical exposure hazards.

The RDS collects leakage and overflow of process fluids, including target solution, from process tanks and vessels and from hot cells. Fluids collected in the RDS can be returned to production or transferred to the RLWS for disposal. The RDS involves operations with irradiated target solution processed for reuse or for waste disposal. Because of the presence of uranium, the RDS has the potential for a criticality as well as radiological and chemical exposure hazards.

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 also defined to result in bounding radiological consequences for the SHINE facility.

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

The MHA 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.

<u>Scenario 1 - Spill of Target Solution in the Supercell (MEPS Column Misalignment)</u>

A spill of target solution in the supercell has the potential to release radioactive gases, aerosol, and particulates into the hot cell and ventilation system. Potential consequences of spilled target solution in the supercell include radiological dose. To mitigate the impact of spilled target solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (radiological ventilation zone 1 [RVZ1]) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to the engineered safety features actuation system (ESFAS) to isolate the affected cell and limit the amount of target solution introduced into the cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, hot cell inlet (radiological ventilation zone 2 [RVZ2]) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers, and ESFAS-controlled MEPS extraction pump breakers, VTS vacuum transfer pump breakers, and VTS vacuum break valves are provided to limit the amount of target solution introduced into the affected hot cell. This scenario is further described in Subsection 13b.2.4.1.

Scenario 2 - Spill of Target Solution in the Supercell (MEPS Overpressurization)

A spill of target solution in the supercell caused by MEPS overpressurization has the potential to release radioactive gases, aerosol, and particulates into the hot cell and ventilation system. Potential consequences of spilled target solution in the supercell include radiological dose. To prevent deflagrations, which may cause overpressure events, the nitrogen purge system (N2PS) automatically actuates on a failure of PVVS and is relied on to dilute hydrogen concentrations in tanks and vessels in the RPF. Additionally, target solution extraction pumps are provided pressure relief mechanisms. To mitigate the impact of spilled target solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell and limit the amount of target solution introduced into the cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers, and ESFAS-controlled. MEPS extraction pump breakers, VTS vacuum transfer pump breakers, and VTS vacuum break valves are provided to limit the amount of target solution introduced into the affected hot cell. This scenario is further described in Subsection 13b.2.4.1.

Scenario 3 - Spill of Molybdenum Eluate Solution in the Supercell (Overfill or Drop of Rotovap Flask)

A spill of the molybdenum solution in the MEPS purification cell may result in the release of radioactive gases, aerosol, and particulates into the hot cell and ventilation system. Potential consequences of spilled eluate solution in a hot cell include radiological dose. To mitigate the impact of spilled eluate solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, and hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers. The resulting sequence of events for this scenario is analogous to the MEPS eluate spill described in Subsection 13b.2.4.2.

Scenario 4 - Spill of Target Solution in the Supercell (IXP Column Misalignment)

A spill of target solution in the IXP extraction cell caused by IXP column misalignment has the potential to release radioactive gases, aerosol, and particulates into the hot cell and ventilation system. Potential consequences of spilled target solution in supercell include radiological dose. To mitigate the impact of spilled target solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell and limit the amount of target solution introduced into the cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers, and ESFAS-controlled IXP extraction pump breakers, VTS vacuum transfer pump breakers, and VTS vacuum break valves are provided to limit the amount of target solution introduced into the affected hot cell. This scenario is further described in Subsection 13b.2.4.1.

Scenario 5 - Spill of Target Solution in the Supercell (IXP Overpressurization)

A spill of target solution in the IXP extraction cell caused by IXP column overpressurization has the potential to release radioactive gases, aerosol, and particulates into the hot cell and ventilation system. Potential consequences of spilled target solution in the supercell include radiological dose. To prevent hydrogen deflagrations, which may cause overpressure events, the N2PS automatically actuations on a failure of PVVS and is relied on to dilute hydrogen concentrations in tanks and vessels in the RPF. Additionally, target solution extraction pumps are provided pressure relief mechanisms. To mitigate the impact of spilled target solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell and limit the amount of target solution introduced into the cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers, and ESFAS-controlled IXP extraction pump breakers, VTS vacuum transfer pump breakers, and VTS vacuum break valves are provided to limit the amount of target solution introduced into the affected hot cell. This scenario is further described in Subsection 13b.2.4.1.

Scenario 6 - Spill of Target Solution in the Supercell (Liquid Nitrogen Leak in IXP Hot Cell)

A liquid nitrogen leak in the IXP hot cell may damage components in the supercell and result in a spill of target solution in the hot cell, with the potential to release radioactive gases, aerosol, and particulates into the supercell and ventilation system. Potential consequences of spilled target solution in the supercell include radiological dose. To mitigate the impact of spilled target solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell and limit the amount of target solution introduced into the cell, hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers, hot cell outlet (RVZ1) ducts are equipped with carbon filters, and ESFAS-controlled IXP extraction pump breakers, VTS vacuum transfer pump breakers, and VTS vacuum break valves are provided to limit the amount of target solution introduced into the affected hot cell. This scenario is further described in Subsection 13b.2.4.1.

Scenario 7 - Spill of Iodine Solution in the Supercell (Overfill or Drop of Iodine Solution Bottle)

A spill of iodine eluate solution in the IXP cell results in the release of radioactive gases, aerosols, and particulates into the hot cell and ventilation system. Potential consequences of iodine solution spilling inside the IXP cell include radiological dose. To mitigate the impact of spilled iodine solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, and hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers. The resulting sequence of events for this scenario is analogous to the MEPS eluate spill described in Subsection 13b.2.4.2.

Scenario 8 - Spill of Target Solution in the Pipe Trench from a Single Pipe

A spill of target solution in the pipe trench results in the release of radioactive gases, aerosols, and particulates into the pipe trench. Potential consequences of spilled target solution inside the pipe trench include radiological dose. To mitigate the impact of spilled target solution, the following controls are applied: the pipe trench is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the pipe trench, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors (i.e., the RVZ1 and RVZ2 RCA exhaust radiation monitors) that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. This scenario is further described in Subsection 13b.2.4.3.

