NORTHWEST MEDICAL ISOTOPES

Chapter 13.0 – Accident Analysis

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 3
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Acronyms and Abbreviations

^{99}Mo	molybdenum-99
^{99m}Tc	technetium-99m
^{235}U	uranium-235
^{241}Am	americium-241
AAC	augmented administrative control
AC	administrative control
ACI	American Concrete Institute
AEC	active engineered control
AEGL	Acute Exposure Guideline Level
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALOHA	areal locations of hazardous atmospheres
ARF	airborne release fraction
ASCE	American Society of Civil Engineers
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
DAC	derived air concentration
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
EDE	effective dose equivalent
EOI	end of irradiation
ETA	event tree analysis
FEMA	Federal Emergency Management Agency
FMEA	failure modes and effects analysis
FTA	fault tree analysis
HAZOP	hazards and operability
HEGA	high-efficiency gas adsorption
HEPA	high-efficiency particulate air
HIC	high-integrity canister
HNO_3	nitric acid
HVAC	heating, ventilation, and air conditioning
IBC	International Building Code
IROFS	items relied on for safety
IRU	iodine removal unit
ISA	integrated safety analysis
ISG	Interim Staff Guidance
IX	ion exchange
LEU	low enriched uranium
LPF	leak path factor
MAR	material at risk
Mo	molybdenum
MURR	University of Missouri Research Reactor
NaOH	sodium hydroxide
NDA	nondestructive assay
NIOSH	National Institute for Occupational Safety and Health
NO_x	nitrogen oxide

NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NWMI	Northwest Medical Isotopes, LLC
NWS	National Weather Service
OSTR	Oregon State University TRIGA Reactor
OSU	Oregon State University
P&ID	piping and instrumentation drawing
PEC	passive engineered control
PFD	process flow diagram
PHA	preliminary hazards analysis
PMP	probable maximum precipitation
QRA	quantitative risk assessment
RASCAL	Radiological Assessment System for Consequence Analysis
RF	respirable fraction
RPF	Radioisotope Production Facility
RSAC	Radiological Safety Analysis Code
SNM	special nuclear material
SSC	structures, systems, and components
ST	source term
TCE	trichloroethylene
TEDE	total effective dose equivalent
U	uranium
U.S.	United States
UN	uranyl nitrate

Units

°C	degrees Celsius
°F	degrees Fahrenheit
Ci	curie
Cm	centimeter
ft	feet
ft^3	cubic feet
g	gram
hr	hour
$in.^2$	square inch
kg	kilogram
km	kilometer
km^2	square kilometer
L	liter
lb	pound
m	meter
M	molar
m^3	cubic meter
mg	milligram
mi	mile
mi^2	square mile
mil	thousandth of an inch
min	minute
mrem	millirem
oz	ounce
ppm	parts per million
rem	roentgen equivalent man
sec	second
Sv	sievert
wk	week
wt%	weight percent
yr	year

13.0 RADIOISOTOPE PRODUCTION FACILITY ACCIDENT ANALYSIS

The proposed action is the issuance of a U.S. Nuclear Regulatory Commission (NRC) Construction Permit and Operating License under Title 10, *Code of Federal Regulations*, Part 50 (10 CFR 50) "Domestic Licensing of Production and Utilization Facilities," and provisions of 10 CFR 70, "Domestic Licensing of Special Nuclear Material," and 10 CFR 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," that would authorize Northwest Medical Isotopes, LLC (NWMI) to construct and operate a molybdenum-99 (^{99}Mo) Radioisotope Production Facility (RPF) at a site located in Columbia, Missouri. The RPF is being designed to have a nominal operational processing capability of one batch per week of up to [Proprietary Information].

The primary mission of the RPF will be to recover and purify radioactive ^{99}Mo generated via irradiation of low-enriched uranium (LEU) targets in off-site non-power reactors. The purified ^{99}Mo will be packaged and transported to medical industry users where the radioactive decay product, technetium-99m (^{99m}Tc), can be employed as a valuable resource for medical imaging.

This section analyzes potential hazards and accidents that could be encountered in the RPF during operations involving special nuclear material (SNM) (irradiated and unirradiated), radioisotope recovery and purification, and the use of hazardous chemicals relative to these radiochemical processes. Irradiation services and transportation activities are not analyzed in this chapter.

This chapter evaluates the various processing and operational activities at the RPF, including:

- Receiving LEU from U.S. Department of Energy (DOE)
- Producing LEU target materials and fabrication of targets
- Packaging and shipping LEU targets to the university reactor network for irradiation
- Returning irradiated LEU targets for dissolution, recovery, and purification of ^{99}Mo
- Recovering and recycling LEU to minimize radioactive, mixed, and hazardous waste generation
- Treating/packaging wastes generated by RPF process steps to enable transport to a disposal site

Chapter Organization

Section 13.1 describes hazard and accident analysis methodologies applied to the RPF integrated safety analysis (ISA) (Section 13.1.1). Section 13.1.2 identifies the accident initiating events, and Section 13.1.3 summarizes the results of the RPF preliminary hazards analysis (PHA) (NWMI-2015-SAFETY-001, *NWMI Radioisotope Production Facility Preliminary Hazards Analysis*). The PHA discussion in Section 13.1.3 identifies the accident scenarios that required further evaluation.

Section 13.2 presents analyses of radiological and criticality accidents, including:

- Section 13.2.1 (Reserved)
- Section 13.2.2 discusses spills and spray accidents
- Section 13.2.3 discusses dissolver offgas accidents
- Section 13.2.4 discusses leaks into auxiliary systems accidents
- Section 13.2.5 discusses loss of electrical power
- Section 13.2.6 discusses natural phenomena accidents
- Section 13.2.7 identifies the additional accident sequences evaluated and associated items relied on for safety (IROFS)

Section 13.3 presents bounding accidents involving hazardous chemicals.

The data presented in the following subsections are based on a comprehensive PHA, conservative assumptions, draft quantitative risk assessments (QRA), and scoping calculations. These items provide an adequate basis for the construction application.

13.1 ACCIDENT ANALYSIS METHODOLOGY AND PRELIMINARY HAZARDS ANALYSIS

13.1.1 Methodologies Applied to the Radioisotope Production Facility Integrated Safety Analysis Process

This section describes methodologies applied to the RPF ISA. The ISA process comprises the PHA and the follow-on development and completion of QRAs to address events and hazards identified in the PHA as requiring further evaluation.

The ISA process flow diagram is provided Figure 13-1. The ISA process (being adapted for this application) consists of conducting a PHA of a system using a combination of written process descriptions, process flow diagrams (PFD), process and instrument drawings (P&ID), and supporting calculations to identify events that could lead to adverse consequences. Those adverse consequences are evaluated qualitatively by the ISA team members to identify the likelihood and severity of consequences using guidance on event frequencies and consequence categories consistent with the regulatory guidelines.

Each event with an adverse consequence that involves licensed material or its byproducts is evaluated for risk using a risk matrix that enables the user to identify unacceptable intermediate- and high-consequence risks. For the unacceptable intermediate- and high-consequence risks events, the IROFS developed to prevent or mitigate the consequences of the events and an event tree analysis are used to demonstrate that the risk can be reduced to acceptable frequencies through preventative or mitigative IROFS.

Fault trees and failure mode and effects analysis can be used to (1) provide quantitative failure analysis data (failure frequencies) for use in the event tree analysis of the IROFS, as necessary, or (2) quantitatively analyze an event from its basic initiators to demonstrate that the quantitative failure frequency is already highly unlikely under normal standard industrial conditions, thus not needing the application of IROFS. Once the IROFS are developed, management measures are identified to ensure that the IROFS failure frequency used in the analysis is preserved and the IROFS are able to perform their intended function when needed.

The following subsections summarize the RPF ISA methodologies.