Scenario 9 - Spill of Target Solution in the Pipe Trench from Multiple Pipes

A spill of target solution in the pipe trench results in the release of radioactive gases, aerosols, and particulates into the below-grade confinement. Potential consequences of spilled target solution in the pipe trench include radiological dose. To mitigate the impact of spilled target solution resulting from the failure of multiple target solution-carrying pipes, the following controls are applied: the pipe trench is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the pipe trench, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. This scenario is further described in Subsection 13b.2.4.3.

Scenario 10 - Spill of Target Solution in a Tank Vault (Hold Tank Leak or Rupture)

A spill of target solution in a tank vault results in a release of radioactive gases, aerosols, and particulates into the tank vault. Potential consequences of target solution spilling in the tank vault include radiological dose. To mitigate the impact of spilled target solution, the following controls are applied: the tank vault is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the tank vault, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors (i.e., the RVZ1 and RVZ2 RCA exhaust radiation monitors) that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. This scenario is further described in Subsection 13b.2.4.4.

Scenario 11 - Spill of Target Solution in a Tank Vault (Hold Tank Deflagration)

A spill of target solution in a tank vault caused by a hold tank deflagration results a release of radioactive gases, aerosols, and particulates into the tank vault. Potential consequences of target solution spilling in the tank vault include radiological dose. To prevent a deflagration in the hold tank, the N2PS automatically actuates on a failure of PVVS and is relied upon to dilute hydrogen concentrations. To mitigate the impact of spilled target solution, the following controls are applied: the tank vault is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the tank vault, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors (i.e., the RVZ1 and RVZ2 RCA exhaust radiation monitors) that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. This scenario is further described in Subsection 13b.2.4.4.

Scenario 12 - Spill of Target Solution in a Tank Vault (Seismic Event)

A spill of target solution in a tank vault caused by a seismic event results in a release of radioactive gases, aerosols, and particulates into the tank vault. Potential consequences of target solution spilling in the tank vault include radiological dose. To prevent seismically caused damage, the process tanks are designed to withstand earthquakes. To mitigate the impact of spilled target solution, the following controls are applied: the tank vault is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the tank vault, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors (i.e., the RVZ1 and RVZ2 RCA exhaust radiation monitors) that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. This scenario is further described in Subsection 13b.2.4.4.

Scenario 13 - Spill of Molybdenum Eluate in the Supercell (Deflagration)

Loss of sweep gas flow from PVVS through the eluate tank in the supercell may result in a buildup of hydrogen in the eluate tank and a subsequent deflagration. A spill of molybdenum eluate caused by a deflagration in the eluate tank results in the release radioactive gases, aerosols, and particulates into the hot cell. Potential consequences of spilled eluate solution in a hot cell include radiological dose. To prevent deflagrations in tanks and vessels in the RPF, the N2PS automatically actuates upon a loss of PVVS and is relied upon to dilute hydrogen concentrations. To mitigate the impact of spilled eluate solution, the following controls are applied: the supercell is designed as a confinement boundary, hot cell exhaust ventilation (RVZ1) is equipped with radiation monitors (i.e., the RVZ1 supercell area 1-10 radiation monitors) that provide a signal to ESFAS to isolate the affected cell, hot cell outlet (RVZ1) ducts are equipped with carbon filters, and hot cell inlet (RVZ2) and outlet (RVZ1) ventilation ducts are equipped with ESFAS-controlled redundant isolation dampers. This scenario is further described in Subsection 13b.2.4.2.

Scenario 14 - Target Solution Leaking out of the Supercell (MEPS Preheater Tube Leak)

A leak in the MEPS extraction column preheater allows target solution to enter the hot water loop. Potential consequences of target solution leaking into the hot water loop, which is partially located outside of the supercell, include radiological dose. To prevent the target solution from circulating through the water loop, radiation monitors in the hot water loop detect target solution in-leakage and provide a signal to ESFAS to close the isolation valves on the supply and return of the hot water loop at the supercell boundary. This scenario was evaluated qualitatively and is not described in Section 13b.2 because the accident sequence is prevented.

Scenario 15 - Extraction Column Three-Way Valve Misalignment

A controller or operator error resulting in a misaligned three-way valve causes target solution to flow towards the chemical skid, which is located outside of the supercell. Potential consequences of this event include radiological dose. To prevent target solution from entering the chemical skid, a check valve in the chemical wash line prevents target solution backflow, and ESFAS monitoring of extraction three-way valve position causes the valves to de-energize and reposition whenever they are incorrectly aligned. This scenario was evaluated qualitatively and is not described in Section 13b.2 because the accident sequence is prevented.

Scenario 16 - Spill of Target Solution in a Valve Pit (Pipe Rupture or Leak)

A spill of target solution in a valve pit caused by a pipe rupture or leak results in a release of radioactive gases, aerosols, and particulates into the valve pit. Potential consequences of spilled target solution in the valve pit include radiological dose. To mitigate the consequences of spilled target solution, the following controls are applied: the valve pit is designed as a confinement boundary, RDS drains prevent the accumulation of target solution in the valve pit, the RDS sump tank liquid detection sensor detects fluid in-leakage and provides a signal to ESFAS to stop any in-process transfers of solution within the facility via the opening ESFAS-controlled VTS vacuum transfer pump breakers and VTS vacuum break valves, and the RVZ1 and RVZ2 building exhausts are equipped with radiation monitors (i.e., the RVZ1 and RVZ2 RCA exhaust radiation monitors) that provide a signal to ESFAS to isolate the building ventilation supply and exhaust dampers on high radiation. The resulting sequence of events for this scenario is analogous to the target solution leak in a pipe trench, which is further described in Subsection 13b.2.4.3.

Scenario 17 - Spill of Radioactive Liquid Waste in the RLWI Shielded Enclosure

A pipe leak or rupture in the RLWI shielded enclosure results in a release of radioactive gases, aerosols, and particulates into the enclosure. Potential consequences of a pipe leak or rupture in the RLWI shielded enclosure include radiological dose. To prevent unacceptable doses to workers, RLWS operating procedures provide limitations on concentration of waste solutions and require a minimum holdup time in the blending tank prior to transfer to the RLWI enclosure. This scenario is further described in Subsection 13b.2.4.5.

Scenario 18 - Heavy Load Drop onto RLWI Shielded Enclosure or Supercell

A crane failure or operator error resulting in a heavy load drop on the RLWI shielded enclosure or the supercell causes damage to the affected structure and internal equipment. Potential consequences of a heavy load drop on the RLWI shielded enclosure or supercell include radiological dose. To prevent a heavy load drop on the enclosure or the supercell, crane operation procedures include safe load paths to avoid the RLWI enclosure and supercell, and require suspension of supercell and RLWI activities during a heavy lift. The supercell damage scenario was evaluated qualitatively and is not described in Section 13b.2 because the accident sequence is prevented. The RLWI enclosure damage scenario is further described in Subsection 13b.2.4.5.

Scenario 19 - Heavy Load Drop onto a Tank Vault or Pipe Trench Cover Block

A crane failure or operator error resulting in a heavy load drop on a tank vault or pipe trench cover block causes a damage to the cover block and internal equipment. Potential consequences of a heavy load drop include radiological dose. To prevent damage to a cover block, the cover blocks have been designed to withstand a heavy load drop. This scenario was evaluated qualitatively and is not described in Section 13b.2 because the accident sequence is prevented.

13b.1.2.4 RPF Inadvertent Nuclear Criticality

Nuclear criticality safety (NCS) in the RPF is accomplished through the use of criticality safety controls to prevent criticality during normal and abnormal conditions. Each process that involves the use, handling, or storage of SNM is evaluated by the SHINE nuclear criticality safety staff under the requirements of the NCS program. Radiological consequences of criticality accidents are not included in the accident analysis because preventative controls are used to ensure criticality events are highly unlikely. Further discussion of the criticality safety bases for RPF processes is included in Section 6b.3.

13b.1.2.5 RPF Fire

The RPF was evaluated for internal fire risks based on the fire hazards analysis (FHA). The FHA documents the facility fire areas and each area was individually evaluated for fire risks. Internal facility fires are generally evaluated as an initiating event for the release of radioactive material and are included in the scenarios evaluated in Section 13a2.1 and this section. Two unique scenarios are described below and evaluated in detail in Section 13b.2.

The main production facility maintains a facility fire protection plan to reduce the risks of fires, as described in Section 9a2.3.

Scenario 1 - PVVS Carbon Delay Bed Fire (Beds 1, 2, or 3)

An upset or malfunction in the PVVS (high moisture or high temperature) results in ignition of the carbon media in a delay bed. A fire in carbon delay beds 1, 2, or 3 results in a release of the captured radioactive material into the PVVS downstream of the delay bed and to the environment via the facility exhaust stack. A release to the environment results in radiological exposure to the public. Release of radioactive material in excess of acceptable levels is prevented by the carbon delay bed exhaust temperature sensors. The temperature sensors provide a signal to ESFAS to close the PVVS carbon delay bed isolation valves for the affected carbon delay bed and bypass the affected bed in the event of high exhaust temperature indicative of a fire in a bed. The isolation valves for carbon delay beds 1, 2, and 3 function to prevent fire propagation to downstream carbon delay beds. This scenario is further described in Subsection 13b.2.6.1.

Scenario 2 - PVVS Carbon Delay Bed Fire (Beds 4, 5, 6, 7, or 8)

An upset or malfunction in the PVVS (high moisture or high temperature) results in ignition of the carbon media in a delay bed. A fire in carbon delay beds 4, 5, 6, 7, or 8 results in a release of the captured radioactive material into the PVVS downstream of the delay bed and to the environment via the facility exhaust stack. A release to the environment results in radiological exposure to the public. In the event that delay bed 4, 5, 6, 7, or 8 ignites, there is no automatic isolation in place to prevent propagation to downstream beds. It is assumed that carbon delay bed 4 ignites and the fire propagates to the remaining carbon delay beds, releasing the radionuclide inventory of the five beds. The resulting release of radioactive material is below acceptable levels. This scenario is further described in Subsection 13b.2.6.4.

Scenario 3 - PVVS Carbon Guard Bed Fire

An upset or malfunction in the PVVS (high moisture or high temperature) results in ignition of the carbon media in a guard bed. A fire in the guard bed results in a release of the captured radioactive material into the PVVS downstream of the guard bed, into the delay beds, and to the environment via the facility exhaust stack. A release to the environment results in radiological exposure to the public. Release of radioactive material in excess of acceptable levels is prevented by the downstream carbon delay beds, which reduce or delay radioisotope release. This scenario is further described in Subsection 13b.2.6.3.

13b.1.2.6 RPF Chemical Accidents

Potential chemical exposures in the RPF were evaluated to identify chemical hazards and necessary controls. The bounding inventories of chemicals used in the main production facility were identified and evaluated for exposure to workers and the public. Only exposure to uranium oxide presents a risk that exceeds the applicable evaluation criteria. This scenario is discussed further in Section 13b.3.

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 than 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 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

As described in Subsection 13a2.1.1, the postulated maximum hypothetical accident (MHA) for the SHINE facility is a 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.

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 radiological or chemical consequences. The facility-wide LOOP is further discussed in Subsection 13a2.1.5 and Subsection 13a2.2.5.

13b.2.3 EXTERNAL EVENTS

A seismic event was identified as an initiating event for several critical equipment malfunction accidents. The accident analysis associated with these events is presented below.