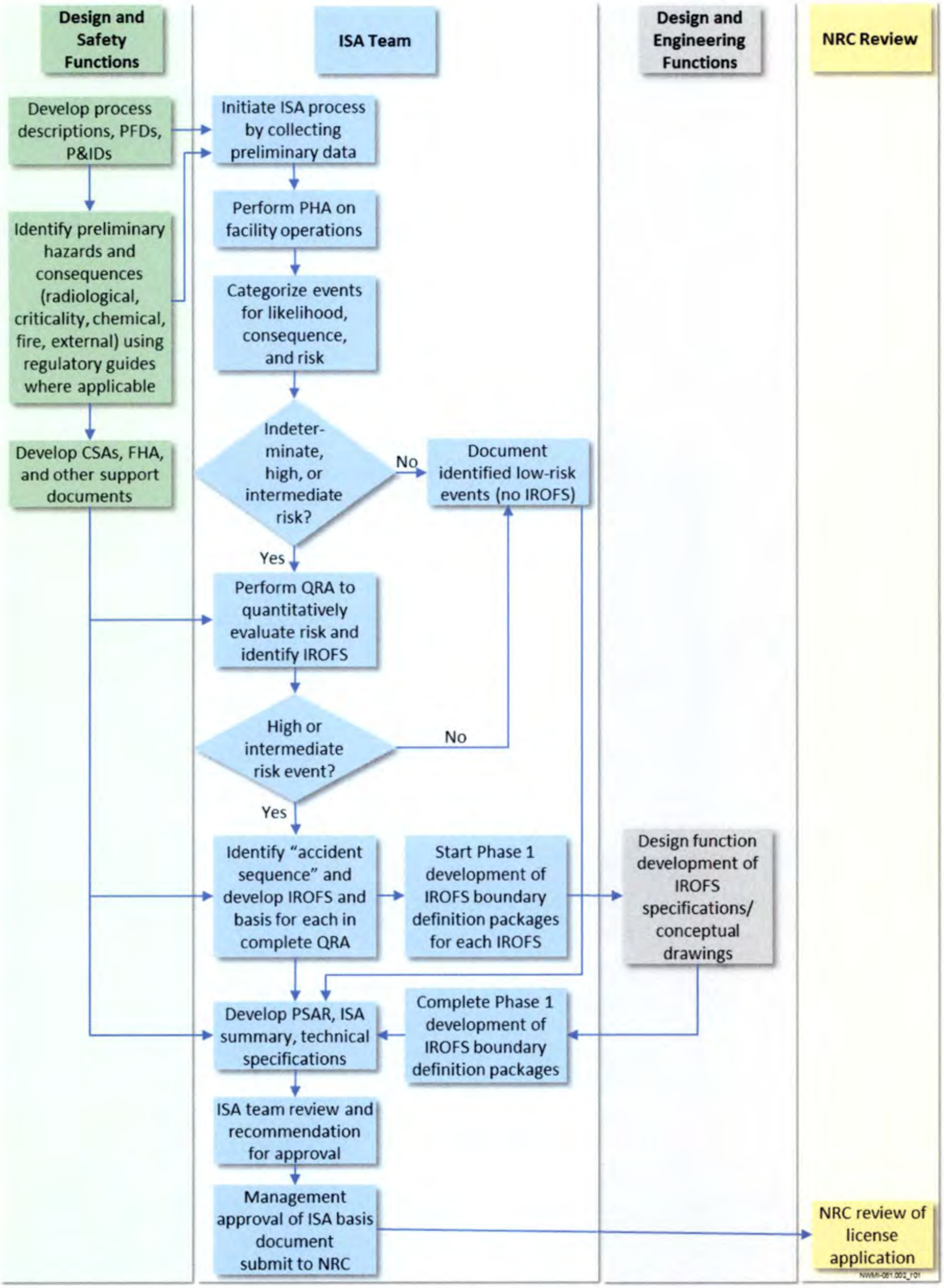

Figure 13-1. Integrated Safety Analysis Process Flow Diagram

13.1.1.1 Accident Likelihood Categories, Consequence Severity Categories, and Risk Matrix

Table 13-1 shows the accident likelihood categories applied to the RPF ISA process. Table 13-2 shows qualitative guidelines for applying the likelihood categories from Table 13-1. Table 13-3 shows accident consequence severity categories from 10 CFR 70.61, "Performance Requirements." Table 13-4 shows the RPF risk matrix, which is a product of the likelihood and consequence severity categories from Table 13-1 and Table 13-3, respectively.

Table 13-1. Likelihood Categories

	Likelihood category	Event frequency limit
Not unlikely	3	More than 10^{-3} events per year
Unlikely	2	Between 10^{-3} and 10^{-5} events per year
Highly unlikely	1	Less than 10^{-5} per events per year

Table 13-2. Qualitative Likelihood Category Guidelines

Likelihood category	Initiator
3	An event initiated by a human error
3	An event initiated by failure of a process system processing corrosive materials
3	An event initiated by a fire or explosion in areas where combustibles or flammable materials are present
3	An event initiated by failure of an active control system
3	A damaging seismic event
3	A damaging high wind event
3	A spill of material
3	A failure of a process variable monitored or unmonitored by a control system
3	A valve out of position or a valve that fails to seat and isolate
3	Most standard industrial component failures (valves, sensors, safety devices, gauges, etc.)
3	An adverse chemical reaction caused by improper quantities of reactants, out-of-date reactants, out-of-specification reaction environment, or the wrong reactants are used
3	Most external man-made events (until confirmed using an approved method)
2	An event initiated by the failure of a robust passive design feature with no significant internal or external challenges applied (e.g., spontaneous rupture of an all-welded dry nitrogen system pipe operating at or below design pressure in a clean, vibration-free environment)
1-2	An adverse chemical reaction when proper quantities of in-date chemicals are reacted in the proper environment
1	Natural phenomenon such as tsunami, volcanos, and asteroids for the Missouri facility site

Table 13-3. Radioisotope Production Facility Consequence Severity Categories Derived from 10 CFR 70.61

Category description	Consequence category	Workers	Off-site public	Environment
High consequence	3	• Radiological dose[a] > 1 Sv (100 rem) • Airborne, radiologically contaminated nitric acid >170 ppm nitric acid (AEGL-3, 10-min exposure limit) • Unshielded[b] nuclear criticality	• Radiological dose[a] > 0.25 Sv (25 rem) • Toxic intake > 30 mg soluble U • Airborne, contaminated nitric acid > 24 ppm nitric acid (AEGL-2, 60-min exposure limit)	
Intermediate consequence	2	• Radiological dose[a] between 0.25 Sv (25 rem) and 1 Sv (100 rem) • Airborne, radiologically contaminated nitric acid > 43 ppm nitric acid (AEGL-2, 10-min exposure limit)	• Radiological dose[a] between 0.05 Sv (5 rem) and 0.25 Sv (25 rem) • Airborne, contaminated nitric acid > 0.16 ppm nitric acid (AEGL-1, 60-min exposure limit)	24-hr radioactive release > 5,000 × Table 2 of 10 CFR 20,[c] Appendix B
Low consequence	1	Accidents with lower radiological, chemical, and/or toxicological exposures than those above from licensed material and byproducts of licensed material	Accidents with lower radiological, chemical, and/or toxicological exposures than those above from licensed material and byproducts of licensed material	Radiological releases producing lower effects than those listed above from licensed material

Source: 10 CFR 70.61, "Performance Requirements," *Code of Federal Regulations*, Office of the Federal Register, as amended.

[a] As total effective dose equivalent.

[b] A shielded criticality accident is also considered a high-consequence event.

[c] 10 CFR 20, "Standards for Protection Against Radiation," *Code of Federal Regulations*, Office of the Federal Register, as amended.

AEGL = Acute Exposure Guideline Level. U = uranium.

Table 13-4. Radioisotope Production Facility Risk Matrix

Severity of consequences	Likelihood of occurrence: Highly unlikely (Likelihood category 1)	Likelihood of occurrence: Unlikely (Likelihood category 2)	Likelihood of occurrence: Not unlikely (Likelihood Category 3)
High consequence (Consequence category 3)	Risk index = 3 Acceptable risk	Risk index = 6 Unacceptable risk	Risk index = 9 Unacceptable risk
Intermediate consequence (Consequence category 2)	Risk index = 2 Acceptable risk	Risk index = 4 Acceptable risk	Risk index = 6 Unacceptable risk
Low consequence (Consequence category 1)	Risk index = 1 Acceptable risk	Risk index = 2 Acceptable risk	Risk index = 3 Acceptable risk

13.1.1.2 Accident Consequence Analysis

The ISA process requires an understanding of the source terms and consequences of an adverse event to determine if the event is low, intermediate, or high consequence, as compared with the hazard criteria identified in Table 13-4. NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, offers methodologies to calculate the quantitative consequences of events. For simplicity and prudent expenditure of resources, the RPF ISA assumes a worst-case approach using a few bounding evaluations of events that are identified through either:

- Calculations (e.g., the source term and radiation doses caused by contained material in the system)
- Studies of representative accidents (e.g., comparison of accidental criticalities in industry with processes similar to those at the RPF)
- Bounding release calculations using approved methods (e.g., using RASCAL [Radiological Assessment System for Consequence Analysis] to model bounding facility releases that affect the public)
- Reference to nationally recognized safety organizations (e.g., use of Acute Exposure Guideline Levels [AEGL] from the U.S. Environmental Protection Agency to identify chemical exposure limits for each consequence category)
- Approved methods for evaluation of natural and man-made phenomenon and comparison to the design basis (e.g., calculation of explosive damage potential from the nearest railroad line on the facility)

Accident consequence analysis results are identified before or during the ISA process following preliminary reviews of the processes, and as the process hazard identification phase identifies new potential hazards.

Initial hazards identified by the preliminary reviews include:

- High radiation dose to workers and the public from irradiated target material during processing
- High radiation dose due to accidental nuclear criticality
- Toxic uptake of licensed material by workers or the public during processing or accidents
- Fires and explosions associated with chemical reactions and use of combustible materials and flammable gases
- Chemical exposures associated with chemicals used in processing the irradiated target material
- External events (both natural and man-made) that impact the facility operations

13.1.1.3 What-If and Structured What-If

RPF activities that will be mainly conducted by personnel using a sequence of actions to affect a process were evaluated using what-if or structured-what-if techniques to identify process hazards that can lead to unacceptable risk. These methods allow free-form evaluation of the activity by ISA team members, which can be enhanced by using a list of key guidewords addressing the specific hazards identified in the facility (e.g., the deviations to normal condition criticality safety controls like spacing, mass, moderation; material spills; wrong materials, place, or time for activities; etc.). The key words for each structured what-if evaluation are documented in the PHA.

13.1.1.4 Hazards and Operability Study Method

For processes that are part of a processing system and have well-defined PFDs and/or P&IDs, the more structured hazards and operability (HAZOP) approach was used. This method systematically evaluates each node of a process using a set of key words that enables the team to systematically identify adverse changes in the process and evaluate those changes for adverse consequences. The key words for each evaluation are documented in the PHA.

13.1.1.5 Event Tree Analysis

An event tree analysis (ETA) is a bottoms-up, logical modeling technique for both success and failure that explores responses through a single initiating event and lays a path for assessing probabilities of the outcomes and overall system analysis. ETA uses a modeling technique referred to as an event tree, which branches events from one single event using Boolean logic.

The ISA uses ETA in two primary ways. For those initiating events where the ISA team is uncertain of the likelihood of reaching the adverse consequence, the method can be used during the QRA to follow the sequence of events leading to an adverse consequence and thus quantify the adverse event's frequency given the initiator. ETA is also used in the QRA process to demonstrate that the IROFS, selected to prevent an adverse event, reduce the failure frequency to a level that satisfies the performance requirements (e.g., the frequency of a high-consequence event is reduced to highly unlikely).

13.1.1.6 Fault Tree Analysis

Fault tree analysis (FTA) is a top-down, deductive failure analysis in which an undesirable system state is analyzed with Boolean logic to combine a series of lower-level initiating events. The process enables the user to understand how systems can fail, identify the best ways to reduce risk, and/or determine event rates of an accident or a particular system-level functional failure. This analysis method is mainly used in QRAs when a failure frequency or probability is needed for a specific component, an IROFS, or some other complex process.

13.1.1.7 Failure Modes and Effects Analysis

Failure modes and effects analysis (FMEA) is an inductive reasoning (forward logic) single point of failure analysis that is also quantitative in nature. FMEA involves reviewing as many components, assemblies, and subsystems as possible to identify failure modes, along with associated causes and effects. For each component, the failure modes and associated effects on the rest of the system are recorded in a FMEA worksheet. This is an exhaustive analysis technique that can be used to evaluate the reliability of a complex, active engineered control (AEC) type of IROFS.

13.1.2 Accident-Initiating Events

Each of the following accident initiating events was included in the PHA. Loss of power as an accident event is discussed further in Section 13.2.5.

- Criticality accident
- Loss of electrical power
- External events (meteorological, seismic, fire, flood)
- Critical equipment malfunction
- Operator error
- Facility fire (explosion is included in this category)
- Any other event potentially related to unique facility operations

The PHA (NWMI-2015-SAFETY-001) identifies and categorizes accident sequences that require further evaluation. Table 13-5 defines the top-level accident sequence notation used in the RPF PHA.

Table 13-5. Radioisotope Production Facility Preliminary Hazard Analysis Accident Sequence Category Designator Definitions

PHA top-level accident sequence category[a]	Definition
S.C.	Criticality
S.F.	Fire or explosion
S.R.	Radiological
S.M.	Man-made
S.N.	Natural phenomena
S.CS.	Chemical safety

[a] The alpha category designator is followed in the PHA by a two-digit number "XX" that refers to the specific accident sequence (e.g., S.C.01, S.F.07). Specific accident sequences are discussed in Sections 13.1.3 and 13.3.

PHA = preliminary hazard analysis.

Table 13-6 provides a crosswalk between the PHA top-level accident sequence categories and the NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content*, Part 1 Interim Staff Guidance (ISG) accident initiating events listed above. As noted at the bottom of Table 13-6, PHA accident sequences involve one or more of the NUREG-1537 Part 1 ISG accident initiating event categories, as noted by ✓ in the corresponding table cell, but the PHA accident sequences themselves are not necessarily initiated by the ISG accident initiating event. Table 13-6 shows how PHA accident sequences correspond with ISG accident initiating events, and demonstrates that the PHA considers the full range of accident events identified in the ISG.

Table 13-6. Crosswalk of NUREG-1537 Part 1 Interim Staff Guidance Accident Initiating Events versus Radioisotope Production Facility Preliminary Hazards Analysis Top-Level Accident Sequence Categories

NUREG-1537[a] Part 1 ISG accident initiating event category	PHA Top-Level Accident Sequence Category[b]					
	S.C.	S.F.	S.R.	S.M.	S.N.	S.CS.
Criticality accident	✓	✓			✓	
Loss of electrical power			✓		✓	
External events (meteorological, seismic, fire, flood)	✓	✓		✓	✓	✓
Critical equipment malfunction	✓	✓	✓	✓		✓
Operator error	✓		✓	✓		✓
Facility fire (explosion is included in this category)		✓	✓			
Any other event potentially related to unique facility operations	✓		✓	✓		

[a] NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content*, Part 1, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C., February 1996.

[b] PHA accident sequences involve one or more of the NUREG-1537 Part 1 ISG accident initiating event categories, as noted by an ✓ in the corresponding table cell, but the PHA sequences themselves are not necessarily initiated by the ISG accident initiating event.

ISG = Interim Staff Guidance. PHA = preliminary hazard analysis.

The RPF PHA subdivides the RPF process into eight primary nodes based on facility design documentation. Table 13-7 lists the RPF primary nodes and corresponding subprocesses, as identified in the PHA.

Table 13-7. Radioisotope Production Facility Preliminary Hazards Analysis Primary Process Nodes and Subprocesses (2 pages)

Node no.	Node name	Subprocesses encompassed in node
1.0.0	Target fabrication process	• Fresh uranium receipt and storage • Fresh uranium dissolution • Uranyl nitrate blending and feed preparation • Nitrate extraction • Recycled uranyl nitrate concentration • [Proprietary Information] • [Proprietary Information] • [Proprietary Information] • [Proprietary Information] • [Proprietary Information] • [Proprietary Information] • Uranium scrap recovery • Target assembly, loading, inspection, quality checking, verification, packaging and storage
2.0.0	Target dissolution process	• [Proprietary Information] • [Proprietary Information] • Primary process offgas treatment • Fission gas retention
3.0.0	Molybdenum recovery and purification process	• Feed preparation • First stage recovery • First stage purification preparation • First stage purification • Second stage purification preparation • Second stage purification • Final purification adjustment • ^{99}Mo preparation for shipping
4.0.0	Uranium recovery and recycle process	• Impure uranium lag storage • First-cycle uranium recovery • Second-cycle uranium purification • Product uranium lag storage • Other support (storage vessels, transfer lines, solid waste handling for resin bed replacement)

Table 13-7. Radioisotope Production Facility Preliminary Hazards Analysis Primary Process Nodes and Subprocesses (2 pages)

Node no.	Node name	Subprocesses encompassed in node
5.0.0	Waste handling system process	• Liquid waste storage • High dose liquid waste volume reduction • Condensate storage and recycling • Concentrated high dose liquid waste storage/preparation • Low dose liquid waste volume reduction and storage • Liquid waste solidification • Solid waste handling • Waste encapsulation • TCE solvent reclamation • Mixed waste accumulation
6.0.0	Target receipt and disassembly process	• Cask receipt and target unloading • Target Inspection • Target disassembly • [Proprietary Information] • Target disassembly stations • Gaseous fission product control • [Proprietary Information] • Empty target hardware handling
7.0.0	Ventilation system	• (No subprocesses identified in PHA. Ventilation system provides cascading pressure zones, a common air supply system with makeup air as necessary, heat recovery for preconditioning incoming air, and HEPA filtration.)
8.0.0	Natural phenomena, man-made external events, and other facility operations	• Natural phenomena • Man-made external events • Chemical storage and preparation areas • On-site vehicle operation • General storage, utilities, and maintenance activities • Laboratory operations • Hot cell support activities • Waste storage operations including packaging and shipment

^{99}Mo = molybdenum-99
HEPA = high-efficiency particulate air.
PHA = preliminary hazards analysis.
TCE = trichloroethylene.

Table 13-8 shows a crosswalk that identifies the applicability of RPF PHA top-level accident sequence categories to the primary process nodes. The information in this table is referenceable to Table 13-6 and ultimately shows the relationship between the PHA process nodes and the NUREG-1537 Part 1 ISG accident initiating event categories via the PHA top-level accident scenario categories.

Table 13-8. Crosswalk of Radioisotope Production Facility Preliminary Hazards Analysis Process Nodes and Top-Level Accident Sequence Categories

Primary process node	PHA Top-Level Accident Sequence Category					
	S.C. (criticality)	S.F. (fire)	S.R. (radiological)	S.M. (man-made)	S.N. (natural phenomena)	S.CS. (chemical safety)
Target fabrication (Node 1.0.0)	✓	✓	✓			
Target dissolution (Node 2.0.0)	✓	✓	✓			
Molybdenum recovery and purification (Node 3.0.0)	✓	✓	✓			
Uranium recovery and recycle (Node 4.0.0)	✓	✓	✓			
Waste handling system (Node 5.0.0)	✓	✓	✓			
Target receipt and disassembly (Node 6.0.0)	✓		✓			
Ventilation system (Node 7.0.0)	✓	✓	✓			
Natural phenomena, man-made external events, and other facility operations (Node 8.0.0)	✓	✓	✓	✓	✓	✓

Note: The ✓ in a table cell indicates that the accident sequence category applies to the process node. If it does not, the cell is blank.

PHA = preliminary hazards analysis.

13.1.3 Preliminary Hazards Analysis Results

This section presents the radiological, criticality, and chemical hazards that could result in high or intermediate consequences.

13.1.3.1 Hazard Criteria

Methodologies and hazard criteria are identified in Section 13.1.1. Numerous hazards are present during the handling and processing the materials in the RPF. The target material is fissile LEU consisting of uranium enriched up to 19.95 weight percent (wt%) uranium-235 (^{235}U). This material presents a criticality accident hazard in the processes that involve high concentrations of uranium. Both 10 CFR 50 and 10 CFR 70 require that accidental nuclear criticalities be prevented using the double-contingency principle, as defined in ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors*. The RPF separates ^{99}Mo from among the fission products in the irradiated LEU target material. The fission products, including ^{99}Mo, present a high-dose hazard that must be properly contained and shielded to protect workers and the public. Radiation protection standards are given in 10 CFR 20, "Standards for Protection Against Radiation," and its appendices.

The RPF also uses high concentrations of acids, caustics, and oxidizers, both separate from and mixed with licensed material, that present chemical hazards to workers. The National Institute for Occupational Safety and Health (NIOSH) provides acute exposure guidelines (CDC, 2010) that evaluate chemical exposure hazards to workers and the public from chemicals and toxic licensed material.

The facility can also be impacted by various internal and external man-made and natural phenomena events that have the potential to damage structures, systems, and components (SSC) that control the licensed material, thereby leading to intermediate- and high-consequence events.

Known and credited safety features for normal operations include:

- The hot cell shielding boundary, credited for shielding workers and the public from direct exposure to radiation (an expected operational hazard)
- The hot cell confinement boundaries, credited with confining fissile and high-dose solids, liquids, and gases, and controlling gaseous releases to the environment

Administrative and passive engineered design features that control uranium batch size, volume, geometry and interaction are credited for maintaining critically safe (i.e., subcritical) configurations during normal operations with fissile material. The RPF PHA identifies abnormal operation event initiators that require further evaluation for IROFS to ensure that the double-contingency principle is satisfied.