Severe weather was evaluated as an initiating event for the accident analysis. The main production facility structure is designed to withstand credible severe weather conditions and prevent damage to facility internal structures, systems, and components (SSCs). Based on the design of the main production facility structure, the risk associated with the potential release of radiological material or chemicals due to severe weather events is reduced to within acceptable limits.

Flooding was evaluated as an initiating event for the accident analysis. The main production facility has internal flood control measures to prevent the intrusion of water into areas that would be affected by the intrusion of water. The internal flood control measures are discussed below. Additionally, the local probable maximum precipitation event for the main production facility does not exceed the first floor entrance elevations. Consequently, there are no radiological consequences associated with external flooding events.

External fires were evaluated as an initiating event for the accident analysis. Generally, external fires do not have radiological consequences for the facility because safe shutdown conditions can be achieved without operator actions. The likelihood of significant external fires is highly unlikely because the facility is located on open terrain with no nearby prairie or forest and there are no natural gas lines that interact with the main production facility. The nearest natural gas line terminates approximately 60 feet from the main production facility. Vehicle fires were also considered. A vehicle fire at the loading dock presents a limited risk. The loading dock is designed to prevent combustible liquid spills from entering into the building, and the shipping/receiving area is separated from the loading dock itself. An external fire from a vehicle in the loading dock would be locally contained and does not produce radiological consequences.

13b.2.4 RPF CRITICAL EQUIPMENT MALFUNCTION

Several accident scenarios involve a release of radioactive solution into the supercell. Two types of solutions are present in the supercell, irradiated target solution and product eluate solutions. Spills of these solution are analyzed to determine their radiological consequences.

Operator errors were evaluated in the accident analysis as an initiating event for a number of critical equipment malfunctions. Operator errors and their effects are discussed in Subsection 13b.1.2.

13b.2.4.1 Spill of Target Solution in the Supercell

Initial Conditions

At the time of the initiating event, target solution is being pumped through the molybdenum extraction and purification system (MEPS) extraction cell. The target solution has decayed for [ ]PROP/ECI in the TSV dump tank prior to beginning the extraction process. The target solution irradiation assumptions are described in Section 13a2.2.

Initiating Event

An event causes a break in the MEPS piping between the extraction pump discharge and the extraction column. The break downstream of the pump discharge causes spray and aerosolization of the target solution without any extraction of isotopes by the extraction column. Potential initiating events for this scenario and analogous scenarios for the iodine and xenon purification (IXP) system cell are discussed further in Subsection 13b.1.2.3; Scenarios 1, 2, 4, 5, and 6.

Sequence of Events

1. A break in the MEPS piping between the extraction pump discharge and the extraction column occurs.
2. Aerosolized target solution sprays from the break into the hot cell, releasing radioactive material into the hot cell and causing the cell to become pressurized to the nominal pressure of the cell drain loop seal.
3. RVZ1 supercell area 2/6/7 radiation monitors in the hot cell exhaust ventilation detect high airborne radiation and cause the engineered safety features actuation system (ESFAS) to shut down the vacuum transfer system (VTS), shut down the extraction pump, and isolate the hot cell ventilation.

4. Leakage of radioactive material from the hot cell to the RPF and the environment through the ventilation dampers occurs, resulting in radiological consequences to workers and the public.

The maximum volume of spilled target solution in this accident scenario is limited by the volume of the vacuum lift tanks and installed piping of the MEPS. The ESFAS shutdown of the VTS prevents additional target solution from entering the hot cell after high radiation has been detected. The analyzed volume of target solution for this scenario is 30 liters, which is conservatively based on the volume of two vacuum lift tanks plus additional pipe volume.

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

- Supercell confinement boundary
- Radiological ventilation zone 1 (RVZ1) supercell area 2/6/7 radiation monitors
- Hot cell RVZ1 outlet carbon filters (radioiodine)
- Inlet (radiological ventilation zone 2 [RVZ2]) and outlet (RVZ1) ventilation isolation dampers
- MEPS or IXP extraction pump breakers
- VTS vacuum transfer pump breakers
- VTS vacuum break valves
- ESFAS Supercell Isolation function
- ESFAS VTS Safety Actuation function
- Target solution decay time requirements
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Damage to Equipment

The leak of target solution in the supercell does not cause subsequent damage to equipment.

Transport of Radioactive Material

The methods used to calculate radioactive material transport are described in Section 13a2.2. The LPF model terms used in this accident are provided in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is 30 liters of target solution from the IU at [ ]PROP/ECI post-shutdown. The action of the TOGS during this [ ]PROP/ECI period removes more than 67 percent of the iodine present in the solution at shutdown. It is conservatively assumed that 35 percent of the post-shutdown iodine inventory is released to the supercell during the accident. Additionally, partitioning fractions are applied to the noble gases present in target solution. Development of the accident source term for this scenario is discussed further in Section 13a2.2.

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 13b.2-2.

13b.2.4.2 Spill of Eluate Solution in the Supercell

Initial Conditions

At the time of the initiating event, eluate solution in the MEPS eluate tank is spilled onto the floor of the hot cell, releasing radioactive material into the hot cell atmosphere.

Initiating Event

An event causes the failure of the MEPS eluate tank, which results in a spill of eluate solution. Potential initiating events for this scenario and analogous scenarios for the purification and IXP cells are discussed further in Subsection 13b.1.2.3; Scenarios 3, 7, and 13.

Sequence of Events

1. A break in the MEPS eluate tank occurs.
2. Eluate solution spills from the tank into the hot cell, releasing radioactive material into the hot cell and causing the cell to become pressurized to the nominal pressure of the cell drain loop seal.
3. RVZ1 supercell area 3/5/8/10 radiation monitors in the hot cell exhaust ventilation detect high airborne radiation and cause ESFAS to isolate hot cell ventilation.
4. Leakage of radioactive material from the hot cell to the RPF and the environment through the ventilation dampers occurs, resulting in radiological consequences to workers and the public.