13.1.3.2 Radioisotope Production Facility Accident Sequence Evaluation

A structured what-if analysis was used to evaluate RPF system nodes where operators are primarily involved with licensed material manipulations. All process system nodes were analyzed using a HAZOP approach with special emphasis on criticality, radiological, and chemical safety hazards. Fire safety issues are addressed in every node and addressed generally in Node 8.0.0. Fire safety issues include the explosive hazard associated with hydrogen gas generation via radiolytic decomposition of water in process solutions and due to certain chemical reactions encountered during dissolution processes. Most hot cell processing areas contain very few combustible materials, either transient or fixed.

The RPF PHA has identified adverse events listed in Table 13-9 through Table 13-16. Adverse events are identified as:

- Standard industrial events that do not involve licensed material
- Acceptable accident sequences that satisfy performance criteria by being low consequence and/or low frequency
- Unacceptable accident sequences that require further evaluation via the QRA process

An accident sequence number was assigned to each accident initiator that results in the same, or similar, bounding accident sequence result and consequence. The same accident sequence designator can appear in multiple nodes. (Table 13-5 provides definitions of accident sequence category designators.)

Table 13-9. Adverse Event Summary for Target Fabrication and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
1.1.1.1, 1.1.1.2, 1.6.1.1, 1.8.1.1, 1.8.2.1, and 1.8.3.1	Operator double batches allotted amount of material (fresh U, scrap U, [Proprietary Information], target batch) into one location or container during handling	Accidental criticality issue – Too much fissile mass in one location may become critical	S.C.02, Failure of administrative control on mass (batch limit) during handling of fresh U, scrap U, [Proprietary Information], and targets
1.1.1.3	Supplier ships greater than 20 wt% ^{235}U to site	Accidental criticality issue – Too much ^{235}U put into a container or solution vessel, exceeding assumed amounts	S.C.01, Failure of site enrichment limit
1.1.1.6, 1.1.1.7, 1.6.1.2, 1.6.1.4, 1.8.1.2, 1.8.1.3, 1.8.1.6, 1.8.2.2, 1.8.2.3, 1.8.3.2, 1.8.3.3, 1.8.3.4, and 1.8.3.5	Operator handling various containers of uranium or batches of uranium components brings two containers or batches closer together than the approved interaction control distance	Accidental criticality issue – Too much uranium mass in one location	S.C.03, Failure of administrative control on interaction limit during handling of fresh U, scrap U, [Proprietary Information], and targets
1.2.1.1, 1.2.1.11, 1.2.1.14, 1.2.1.25, 1.3.1.1, 1.3.1.6, 1.3.1.11, 1.3.1.17, 1.4.1.19, 1.4.1.20, 1.4.1.21, 1.4.1.23, 1.4.2.6, 1.4.2.10, 1.4.2.15, 1.4.3.14, 1.4.3.26, 1.4.3.31, 1.4.4.1, 1.4.4.6, 1.4.4.10, 1.4.4.15, 1.5.1.21, 1.5.1.23, 1.5.1.26, 1.5.2.16, 1.7.1.1, 1.7.1.11, 1.7.1.14, 1.7.1.25, 1.9.1.1, 1.9.1.6, 1.9.1.10, and 1.9.1.15	Failure of safe geometry confinement	Accidental criticality from fissile solution not confined in safe geometry	S.C.04, Spill of fissile material from safe geometry system confinement
1.2.1.2 and 1.7.1.2	Uranium-containing solution leaks out of safe geometry confinement into the heating/cooling jacketed space	Accidental criticality from fissile solution not confined in safe geometry	S.C.05, Leak of fissile solution into heating/ cooling jacket on vessel
1.2.1.3, 1.4.3.33, 1.4.3.34, and 1.7.1.3	Uranium solution is transferred via a leak between the process system and the heater/cooling jackets or coils on a tank or in an exchanger	Accidental criticality from fissile solution not confined in safe geometry	S.C.07, Leak of fissile solution across auxiliary system boundary (chilled water or steam)

Table 13-9. Adverse Event Summary for Target Fabrication and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
1.2.1.8, 1.3.1.4, 1.4.1.15, 1.4.2.4, 1.4.3.18, 1.4.4.4, 1.5.1.20, 1.5.2.11, 1.7.1.8, and 1.9.1.4	Failure of safe geometry dimension caused by configuration management (installation, maintenance), internal or external event	Accidental criticality from fissile solution not confined in safe geometry	S.C.19, Failure of passive design feature – Component safe geometry dimension
1.2.1.12, 1.3.1.9, 1.4.2.8, 1.4.4.8, 1.4.5.4, 1.7.1.12, and 1.9.1.8	Tank overflow into process ventilation system	Accidental criticality issue – Fissile solution entering a system not necessarily designed for fissile solutions	S.C.06, Overfill of a tank or component causing fissile solution entering the process vessel ventilation system
1.3.1.2, 1.4.2.2, 1.4.4.2, and 1.9.1.20	Uranium precipitate or other high uranium solids accumulate in safe geometry vessel	Accidental criticality from fissile solution not confined to safe geometry and interaction controls within allowable concentrations	S.C.20, Failure of concentration limits – Precipitation of uranium in safe geometry tank
1.2.1.26, 1.3.1.7, 1.5.1.3, and 1.5.2.5	Uranium solution backflows into an auxiliary support system (water line, purge line, chemical addition line) due to various causes	Accidental criticality issue – Fissile solution entering a system not necessarily designed for fissile solutions	S.C.08, Fissile solution backflow into an auxiliary system at a fill point boundary
1.4.1.6, 1.4.1.12, and 1.4.1.16	Failure of safe geometry confinement due to inadvertent transfer to U-bearing solution across a boundary into non-favorable geometry	Accidental criticality from fissile solution not confined in safe geometry	S.C.11, Fissile material contamination of contactor regeneration aqueous waste stream - boundary to unsafe geometry system
1.4.3.1, 1.4.3.9, 1.4.3.19, 1.4.3.21, 1.4.5.9, and 1.4.5.11	Failure of safe geometry confinement due to inadvertent transfer to U-bearing solution across a boundary into non-favorable geometry	Accidental criticality from fissile solution not confined in safe geometry	S.C.09, Fissile material contamination of evaporator condensate - boundary to unsafe geometry system
1.6.1.3	Failure of safe geometry confinement due to inadvertent transfer to U-bearing solution across a boundary into non-favorable geometry	Accidental criticality from fissile solution not confined in safe geometry	S.C.12, Wash of [Proprietary Information] with wrong reagent contaminating wash solution with fissile U; boundary to unsafe geometry system
1.1.1.11	Dusty surface generated during shipping on uranium pieces spontaneously ignites due to pyrophoric nature of uranium	Potential exposure to workers due to airborne uranium generation	S.F.01, Pyrophoric fire in uranium metal

Table 13-9. Adverse Event Summary for Target Fabrication and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
1.2.1.6, 1.2.1.11, 1.7.1.6, and 1.7.1.11	Hydrogen buildup in tanks or system, leading to explosive concentrations	Explosion leading to radiological and criticality concerns	S.F.02, Accumulation of flammable gas in tanks or systems
1.4.1.17, 1.4.1.21, and 1.4.1.23	Fire in process system containing high concentration uranium spreads the uranium	Radiological and criticality issue – Radiological airborne release of uranium and uncontrolled spread of uranium outside safe geometry confinement	S.F.07, Fire in nitrate extraction system - flammable solvent with uranium
1.6.1.6, 1.6.1.9, and 1.6.1.12	Air inleakage into the reduction furnace during H_2 purge cycle or H_2 inleakage into reduction furnace before inerting with nitrogen can lead to an explosive mixture in the presence of an ignition source	Accidental criticality issue – Uncontrolled spread of uranium outside safe geometry confinement	S.F.03, Hydrogen detonation in reduction furnace
1.6.1.8	Loss of cooling of exhaust or fire in the reduction furnace leads to high temperatures in downstream ventilation component and accelerated release of adsorb radionuclides	Radiological issue – Potential accelerated release of high-dose radionuclides to the stack (worker and public exposure)	S.F.04, High temperature damage to process ventilation system due to loss of cooling in reduction furnace exhaust or fire in reduction furnace
1.2.1.11, 1.2.1.14, 1.4.1.17, 1.4.1.19, 1.4.1.20, 1.4.1.21, 1.4.1.23, 1.4.2.6, 1.4.3.14, 1.4.3.26, 1.4.3.31, 1.4.3.32, 1.7.1.11, 1.7.1.14, and 1.9.1.6	High concentration uranium solution is sprayed from the system, causing high airborne radioactivity	Radiological release of uranium solution spray that remains suspended in the air, exposing workers or the public	S.R.03, Solution spray release potentially creating airborne uranium above DAC limits
1.2.1.11, 1.2.1.12, 1.2.1.14, 1.2.1.25, 1.3.1.1, 1.3.1.6, 1.3.1.11, 1.3.1.17, 1.4.1.17, 1.4.1.18, 1.4.1.19, 1.4.1.21, 1.4.2.1, 1.4.2.6, 1.4.2.8, 1.4.2.10, 1.4.2.15, 1.4.3.14, 1.4.3.26, 1.4.3.31, 1.4.4.6, 1.4.4.10, 1.4.4.15, 1.5.1.21, 1.7.1.11, 1.7.1.14, 1.7.1.25, 1.9.1.1, 1.9.1.6, 1.9.1.8, 1.9.1.10, and 1.9.1.15	High concentration uranium solution is spilled from the system	Potential radiological exposure to workers from uranium-contaminated solution	S.R.01, Uranium-contaminated solution spill

Table 13-9. Adverse Event Summary for Target Fabrication and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
1.2.1.21, 1.2.1.22, 1.4.5.13, 1.7.1.21, and 1.7.1.22	Boiling or carryover of steam or high concentration water vapor into the primary ventilation system, affecting retention beds from partial or complete loss of cooling system capabilities	Radiological release from retention beds	S.R.04, Liquid enters process vessel ventilation system damaging IRU or retention beds releasing retained radionuclides
1.3.1.16 and 1.4.1.24	High-dose solution (failure of the uranium recovery process) results in high-dose radionuclides entering the first stage of processing uranium [Proprietary Information] (eventually handled by the worker)	Potentially high radiological exposure to workers	S.R.05, High-dose solution enters the UN blending and storage tank
1.8.3.7	Loading limits are not adhered to by the operators or the closure requirements are not satisfied, and the cask does not provide the containment or shielding function that it is designed to perform	High-dose to workers or the public from improperly shielded cask	S.R.28, Target or waste shipping cask not loaded or secured according to procedure, leading to personnel exposure

^{235}U = uranium-235.
DAC = derived air concentration.
H_2 = hydrogen gas.
IRU = iodine removal unit.
PHA = process hazards analysis.
U = uranium.
UN = uranyl nitrate.