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

- Supercell confinement boundary
- RVZ1 supercell area 3/5/8/10 radiation monitors
- Hot cell RVZ1 outlet carbon filters (radioiodine)
- Inlet (RVZ2) and outlet (RVZ1) ventilation isolation dampers
- ESFAS Supercell Isolation function
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Damage to Equipment

The leak of target solution in the supercell does not cause subsequent damage to equipment.

Transport of Radioactive Material

The methods used to calculate radioactive material transport are described in Section 13a2.2. The LPF model terms used in this accident are provided in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is the extraction column eluate, which contains radionuclides from one entire target solution batch. The initial MAR is partitioned by the extraction column to produce the accident-specific MAR. Accident-specific partitioning factors are applied to the irradiated target solution batch as described in Section 13a2.2. Development of the accident source term for this scenario is discussed further in Section 13a2.2.

Radiological Consequences

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.2.4.3 Spill of Target Solution in the RPF Pipe Trench

Initial Conditions

A batch of irradiated target solution is being transferred within the RPF pipe trench. The target solution has been irradiated using the assumptions in Section 13a2.2 and has been held for decay in the TSV dump tank for [ ]PROP/ECI.

Initiating Event

An event causes a single pipe or multiple pipes containing target solution to break in the pipe trench. Potential initiating events for this scenario and the analogous scenario for a spill in a valve pit are discussed further in Subsection 13b.1.2.3; Scenarios 8, 9, and 16.

Sequence of Events

1. A single pipe or multiple pipes containing target solution within the pipe trench break, spilling target solution into the trench.
2. The target solution collects on drip pans in the trench and drains to the radioactive drain system (RDS).
3. The RDS sump tank liquid detection provides a signal to ESFAS, initiating a VTS Safety Actuation, and limiting the MAR.
4. Radioactive material is released into the pipe trench atmosphere.
5. A portion of the released material leaks through the process confinement boundary (trench cover) into the RPF and the environment, resulting in radiological consequences to workers and the public.

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

- Process confinement boundary (trench or pit cover and cover seal)
- RDS liquid detection
- ESFAS VTS Safety Actuation
- Target solution decay time requirements
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Damage to Equipment

The leak of target solution into the pipe trench does not cause further damage to equipment.

Transport of Radioactive Material

The methods used to calculate radioactive material transport are described in Section 13a2.2. The LPF model terms used in this accident are provided in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is a batch of target solution from the IU at [ ]PROP/ECI post-shutdown. The action of the TOGS during this [ ]PROP/ECI period removes more than 67 percent of the iodine present in the solution at shutdown. It is conservatively assumed that 35 percent of the post-shutdown iodine inventory is released to the pipe trench during the accident. Additionally, partitioning fractions are applied to the noble gases present in target solution. Development of the accident source term for this scenario is discussed further in Section 13a2.2.

Radiological Consequences

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.2.4.4 Spill of Target Solution from a Tank

A spill of target solution from any of the below-grade hold or storage tanks results in a release of target solution into the associated tank vault. Radionuclides from the target solution become airborne and migrate into the RPF and the environment.

The liquid waste blending tanks contain large volumes of dilute target solution that has already undergone extraction and processing. The accident analysis considers freshly-irradiated target solution that has not undergone processing and bounds the failure of the liquid waste blending tank.

Initial Conditions

A full batch of target solution is present in a target solution hold or storage tank at the time of the initiating event. The target solution has been irradiated using the assumptions in Section 13a2.2 and has been held for decay for [ ]PROP/ECI post-shutdown, which accounts for [ ]PROP/ECI of hold time in the TSV dump tank and [ ]PROP/ECI of processing time.

Initiating Event

An event causes a tank containing target solution to break and leak. Potential initiating events are discussed further in Subsection 13b.1.2.3; Scenarios 10, 11, and 12.

Sequence of Events

1. A tank containing target solution breaks, spilling target solution into the tank vault.
2. The target solution collects on the drip pans in the vault and drains to the RDS.
3. Radioactive material is released into the pipe trench atmosphere

4. A portion of the released material leaks through the process confinement boundary (vault cover) into the RPF and the environment, resulting in radiological consequences to workers and the public.

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

- Process confinement boundary (tank vault plugs and seals)
- Target solution decay time requirements
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Damage to Equipment

The leak of target solution into the tank vault does not cause further damage to equipment.

Transport of Radioactive Material

The methods used to calculate radioactive material transport are described in Section 13a2.2. The LPF model terms used in this accident are listed in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is a full batch of target solution from the IU at [ ][PROP/ECI] post-shutdown. The action of the TOGS during the [ ][PROP/ECI] hold-up period in the dump tank removes more than 67 percent of the iodine present in the solution at shutdown. It is assumed that 35 percent of the post-shutdown iodine inventory is released to the tank vault during the accident. Additionally, partitioning fractions are applied to the noble gases present in target solution. Development of the accident source term for this scenario is discussed further in Section 13a2.2.

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 13b.2-2.

13b.2.4.5 Spill of Waste Solution in RLWI

Initial Conditions

A 380-liter batch of waste solution (diluted target solution) is present in the radioactive liquid waste immobilization (RLWI) system immobilization feed tank at the time of the initiating event. The volume of solution in this scenario is based on the volume of the immobilization feed tank with a conservative scaling factor to account for the highest allowable concentration of radionuclides. The waste solution has been irradiated using the assumptions in Section 13a2.2 and has been held for decay for 35 days post-shutdown. The post-shutdown hold time is based on the minimum hold time needed to reduce waste activity to within dose consequence limits and establishes an administrative control. Expected hold times for waste solution are significantly longer than 35 days.

Initiating Event

An event causes the immobilization feed tank or RLWI system piping containing waste solution to break and leak within the RLWI enclosure. Potential initiating events are discussed further in Subsection 13b.1.2.3; Scenarios 17 and 18.