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
2.1.1.1, 2.1.1.11, 2.1.1.13, 2.1.1.17, 2.2.1.5, 2.2.1.12, 2.2.1.15, 2.3.6.5, 2.3.6.12, and 2.3.6.13	Failure of safe geometry confinement	Accidental criticality from fissile solution not confined in safe geometry	S.C.04, Failure of confinement in safe geometry; spill of fissile material solution
2.1.1.2	Uranium-containing solution leaks out of safe geometry confinement into the heating/cooling jacketed space	Accidental criticality from fissile solution not confined in safe geometry	S.C.05, Leak of fissile solution in to heating/cooling jacket on vessel
2.1.1.3	Uranium solution is transferred via a leak between the process system and the heater/cooling jackets or coils on a tank or in an exchanger	Accidental criticality from fissile solution not confined in safe geometry	S.C.07, Leak of fissile solution across auxiliary system boundary (chilled water or steam)
2.1.1.8, 2.2.1.11, and 2.3.6.11	Failure of safe geometry dimension	Accidental criticality from fissile solution not confined in safe geometry	S.C.19, Failure of passive design feature; component safe-geometry dimension
2.1.1.12, 2.1.1.15, and 2.3.1.4	Failure of safe-geometry confinement	Accidental criticality from fissile solution not confined in safe geometry	S.C.13, Fissile solution enters the NO_x scrubber where high uranium solution is not intended
2.1.1.14 and 2.3.4.14	Tank overflow into process ventilation system	Accidental criticality issue – Fissile solution entering a system not necessarily designed for fissile solutions	S.C.06, System overflow to process ventilation involving fissile material
2.3.4.11	Uranium enters carbon retention bed dryer where it can mix with condensate to form a fissile solution	Accidental criticality from fissile material or solution not confined in safe geometry	S.C.24, Build-up of high uranium particulate in the carbon retention bed dryer system
2.1.1.33 and 2.1.1.34	Uranium solution backflows into an auxiliary support system (water line, purge line, chemical addition line) due to various causes	Accidental criticality and high radiological dose – High-dose and fissile solution entering a system not necessarily designed for fissile solutions that exist outside of hot cell walls	S.C.08, System backflow into auxiliary support system

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
2.1.1.18, 2.3.1.21, 2.3.2.21, 2.3.3.24, 2.3.4.3, and 2.3.5.5	Hydrogen build-up in tanks or system leading to explosive concentrations	Explosion leading to radiological and criticality concerns	S.F.02, Accumulation of flammable gas in tanks or systems
2.3.4.20, 2.3.5.2, 2.3.5.6, 2.3.5.10, and 2.3.5.13	A fire develops through exothermic reaction to contaminants in the carbon retention bed and rapidly releases accumulated gaseous high-dose radionuclides	Radiological issue – Potential accelerated release of high-dose radionuclides to the stack (worker and public exposure)	S.F.05, Fire in a carbon retention bed
2.1.1.1, 2.1.1.2, 2.1.1.11, 2.1.1.13, 2.1.1.17, 2.2.1.5, 2.2.1.12, 2.2.1.15, 2.3.6.5, 2.3.6.12, and 2.3.6.13	High-dose and/or high-concentration uranium solution is spilled from the system	Potential radiological exposure to workers from high-dose and/or high uranium-contaminated solution	S.R.01, Radiological release in the form of a liquid spill of high-dose and/or high uranium concentration solution
2.1.1.3	High-dose solution is transferred via a leak between the process system and the heater/cooling jackets or coils on a tank or in an exchanger	Radiological exposure to workers and the public from high-radiological dose not contained in the hot cell containment or confinement boundary	S.R.13, High-dose solution leaks to chilled water or steam condensate system
2.1.1.11, 2.1.1.17, 2.2.1.15, and 2.3.6.13	Spill leading to spray-type release, causing airborne radioactivity above DAC limits for exposure	Radiological dose from airborne spray of product solution from systems	S.R.03, Spray of product solution in hot cell area
2.1.1.23, 2.1.1.26, 2.1.1.27, 2.3.4.1, 2.3.4.12, and 2.3.4.17	Carryover of high vapor content gases or entrance of solutions into the process ventilation header can cause poor performance of the retention bed materials and release radionuclides	High airborne radionuclide release, affecting workers and the public	S.R.04, Carryover of heavy vapor or solution into the process ventilation header causes downstream failure of retention bed, releasing radionuclides
2.3.1.17, 2.3.1.22, 2.3.1.24, 2.3.2.17, 2.3.2.22, 2.3.2.24, 2.3.3.8, 2.3.3.20, 2.3.3.27, 2.3.4.3, 2.3.4.5, 2.3.4.6, and 2.3.4.8	A spill of low-dose condensate occurs for a variety of reasons from the confinement tanks or vessels	Potential radiological dose to workers and the public from spilled liquid	S.R.02, Spill of low-dose condensate
2.3.3.1, 2.3.3.2, 2.3.3.3, 2.3.3.6, 2.3.3.12, 2.3.3.13, 2.3.3.16, 2.3.3.17, 2.3.3.23, 2.3.4.13, 2,3.5.1, 2.3.5.6, 2.3.5.8, and 2.3.5.10	High flows through the IRU increases the release of the retained iodine and increases the high-dose concentration of this gas in the stack	Potential radiological dose to workers and the public from iodine above regulatory limits	S.R.06, High flow through IRU causes premature release of high-dose iodine gas

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
2.3.3.15 and 2.3.5.8	Low temperatures in the IRU inlet gas stream drives release of iodine from the unit	Potential radiological dose to workers and the public from iodine above regulatory limits	S.R.07, Loss of temperature control on the IRU leads to premature release of high-dose iodine
2.3.3.22 and 2.3.5.8	Liquid and water vapor in the IRU inlet gas stream drives release of iodine from the unit	Potential radiological dose to workers and the public from iodine above regulatory limits	S.R.04, Liquid/high vapor in the IRU leads to premature release of high-dose iodine
2.3.4.4, 2.3.4.5, and 2.3.4.6	Loss of vacuum pumps in the dissolver offgas treatment system leads to pressure buildup inside the process and potential release of radionuclides from the system upstream	Potential radiological dose to workers and the public from spilled liquid	S.R.08, Loss of vacuum pumps
2.3.4.11	Uncontrolled loss of media and contact with a liquid with potential for premature release of the adsorbed iodine	Potential radiological dose to workers and the public from iodine above regulatory limits	S.R.09, Loss of IRU media to downstream dryer
2.3.3.28, 2.3.4.19, 2.3.5.9, 2.3.4.15, and 2.3.5.11	Using the wrong retention media (IRU or carbon beds) or using saturated media with potential for ineffective adsorption of high-dose gaseous radionuclides	Potential radiological dose to workers and the public from radionuclides above regulatory limits	S.R.10, Wrong retention media added to bed or saturated retention media
2.3.4.16, 2.3.5.5, and 2.3.5.12	An event causes damage to the structure holding the retention media, and retention media is released to an uncontrolled environment	Potential radiological dose to workers and the public from radionuclides above regulatory limits	S.R.09, Breach of an IRU or retention bed resulting in release of the media
2.1.1.33 and 2.1.1.34	High-dose process solution backflows into an auxiliary support system (water line, purge line, chemical addition line) due to various causes	High radiological dose – High dose process solution enters a system that exits outside of the hot cell walls	S.R.11, System backflow of high-dose solution into an auxiliary support system and outside the hot cell boundary

DAC = derived air concentration.
IRU = iodine removal unit.
NO_x = nitrogen oxide.
PHA = process hazards analysis.

Table 13-11. Adverse Event Summary for Molybdenum Recovery and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
3.3.1.24	Higher radiation dose due to hold-up accumulation or transient batch differences	Higher localized dose in hot cell boundary (unoccupied by workers)	N/A
3.2.3.7, 3.2.4.7, 3.4.3.7, 3.4.4.7, 3.6.3.7, and 3.6.4.7	Chemical spills of nonradiologically contaminated bulk chemicals	Standard industrial accident – Chemical exposure (not involving licensed material) to workers	N/A
3.7.4.5 and 3.7.4.6	Dropped cask or cask component during loading or handling	Standard industrial accident – Worker injury	N/A
3.7.4.2, 3.7.5.2, and 3.7.5.3	Mo product is exposed with no shielding as the result of an accident, shipment mishap, or shipment mishandling after leaving the site	Potential dose to the public and/or environment due to release or mishandling of Mo product during transit	N/A – Addressed by DOT packaging and transportation regulations (10 CFR 71[a])
3.1.1.9, 3.1.1.14, 3.1.1.23, 3.1.2.4, 3.1.2.7, 3.1.2.13, 3.1.2.16, 3.1.2.17, 3.2.1.6, 3.2.1.10, 3.2.1.20, 3.2.1.22, 3.2.1.23, 3.2.2.9, 3.2.2.13, 3.2.3.6, 3.2.3.8, 3.2.5.9, 3.2.5.14, 3.2.5.23, 3.8.1.9, 3.8.1.13, and 3.8.1.22	Failure of safe-geometry confinement	Accidental criticality from fissile solution not confined in safe geometry	S.C.04, Failure of confinement in safe geometry; spill of fissile material solution
3.1.1.4, 3.1.1.16, 3.2.5.4, 3.2.5.16, and 3.8.1.4	Tank overflow into process ventilation system	Accidental criticality issue – Fissile solution entering a system not necessarily designed for fissile solutions	S.C.06, System overflow to process ventilation involving fissile material
3.1.1.23, 3.2.1.23, 3.2.5.23, and 3.8.1.22	Uranium solution is transferred via a leak between the process system and the heater/cooling jackets or coils on a tank or in an exchanger	Accidental criticality from fissile solution not confined in safe geometry	S.C.07, Leak of fissile solution across auxiliary system boundary (chilled water or steam)
3.2.1.4, 3.2.1.5, 3.2.2.3, 3.2.2.4, 3.2.2.5, 3.2.3.6, and 3.2.4.6	Fissile product solution transferred to a system not designed for safe-geometry confinement	Criticality safety issue – Fissile solution directed to a system not intended for fissile solution	S.C.10, Inadvertent transfer of solution to a system not designed for fissile solutions

Table 13-11. Adverse Event Summary for Molybdenum Recovery and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
3.1.1.13, 3.1.2.9, 3.2.1.15, 3.2.5.13, and 3.8.1.12	Failure of safe-geometry dimension	Accidental criticality from fissile solution not confined in safe geometry	S.C.19, Failure of passive design feature; component safe-geometry dimension
3.1.1.25, 3.2.5.25, 3.3.1.25, 3.5.1.25, and 3.8.1.24	Hydrogen buildup in tanks or system, leading to explosive concentrations	Explosion leading to radiological and criticality concerns	S.F.02, Accumulation of flammable gas in tanks or systems
3.7.1.1, 3.7.1.2, 3.7.2.1, 3.7.3.1, 3.7.3.2, and 3.7.4.1	Operator spills Mo product solution during remote handling operations	Radiological spill of high-dose Mo solution	S.R.01, Radiological spill of Mo product during remote handling
3.1.1.9, 3.1.1.14, 3.1.1.23, 3.1.2.7, 3.1.2.13, 3.1.2.16, 3.1.2.17, 3.2.1.6, 3.2.1.20, 3.2.1.22, 3.2.1.23, 3.2.2.7, 3.2.2.9, 3.2.2.13, 3.2.3.6, 3.2.3.8, 3.2.3.10, 3.2.4.10, 3.2.5.9, 3.2.5.14, 3.2.5.23, 3.3.1.9, 3.3.1.14, 3.3.1.18, 3.3.1.22, 3.3.1.23, 3.3.2.4, 3.3.2.7, 3.3.2.13, 3.3.2.16, 3.3.2.17, 3.4.1.5, 3.4.1.9, 3.4.1.19, 3.4.1.21, 3.4.1.22, 3.4.2.6, 3.4.2.7, 3.4.2.12, 3.4.3.6, 3.4.3.8, 3.4.3.10, 3.4.3.14, 3.4.4.6, 3.4.4.10, 3.4.4.14, 3.5.1.9, 3.5.1.14, 3.5.1.16, 3.5.1.23, 3.5.2.4, 3.5.2.7, 3.5.2.13, 3.5.2.16, 3.5.2.17, 3.6.1.5, 3.6.1.6, 3.6.1.10, 3.6.1.20, 3.6.1.20, 3.6.1.23, 3.6.2.7, 3.6.2.9, 3.6.2.13, 3.6.3.8, 3.6.3.10, 3.6.3.14, 3.6.4.10, 3.6.4.14, 3.8.1.9, 3.8.1.13, and 3.8.1.22	Spill of product solution in the hot cell area	Radiological dose from spill of product solution from systems	S.R.01, Spill of product solution in hot cell area

Table 13-11. Adverse Event Summary for Molybdenum Recovery and Identification of Accident Sequences Needing Further Evaluation (3 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
3.1.1.9, 3.2.1.10, 3.2.1.22, 3.2.2.7, 3.2.2.9, 3.2.3.8, 3.2.3.10, 3.2.4.10, 3.2.5.9, 3.3.1.9, 3.3.1.18, 3.3.1.22, 3.3.2.7, 3.4.1.10, 3.4.1.22, 3.4.2.7, 3.4.3.8, 3.5.1.9, 3.5.1.23, 3.6.1.10, 3.6.2.7, 3.6.3.8, and 3.8.1.9	Spill leading to spray-type release, causing airborne radioactivity above DAC limits for exposure	Radiological dose from airborne spray of product solution from systems	S.R.03, Spray of product solution in hot cell area
3.1.1.7, 3.1.1.22, 3.2.5.7, 3.2.5.22, 3.3.1.4, 3.3.1.7, 3.3.1.16, 3.5.1.4, 3.5.1.7, 3.5.1.16, 3.5.1.22, 3.8.1.7, and 3.8.1.13	Boiling or carryover of steam or high-concentration water vapor into the primary process offgas ventilation system affecting retention beds with partial or complete loss of cooling system capabilities	Radiological release from retention beds	S.R.04, Loss of cooling, leading to liquid or steam carryover into the primary offgas treatment train
3.7.4.3	A Mo product cask is removed from the hot cell boundary with improper shield plug installation	Potential dose to workers, the public, and/or environment due to release or mishandling of Mo product during transit	S.R.12, Mo product is released during shipment
3.3.1.23, 3.3.2.16, 3.4.1.22, 3.5.1.23, and 3.6.1.23	High-dose radionuclide solution leaks through an interface between the process system and a heating/cooling jacket coil into a secondary system (e.g., chilled water or steam condensate) releasing radionuclides to workers, the public, and environment	High-dose radionuclide solution that leaks to the environment through another system to expose workers or the public	S.R.13, High dose radionuclide containing solution leaks to chilled water or steam condensate system

[a] 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, Office of the Federal Register, as amended.

DAC = derived air concentration.
DOT = U.S. Department of Transportation.
Mo = molybdenum.
N/A = not applicable.
PHA = process hazards analysis.

Table 13-12. Adverse Event Summary for Uranium Recovery and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
4.1.1.4, 4.1.1.18, 4.2.1.4, 4.2.1.6, 4.2.1.17, 4.2.1.18, 4.2.3.6, 4.2.8.4, 4.2.8.18, 4.2.10.4, 4.3.1.4, 4.3.1.6, 4.3.1.18, 4.3.1.19, 4.3.3.6, 4.3.8.4, 4.3.8.18, 4.3.10.4, 4.4.1.4, 4.4.1.17, 4.5.1.4, 4.5.1.17, 4.5.2.4, 4.5.2.17, 4.5.3.4, and 4.5.3.14	Tank overflow into process ventilation system	Accidental criticality issue – Fissile solution enters a system not necessarily designed for fissile solutions	S.C.06, System overflow to process ventilation involving fissile material
4.1.1.6, 4.2.1.7, 4.2.2.4, 4.2.3.4, 4.2.3.7, 4.2.3.8, 4.2.8.7, 4.3.1.7, 4.3.2.4, 4.3.3.4, 4.3.3.7, 4.3.3.8, 4.3.8.7, 4.4.1.6, 4.5.2.6, and 4.5.3.6	Uranium solution backflows into an auxiliary support system (water line, purge line, chemical addition line) due to various causes	Accidental criticality issue – Fissile solution enters a system not necessarily designed for fissile solutions	S.C.08, System backflow into auxiliary support system
4.1.1.14, 4.2.1.14, 4.2.3.16, 4.2.8.15, 4.3.1.15, 4.3.3.16, 4.3.8.15, 4.3.9.20, 4.4.1.14, 4.5.1.14, 4.5.2.14, and 4.5.3.11	Failure of safe geometry dimension caused by configuration management (installation, maintenance) or external event	Accidental criticality from fissile solution not confined in safe geometry	S.C.19, Failure of passive design feature; component safe-geometry dimension
4.1.1.8, 4.1.1.9, 4.1.1.12, 4.1.1.13, 4.1.1.16, 4.2.1.9, 4.2.1.13, 4.2.5.11, 4.2.8.10, 4.2.8.13, 4.2.8.14, 4.2.8.17, 4.2.9.18, 4.3.1.10, 4.3.1.11, 4.3.1.14, 4.3.1.17, 4.3.1.18, 4.3.5.11, 4.2.8.10, 4.3.8.13, 4.3.8.14, 4.3.8.17, 4.3.9.18, 4.4.1.8, 4.4.1.9, 4.4.1.12, 4.4.1.13, 4.4.1.16, 4.5.1.16, 4.5.2.8, 4.5.2.9, 4.5.2.12, 4.5.2.13, and 4.5.2.16	Uranium precipitate or other high uranium solids accumulate in safe-geometry vessel	Accidental criticality from fissile solution not confined to safe geometry and interaction controls within allowable concentrations	S.C.20, Failure of concentration limits
4.1.1.10, 4.1.1.15, 4.1.1.23, 4.2.1.11, 4.2.1.15, 4.2.1.24, 4.2.2.1, 4.2.3.11, 4.2.3.13, 4.2.3.18, 4.2.3.22, 4.2.3.23, 4.2.3.24, 4.2.4.10, 4.2.5.10, 4.2.7.8, 4.2.8.11, 4.2.8.16, 4.2.8.23, 4.2.9.16, 4.2.9.29, 4.2.9.34, 4.3.1.12, 4.3.1.16, 4.3.1.25, 4.3.2.1, 4.3.3.11, 4.3.3.13, 4.3.3.18, 4.3.3.22, 4.3.3.23, 4.3.3.24, 4.3.4.10, 4.3.5.10, 4.3.7.8, 4.3.8.11, 4.3.8.16, 4.3.8.23, 4.3.9.16, 4.3.9.28, 4.3.9.34, 4.4.1.10, 4.4.1.15, 4.4.1.23, 4.5.1.23, 4.5.2.10, 4.5.2.15, 4.5.2.23, 4.5.3.8, 4.5.3.12, and 4.5.3.19	Failure of safe-geometry confinement due to spill of uranium solution from the system	Accidental criticality from fissile solution not confined in safe geometry	S.C.04, Failure of confinement in safe geometry; spill of fissile material solution