Sequence of Events

1. The immobilization feed tank breaks and spills waste solution into the RLWI enclosure.
2. The waste solution collects on the floor of the enclosure and leaks into the RPF and environment, resulting in radiological consequences to workers and the public.

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

- Waste solution holdup times in the radioactive liquid waste storage (RLWS) system before processing in RLWI
- Concentration controls applied to waste solutions
- Heavy load drop controls described in Subsection 13b.1.2.3
- Facility personnel evacuate the immediate area within 10 minutes after receipt of electronic dosimeter or local radiation alarms

Damage to Equipment

The leak of waste solution into the RLWI enclosure does not cause further damage to equipment.

Transport of Radioactive Material

The LPF and airborne release fraction (ARF) values used in this scenario are set at 1.0 instead of using the LPF model values described in Section 13a2.2. The LPF model terms used in this accident are provided in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is 380 liters of waste solution at 35 days post-shutdown. The concentration of radionuclides for the waste solution is determined by multiplication of the ratio of the maximum uranium concentration permitted in the RLWI system to the nominal uranium concentration of target solution. The action of the TOGS during the [ ]$^{\text{PROP/ECI}}$ period when the original target solution was held in the dump tank removes more than 67 percent of the iodine present in the solution at shutdown. It is assumed that 35 percent of the post-shutdown iodine inventory is released to the RLWI enclosure during the accident. Additionally, partitioning fractions are applied to the noble gases present in target solution. Development of the accident source term for this scenario is discussed further in Section 13a2.2.

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 13b.2-2.

13b.2.5 RPF INADVERTENT NUCLEAR CRITICALITY

Inadvertent nuclear criticality events were evaluated in the accident analysis using the same methodology as non-criticality accidents. Nuclear criticality safety is achieved through the use of preventative controls throughout the RPF, which reduces the likelihood of a criticality accident to highly unlikely (or better). Preventative controls were selected based on nuclear criticality safety evaluations conducted under the facility nuclear criticality safety program. The nuclear criticality safety program and the criticality safety basis for RPF processes is described in Section 6b.3.

13b.2.6 RPF FIRE

Facility fires were evaluated in the accident analysis. Facility fire scenarios and their effects are discussed in Subsection 13b.1.2.5. Two facility fire scenarios were evaluated for radiological consequences.

13b.2.6.1 PVVS Carbon Delay Bed Fire (Beds 1, 2, or 3)

Initial Conditions

The PVVS is operating normally, with nominal flow through a carbon delay bed.

The affected carbon delay bed contains noble gases from RPF process streams. The MAR in this scenario is a combination of gases from eight IUs with various modifiers applied to account for decay and processing capacity of target solution batches in the supercell. The purge volumes and decay times used provide a maximum radiological inventory that could be present on an individual bed.

Initiating Event

An upset or malfunction in the PVVS results in high moisture or high temperature flow through the carbon delay bed. The high moisture or high temperature results in ignition of the carbon delay bed absorber media. Potential initiating events are discussed further in Subsection 13b.1.2.5, Scenario 1.

Sequence of Events

1. Ignition of the carbon delay bed occurs, resulting in an exothermic release of stored radioactive material to the PVVS downstream of the delay bed.
2. After a period of time, the entire radioactive material inventory of the affected carbon delay bed is released to the downstream delay beds and to the environment through the PVVS and facility stack.
3. Fire conditions are detected by the exhaust temperature sensors, which send a signal to the ESFAS.
4. ESFAS initiates a Carbon Delay Bed Isolation for the affected carbon delay bed to prevent fire propagation to downstream beds.

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

- PVVS carbon delay bed 1, 2, and 3 temperature sensors
- PVVS carbon delay bed 1, 2, and 3 isolation valves
- ESFAS carbon delay bed 1, 2, and 3 isolation function

Damage to Equipment

The occurrence of fire damages the affected carbon delay 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 LPF model terms used in this accident are provided in Table 13b.2-1. The release rate of material from the bed is based on a conservative calculation of the effects of a fire in the delay bed. It is assumed that radionuclide inventories are released from the bed prior to isolation. Material released from the delay bed passes through the remaining delay beds, with a reduced holdup efficiency, prior to being released to the environment.

Radiation Source Terms

The initial MAR for this scenario is a portion of the noble 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 noble gas inventory is produced by decay of fission products and continuously evolved from the target solution and through the TOGS during operations. The MAR uses selected time intervals for the most recent purges (i.e., [ ]PROP/ECI) to account for the processing capacity of target solution batches in the supercell for the combined eight IUs. The gases accumulate in the carbon delay bed and decay. The MAR assumes the combined noble gas inventory produced by eight IUs over approximately [ ]PROP/ECI of irradiation with the most recent purges of [ ]PROP/ECI. Partitioning fractions for noble gases are used to describe the quantities of noble gases in solution that move to the RPF to account for removal during movement of solution. Additional decay time is applied based on which of the carbon delay beds is assumed to be impacted by the fire event.

Radiological Consequences

The radioactive material is contained in the PVVS system and does not result in material being released into the RPF. 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.2.6.2 PVVS Carbon Delay Bed Fire (Beds 4, 5, 6, 7, or 8)

Initial Conditions

The PVVS is operating normally, with nominal flow through a carbon delay bed.

The affected carbon delay beds contain noble gases from RPF process streams. The MAR in this scenario is a combination of gases from eight IUs with various modifiers applied to account for decay and processing capacity of target solution batches in the supercell. The purge volumes and decay times used provide a maximum radiological inventory that could be present on an individual bed.

Initiating Event

An upset or malfunction in the PVVS results in high moisture or high temperature flow through the carbon delay bed. The high moisture or high temperature results in ignition of the carbon delay bed absorber media. Potential initiating events are discussed further in Subsection 13b.1.2.5, Scenario 2.

Sequence of Events

1. Ignition of the carbon delay bed 4, 5, 6, 7, or 8 occurs, resulting in an exothermic release of stored radioactive material.
2. The fire propagates to carbon delay beds that are downstream of the bed where the fire originated.
3. After a period of time, the entire radioactive material inventory of affected beds is released to the environment through the PVVS and facility stack.

No components are credited for mitigation of the dose consequences for this accident. The consequences of unmitigated release of delay beds 4, 5, 6, 7, and 8 inventories are below the SHINE Safety Criteria.

Damage to Equipment

The occurrence of fire damages the affected carbon delay beds and eliminates their 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 LPF model terms used in this accident are provided in Table 13b.2-1.

Radiation Source Terms

The initial MAR for this scenario is a portion of the noble 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 noble gas inventory is produced by decay of fission products and continuously evolved from the target solution and through the TOGS during operations. The MAR uses selected time intervals for the most recent purges (i.e., [ ]PROP/ECI) to account for the processing capacity of target solution batches in the supercell for the combined eight IUs. The gases accumulate in the carbon delay bed and decay. The MAR assumes the combined noble gas inventory produced by eight IUs over approximately [ ]PROP/ECI of irradiation with the most recent purges of [ ]PROP/ECI. Partitioning fractions for noble gases are used to describe the quantities of noble gases in solution that move to the RPF to account for removal during movement of solution. Additional decay time is applied based on which delay bed is assumed to be impacted by the fire event.

Radiological Consequences

The radioactive material is contained in the PVVS system and does not result in material being released to the RPF. 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.2.6.3 PVVS Carbon Guard Bed Fire

Initial Conditions

The PVVS is operating normally, with nominal flow through a carbon guard bed.

The affected carbon guard bed contains iodine from RPF process streams. The MAR in this scenario is a combination of iodine from eight IUs with various modifiers applied to account for decay and processing capacity of target solution batches in the supercell.

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 adsorber material. Potential initiating events are discussed further in Section 13b.1.2.5, Scenario 3.

Sequence of Events

1. Ignition of the carbon guard bed occurs, resulting in an exothermic release of stored radioactive material to the PVVS downstream of the guard bed.
2. Radioactive material is captured by the downstream carbon delay bed and filtered. One percent of the released radioactive material is released through PVVS and the facility stack to the environment.

The component credited for mitigation of the dose consequences for this accident is:

- PVVS delay bed filtration

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 LPF model terms used in this accident are provided in Table 13b.2-1. For this accident, the guard bed inventory is assumed to be instantly transported to the delay bed. The delay bed is credited to reduce the release of material by 99 percent with no credit taken for carbon guard bed isolation functions.

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 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 material being released into the RPF.

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.

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

Accident Scenario	Radionuclide Group	Receptor Activity Fraction (RAF)
Spill of Target Solution in the Supercell	Nobles	1.00E-03 Public 6.56E-01 Worker
	Iodine	1.49E-06 Public 1.40E-03 Worker
	Non-Volatile	1.38E-07 Public 9.40E-05 Worker
Spill of Eluate Solution in the Supercell	Nobles	1.04E-03 Public 6.96E-01 Worker
	Iodine	1.88E-06 Public 1.77E-03 Worker
	Non-Volatile	1.52E-07 Public 1.06E-04 Worker
Spill of Target Solution in the RPF Pipe Trench	Nobles	1.27E-04 Public 7.71E-02 Worker
	Iodine	2.49E-07 Public 2.26E-04 Worker
	Non-Volatile	1.11E-08 Public 6.71E-06 Worker
Spill of Target Solution from a Tank	Nobles	1.36E-04 Public 8.24E-02 Worker
	Iodine	1.61E-08 Public 1.51E-05 Worker
	Non-Volatile	1.18E-08 Public 7.18E-06 Worker
Spill of Waste Solution in RLWI	Nobles	5.66E-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

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

Accident Scenario	Radionuclide Group	Receptor Activity Fraction (RAF)
PVVS Carbon Delay Bed Fire (Beds 1/2/3)	Nobles	5.66E-03 Public 6.70E+00 Worker
PVVS Carbon Delay Bed Fire (Beds 4/5/6/7/8)	Nobles	5.66E-03 Public 6.70E+00 Worker
PVVS Carbon Guard Bed Fire	Iodine	5.66E-03 Public 6.70E+00 Worker

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

Accident Scenario	Public Dose TEDE (mrem)	Worker Dose TEDE (mrem)
Spill of Target Solution in the Supercell	42	76
Spill of Eluate Solution in the Supercell	88	122
Spill of Target Solution in the RPF Pipe Trench	22	40
Spill of Target Solution from a Tank	24	42
Spill of Waste Solution in RLWI	557	1880
PVVS Carbon Delay Bed Fire (Beds 1/2/3)	117	8
PVVS Carbon Delay Bed Fire (Beds 4/5/6/7/8)	686	48
PVVS Carbon Guard Bed Fire	546	1390

13b.3 ANALYSES OF ACCIDENTS WITH HAZARDOUS CHEMICALS

The probability and consequences of accidents resulting in a hazardous chemical release for chemical hazards that are under NRC regulatory jurisdiction are minimized by considering such chemical hazards in the SHINE Safety Analysis (SSA), as described in Section 13a2.

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 (see Subsection 2.2.3.1.3). 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

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.

A consequence analysis for the public and nearest residence was performed using the PAVAN (An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations) computer code (USNRC, 1982). The 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 worker, the 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 total source term 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 (AEGLs), 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 controls 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 controls 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.
- Personnel evacuate within two minutes after chemical spills within the URSS and TSPS rooms.

These credited chemical process safety controls are incorporated into the technical specifications as described in Section 13a2.