Table 13-12. Adverse Event Summary for Uranium Recovery and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
4.2.3.21, 4.2.4.11, 4.2.6.12, 4.3.3.21, 4.3.4.11, and 4.3.6.12	Failure of safe-geometry confinement due to inadvertent transfer to U-bearing resin to the U IX waste collection tanks through a broken retention element	Accidental criticality from fissile solution not confined in safe geometry	S.C.14, Failure of confinement in safe geometry; transfer of U-bearing resin to U IX waste collection tanks
4.2.5.5, 4.3.1.9, 4.3.5.5, and 4.5.1.5	Failure of safe-geometry confinement due to inadvertent transfer to U-bearing solution to the U IX waste collection tanks	Accidental criticality from fissile solution not confined in safe geometry	S.C.14, Failure of confinement in safe geometry; transfer of U-bearing solution to U IX waste collection tanks
4.2.7.7, 4.3.7.7, and 4.5.3.10	Inadvertent transfer of high uranium-concentration solution or resins to spent resin tanks	Accidental criticality too high of uranium mass in waste stream	S.C.15, Too high of uranium mass in spent resin waste stream
4.2.9.10, 4.2.9.19, 4.2.9.21, 4.2.9.23, 4.2.10.10, 4.2.10.12, 4.3.9.10, 4.3.9.19, 4.3.9.21, 4.3.9.23, 4.3.10.10, and 4.3.10.12	Uranium is inadvertently carried over from the concentrator (1 or 2) to the condenser and subsequently, the condenser condensate collection tanks	Accidental criticality from fissile solution not confined in safe geometry	S.C.09, Carryover of uranium to the condenser or condensate tanks
4.2.9.36 and 4.3.9.36	Uranium solution is transferred via a leak between the process system and heater/cooling jackets or coils on a tank or in an exchanger	Accidental criticality from fissile solution not confined in safe geometry	S.C.07, Uranium-containing solution leaks to chilled water or steam condensate system
4.1.1.8, 4.1.1.22, 4.2.1.9, 4.2.1.17, 4.2.1.23, 4.2.9.11, 4.2.9.14, 4.2.9.17, 4.2.9.23, 4.2.9.30, 4.2.9.32, 4.2.10.14, 4.3.1.10, 4.3.1.18, 4.3.1.24, 4.3.9.11, 4.3.9.14, 4.3.9.17, 4.3.9.23, 4.3.9.30, 4.3.9.32, 4.3.10.14, 4.4.1.8, 4.4.1.22, 4.5.1.9, 4.5.1.22, and 4.5.2.8	Carryover of high-vapor content gases or entrance of solutions into the process ventilation header can cause poor performance of the retention bed materials and release radionuclides	High airborne radionuclide release, affecting workers and the public	S.R.04, Carryover of heavy vapor or solution into the process ventilation header causes downstream failure of retention bed, releasing radionuclides

Table 13-12. Adverse Event Summary for Uranium Recovery and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
4.1.1.10, 4.1.1.15, 4.1.1.23, 4.2.1.11, 4.2.1.15, 4.2.1.24, 4.2.2.1, 4.2.2.4, 4.2.3.11, 4.2.3.13, 4.2.3.18, 4.2.3.22, 4.2.3.23, 4.2.3.24, 4.2.4.10, 4.2.5.10, 4.2.6.11, 4.2.7.8, 4.2.8.11, 4.2.8.16, 4.2.8.23, 4.2.9.16, 4.2.9.28, 4.2.9.34, 4.3.1.12, 4.3.1.16, 4.3.1.25, 4.3.2.1, 4.3.2.4, 4.3.3.11, 4.3.3.13, 4.3.3.18, 4.3.3.22, 4.3.3.23, 4.3.3.24, 4.3.4.10, 4.3.5.10, 4.3.6.11, 4.3.7.8, 4.3.8.11, 4.3.8.16, 4.3.8.23, 4.3.9.16, 4.3.9.28, 4.3.9.34, 4.4.1.10, 4.4.1.15, 4.4.1.23, 4.5.1.11, 4.5.1.15, 4.5.1.23, 4.5.2.10, 4.5.2.15, 4.5.2.23, 4.5.3.8, 4.5.3.12, and 4.5.3.19	High-dose radionuclide solution is spilled from the system	Radiological release of high-dose solution with potential to impact workers, the public, or environment	S.R.01, Spill of product solution in hot cell area
4.2.1.12, 4.2.1.24, 4.2.2.1, 4.2.3.11, 4.2.3.13, 4.2.3.18, 4.2.3.22, 4.2.3.23, 4.2.4.10, 4.2.5.10, 4.2.6.11, 4.2.8.11, 4.2.8.16, 4.2.8.23, 4.2.9.16, 4.2.9.28, 4.2.9.34, 4.2.9.35, 4.3.1.12, 4.3.1.16, 4.3.1.12, 4.3.1.25, 4.3.2.1, 4.3.3.11, 4.3.3.13, 4.3.3.18, 4.3.3.22, 4.3.3.23, 4.3.4.10, 4.3.5.10, 4.3.6.11, 4.3.8.11, 4.3.8.16, 4.3.8.23, 4.3.9.16, 4.3.9.28, 4.3.9.34, 4.3.9.35, 4.4.1.10, 4.4.1.15, 4.4.1.23, 4.5.1.11, 4.5.1.23, 4.5.2.10, 4.5.2.15, 4.5.2.23, and 4.5.3.19	High-dose radionuclide solution is sprayed from the system, causing high airborne radioactivity	Radiological release of high-dose spray that remains suspended in the air, giving high dose to workers or the public	S.R.03, Spray of product solution in hot cell area
4.2.9.37, 4.2.9.36, 4.3.9.36, and 4.3.9.37	High-dose radionuclide solution leaks through an interface between the process system and a heating/cooling jacket coil into a secondary system (e.g., chilled water or steam condensate), releasing radionuclides to workers, the public, and environment	High-dose radionuclide solution that leaks to the environment through another system to expose workers or the public	S.R.13, High-dose, radionuclide-containing solution leaks to chilled water or steam condensate system

Table 13-12. Adverse Event Summary for Uranium Recovery and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
4.1.1.25, 4.2.1.26, 4.2.8.25, 4.3.1.27, 4.3.8.25, 4.4.1.25, 4.5.1.25, 4.5.2.25, and 4.5.3.21	Hydrogen buildup in tanks or system, leading to explosive concentrations	Explosion leading to radiological and criticality concerns	S.F.02, Accumulation of flammable gas in tanks or systems
4.1.1.24, 4.2.1.25, 4.2.8.24, 4.2.10.18, 4.3.1.26, 4.3.8.24, 4.3.10.18, 4.4.1.24, 4.5.1.24, 4.5.2.24, and 4.5.3.20	Higher dose than normal due to double-batching an activity or due to buildup of radionuclides in the system over time	Radiation dose is elevated over normal operational levels, but does not exceed low consequence values for exposure to workers due to shielding	Hot cell shielding is credited as the normal condition, mitigating safety feature for this hazard (adverse condition does not represent failure of the safety function of the IROFS)
4.2.4.8 and 4.3.4.8	High temperature pre-elution or regeneration reagent causes unknown impact on IX resin	Consequence is not fully understood	Tentatively S.R.14
4.2.10.6 and 4.3.10.6	Same as S.C.08 except with low-dose solution from condenser condensate	Low consequence resulting in contaminated system	N/A
4.2.10.8, 4.2.10.11, 4.2.10.17, 4.3.10.8, 4.3.10.11, and 4.3.10.17	Spill or spray of low-dose condensate	Low consequence resulting in contaminated surfaces and dose to worker below intermediate consequence dose levels	N/A

IROFS = items relied on for safety.
IX = ion exchange.
N/A = not applicable.
PHA = process hazards analysis.
U = uranium.

Uranium Recovery Open Item

The following adverse event needs to be further researched.

PHA items 4.2.4.8 and 4.3.4.8 postulate high-temperature 2 molar (M) nitric acid (HNO_3) solution being used on the uranium purification ion-exchange (IX) media as a pre-elution rinse. The consequence of the bounding accident was not fully understood and needs to be further researched. The likelihood was identified as low, as there are no good causes of the high temperature from the supply tank other than an improper mixing sequence. This upset would not cause extremely elevated temperatures nor go undetected.