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

Chemical Name	Location	Inventory (kg)
Alpha-Benzoin Oxime	Subgrade Waste Tanks	4.48E-01
	Supercell	1.00E-02
	RLWI Tank	3.28E-03
Ammonium Hydroxide	Supercell	2.17E-02
Ammonium Nitrate	Subgrade Waste Tanks	1.81E+01
	Supercell	4.96E-02
	RLWI Tanks	1.32E-01
[ ]PROP/ECI	TSPS Room	6.20E-01
	Target Solution Storage	2.79E+00
	Subgrade Waste Tanks	1.86E-01
	Supercell	3.10E-01
	RLWI Tank	2.70E-03
Hydrochloric Acid (38 wt.%)	Subgrade Waste Tanks	1.92E-02
	Supercell	1.00E-02
	RLWI Tank	1.41E-04
Hydrogen Peroxide (30 wt.%)	TSPS Room	2.66E+00
	Supercell	1.00E-02
Nitric Acid (70 wt.% in chemical storage)	Subgrade Waste Tanks	1.79E+01
	Supercell	9.49E-01
	RLWI Tank	1.31E-01
Potassium Hexachlororuthenate	Subgrade Waste Tanks	7.47E-03
	Supercell	1.00E-02
	RLWI Tank	5.46E-05

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

Chemical Name	Location	Inventory (kg)
Potassium Permanganate	Subgrade Waste Tanks Supercell RLWI Tank	4.73E-01 1.00E-02 3.46E-03
Rhodium Chloride	Subgrade Waste Tanks Supercell RLWI Tank	8.96E-03 1.00E-02 6.55E-05
Silver Nitrate	Supercell	1.00E-02
Sodium Hydroxide (50.5 wt.%)	Supercell	5.17E-01
Sodium Iodide	Supercell	1.00E-02
Sodium Sulfite (98 wt. %)	Subgrade Waste Tanks Supercell RLWI Tank	3.12E+00 1.00E-01 2.28E-02
Sulfuric Acid	TSPS Room Target Solution Storage Subgrade Waste Tanks Supercell RLWI Tank	9.67E+00 5.46E+01 5.62E+02 6.07E+00 4.49E+00
Uranium Metal	URSS Room	6.20E+02
Uranium Oxide	URSS Room TSPS Room	7.31E+02 8.60E+00
Uranyl Peroxide	TSPS Room	1.15E+01

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

Chemical Name	Location	Inventory (kg)
Uranyl Sulfate	TSPS Room	1.91E+02
	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 2)

Hazardous Chemical (Release Location)	MAR (kg)	Source Term (mg)	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) (mg/m^3)	Nearest Residence (788 m) (mg/m^3)
Alpha-Benzoin Oxime (Subgrade Waste Tanks)	0.0688	1.38	0.49	5.4	32	1.30E-03/7.68E-05	1.30E-05	8.50E-07
Ammonium Hydroxide (Supercell)	0.1(b)	2490	13	140	840	1.53E+00/1.39E-01	2.89E-02	1.89E-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/ECI (TSPS Room)	0.744	0.76(c)/74.4(c)	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	1.26E-02/3.15E-03	3.57E-05	2.34E-06
Hydrochloric Acid (Supercell)	0.038(b)	1380	2.7	33	150	7.14E-01/7.71E-02	1.90E-02	1.24E-03
Hydrogen Peroxide (TSPS Room)	3.2	1380	14	70	140	1.69E-01/7.71E-02	2.24E-03	1.47E-04
Nitric Acid (Subgrade Waste Tank)	2.7(d)	4820	0.41	62	240	2.55E+00/2.69E-01	7.91E-03	5.19E-04
Potassium Hexachloro-ruthenate (Supercell)	0.012	0.24	0.5	2	20	2.27E-04/1.34E-05	2.26E-06	1.48E-07
Potassium Permanganate (Subgrade Waste Tanks)	0.0727	1.45	8.6	14	150	1.37E-03/8.12E-05	1.37E-05	8.99E-07
Rhodium Chloride(e) (Supercell)	0.012	0.24	1.68	18.5	110	2.27E-04/1.34E-05	2.26E-06	1.48E-07
Silver Nitrate (Supercell)	0.012	0.24	0.05	0.9	5	2.27E-04/1.34E-05	2.26E-06	1.48E-07
Sodium Hydroxide (Supercell)	0.620	12.4	0.5	5	50	1.17E-02/6.93E-04	1.17E-04	7.67E-06
Sodium Iodide (Supercell)	0.012	0.24	13	140	860	2.27E-04/1.34E-05	2.26E-06	1.48E-07

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

Hazardous Chemical (Release Location)	MAR (kg)	Source Term (mg)	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) (mg/m^3)	Nearest Residence (788 m) (mg/m^3)
Sodium Sulfite (Subgrade Waste Tanks)	0.478	9.55	11	120	710	9.02E-03/5.33E-04	9.01E-05	5.91E-06
Sulfuric Acid (Subgrade Waste Tanks)	78.0	1560	0.2	8.7	160	1.47E+00/8.71E-02	1.47E-02	9.65E-04
Uranium Metal[f] (URSS Room)	7.8	0	0.6	5	30	0.00E+00/0.00E+00	0.00E+00	0.00E+00
Uranium Oxide (URSS Room)	40.0	2400	0.68	10	30	2.27E+00/9.99E+00	2.26E-02	1.48E-03
Uranyl Peroxide[g] (TSPS Room)	6.84	1368	0.94	10.4	62	1.29E+00/5.70E+00	1.29E-02	8.46E-04
Uranyl Sulfate[g] (TSPS Room)	191.2	235[c]/19120[c]	0.92	10.2	61	3.91E+00/1.07E+00	1.11E-02	7.25E-04

a. 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. MAR increased to the minimum mass that ALOHA can model for a puddle release.
c. 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. Based on largest capacity subgrade waste tank.
e. 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. Uranium metal is stored as solid pieces. Therefore, there is no hazard from dropping solid metal pieces.
g. 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/m^3, 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

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