Table 13-13. Adverse Event Summary for Waste Handling and Identification of Accident Sequences Needing Further Evaluation (2 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
5.1.1.13	High uranium content product solution is directed to the high-dose waste collection tanks by accident	Solution from this tank is solidified in a non-favorable geometry process with potential to result in accident nuclear criticality at the high uranium concentration	S.C.10, Fissile solution in high-dose waste collection tanks (a non-fissile solution boundary)
5.2.1.13 and 5.2.2.13	High uranium content product solution enters the low-dose waste collection tanks by accident	Solution from this tank is solidified in a non-favorable geometry process with potential to result in accidental nuclear criticality at the high uranium concentration	S.C.10, Fissile solution is directed to the low-dose waste collection tank
5.4.1.1	High uranium content accumulates in the TCE reclamation evaporator	The mass of uranium may exceed a safe mass and result in an accidental nuclear criticality without monitoring and controls	S.C.22, High concentration of uranium in the TCE evaporator residue
5.4.2.1	Dissolved uranium products may accumulate in the silicone oil waste stream	The mass of uranium may exceed a safe mass and result in an accidental nuclear criticality without monitoring and controls	S.C.23, High concentration in the spent silicone oil waste
5.1.1.24 and 5.1.4.23	Hydrogen buildup in tanks or system leads to explosive concentrations	Explosion leads to radiological and criticality concern	S.F.02, Accumulation of flammable gas in tanks or systems
5.1.1.4, 5.1.1.16, 5.1.4.4, 5.1.4.15, and 5.1.4.17	Several tank or components vented to the process vessel ventilation system overflow and send high-dose solution into process ventilation system components that exit the hot cell boundary	Radiological release may cause a high-dose exposure to workers and the public	S.R.04, High-dose solution from a tank or component overflows into the process ventilation system, compromising the retention beds
5.1.1.6 and 5.1.4.6	The purge air system (an auxiliary system that originates outside the hot cell boundary) allows high-dose radionuclides to exit the boundary in an uncontrolled manner	Radiological release may cause a high-dose exposure to workers and the public	S.R.16, High-dose solution backflows into the purge air system
5.1.1.10, 5.1.1.14, 5.1.1.22, 5.1.2.26, 5.1.2.31, 5.1.4.10, 5.1.4.13, 5.1.4.21, 5.1.5.16, 5.1.5.19, 5.1.5.20, 5.3.1.14, 5.3.1.17, and 5.3.1.18	Spills from multiple sources; materials originating from high-dose process solutions are spilled from the system or process that normally confines them	Radiological release may cause a high-dose exposure to workers and the public	S.R.01, High-dose solution spill in the hot cell waste handling area

Table 13-13. Adverse Event Summary for Waste Handling and Identification of Accident Sequences Needing Further Evaluation (2 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
5.1.1.21, 5.1.2.28, and 5.1.4.20	Several tanks or components vented to the process vessel ventilation system evolve high liquid vapor concentrations, resulting in accelerated high-dose radionuclide release to the stack from wetted retention beds	Radiological release may cause a high-dose exposure to workers and the public	S.R.04, High-dose radionuclide release due to high vapor content in exhaust
5.1.1.22, 5.1.2.26, 5.1.2.31, 5.1.2.32, 5.1.4.10, and 5.1.4.21	Catastrophic failure of a component (high pressure or detonation) leads to rapid release of solution and higher airborne levels	Radiological release may cause a high-dose exposure to workers and the public	S.R.03, High-dose solution spray events from equipment upsets may cause high airborne radioactivity
5.1.2.9, 5.1.2.18, 5.1.2.19, and 5.1.2.21	Adverse events in the concentrator or evaporator systems lead to carryover of high-dose solution into the condenser, resulting in high-dose radionuclides in the low-dose waste collection tanks	Radiological exposure levels on the low-dose encapsulated waste may exceed intermediate or high consequence levels	S.R.17, Carryover of high-dose solution into condensate (a low-dose waste stream)
5.1.2.33	Normally low-dose vapor in the condenser leaks through the boundary into the chilled water system	Radiological release may cause a high-dose exposure to workers and the public	S.R.13, Process vapor from the evaporator leaks across the condenser cooling coils into the chilled water system
5.1.5.8	High-dose solution is inadvertently misfed into the solidification hopper	Radiological release may cause a high-dose exposure to workers and the public	S.R.18, High-dose solution flows into the solidification hopper
5.5.1.1	Due to several potential initiators, the payload container or the shipping cask of high-dose encapsulated waste is dropped during transfer from the storage location to the conveyance	Radiological issue – Depending on damage from the drop, workers could receive high-dose radiation exposure. Unshielded package may impact dose rates at the controlled area boundary.	S.R.32, Container or cask dropped during transfer

PHA = process hazards analysis.
TCE = trichloroethylene.

Table 13-14. Adverse Event Summary for Target Receipt and Identification of Accident Sequences Needing Further Evaluation (2 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
6.1.2.4, 6.1.2.8, 6.1.2.9, 6.1.2.11, 6.1.2.14, and 6.1.2.15	Handling damage to the target basket fixed-interaction passive design feature leads to accidental nuclear criticality	Accidental nuclear criticality leads to high dose to workers and potential dose to the public	S.C.21, Target basket passive design control failure on fixed interaction spacing
6.1.2.7, 6.1.2.10, 6.2.1.1, 6.2.1.5, 6.2.2.1, 6.2.2.2, 6.2.2.4, 6.2.2.5, 6.2.3.3, 6.2.4.1, 6.2.4.2, 6.2.4.4, 6.2.6.1, 6.2.6.3, and 6.2.6.4	Too much uranium mass is handled at once either through operator error or inattention to housekeeping	Accidental nuclear criticality leads to high dose to workers and potential dose to the public	S.C.02, Operator exceeds batch handling limits during target disassembly operations in the hot cell
6.2.1.6, 6.2.2.9, 6.2.3.4, and 6.2.6.6	Operator accumulates more targets or [Proprietary Information] containers into specific room than allowed and violates interaction control	Accidental nuclear criticality leads to high dose to workers and potential dose to the public	S.C.03, Failure of administrative control on interaction limit during handling of targets and irradiated [Proprietary Information]
6.2.1.3, 6.2.1.4, 6.2.1.5, 6.2.2.2, 6.2.2.4, 6.2.2.6, 6.2.3.1, 6.2.3.2, 6.2.3.3, 6.2.5.1, 6.2.5.3, 6.2.5.4, 6.2.5.8, 6.2.6.1, 6.2.6.2, 6.2.6.3, and 6.2.6.5	Too much uranium in the solid waste container (that is not safe-geometry) entering the solid waste encapsulation process (where moderator will be added in the form of water)	Accidental nuclear criticality leads to high dose to workers and potential dose to the public	S.C.17, [Proprietary Information] residual determination fails, and used target housings have too much uranium in solid waste encapsulation waste stream
6.1.1.5, and 6.1.1.9	Cask involved in an in-transit accident or improperly closed prior to shipment, leading to streaming radiation	High dose to workers during receipt inspection and opening activities	S.R.28, High dose to workers during shipment receipt inspection and cask preparation activities due to damaged irradiated target cask
6.1.1.10	Cask involved in in-transit accident or targets failed during irradiation, leading to excessive offgassing from damaged targets	High dose to workers during receipt inspection and opening activities	S.R.29, High dose to workers from release of gaseous radionuclides during cask receipt inspection and preparation for target basket removal
6.1.1.11, 6.1.1.12, 6.1.2.1, 6.1.2.13, and 6.1.2.16	Seal between cask and hot cell docking port fails from a number of causes	High dose to workers from streaming radiation and/or high airborne radioactivity	S.R.30, Cask docking port failures lead to high dose to workers due to streaming radiation and/or high airborne radioactivity

Table 13-14. Adverse Event Summary for Target Receipt and Identification of Accident Sequences Needing Further Evaluation (2 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
6.1.1.1	Cask involved in a crane movement incident, leading to streaming radiation	High dose to workers during receipt inspection and opening activities	S.R.32, High dose to workers during shipment receipt inspection and cask preparation activities due to damaged cask in crane movement incident
6.1.2.3 and 6.1.2.5	Improper handling activities result in high external dose rates through the hot cell wall when removing the target basket and setting it in the target basket carousel shielded well	High external dose to workers	S.R.19, High target basket retrieval dose rate
6.1.2.10, 6.1.2.15, 6.2.1.5, 6.2.2.2, 6.2.2.4, 6.2.3.3, 6.2.4.2, 6.2.5.4, 6.2.6.1, and 6.2.6.3	[Proprietary Information] spilled or ejected in an uncontrolled manner during various target and container-handling activities or during target-cutting activities	High dose to workers or the public may result from uncontrolled accumulation of irradiated [Proprietary Information]	S.R.20, Radiological spill of irradiated targets in the hot cell area
6.1.2.15	Operations removing the target basket (potentially in a heavy shielding housing) with a hoist leads to striking the wall and damaging the hot cell wall shielding function	High dose to workers due to degraded shielding	S.R.21, Damage to the hot cell wall providing shielding
6.2.4.5	Delays in processing a batch of removed [Proprietary Information] results in long-term heating outside of target housing	High dose to workers from high airborne radioactivity	S.R.22, Decay heat buildup in unprocessed [Proprietary Information] removed from targets leads to higher high dose radionuclide offgassing
6.2.4.6 and 6.2.4.7	Improper venting of the chamber or premature opening of the valve during processing of a previously added batch results in release of high-dose radionuclides to the hot cell space	High dose to workers from high airborne radioactivity	S.R.23, Offgassing from irradiated target dissolution tank occurs when the upper valve is opened
6.2.5.5, 6.2.5.6, and 6.2.5.7	The seal on the bagless transport door fails and leads to high dose radionuclides escaping the hot cell containment or confinement boundary	High dose to workers from high airborne radioactivity	S.R.24, Bagless transport door failure

PHA = process hazards analysis.

Table 13-15. Adverse Event Summary for Ventilation System and Identification of Accident Sequences Needing Further Evaluation

PHA item numbers	Bounding accident description	Consequence	Accident sequence
7.1.1.7 and 7.1.1.8	Too much uranium accumulated on the HEPA filter allows an accidental criticality when left in the wrong configuration	Accidental nuclear criticality leads to high dose to workers and potential dose to the public	S.C.24, High uranium content on HEPA filters
7.1.1.2, 7.1.1.3, and 7.1.1.6	Hydrogen buildup in the ventilation system, due to insufficient flow to sweep it away, leads to fire in the HEPA filters or carbon beds	A detonation or deflagration event in the ventilation system rapidly releases retained high-dose radionuclides, causing high airborne radioactivity	S.F.06, Accumulation of flammable gas in ventilation system components
7.1.1.10 and 7.2.1.19	Ignition source causes fire in the carbon bed	Fire event in the ventilation system rapidly releases retained high-dose radionuclides, causing high airborne radioactivity	S.F.05, Fire in the carbon bed
7.1.1.11 and 7.2.1.20	Overloading of HEPA filter leads to failure and release of accumulated radionuclide particulate	High dose to workers from high airborne radioactivity	S.R.25, HEPA filter failure
7.1.1.12, 7.1.1.14, and 7.2.1.21	The accumulated high-dose (and low-dose) radionuclides retained in the carbon bed are released through a flow, heat, or chemical reaction from the media (or the media is released)	High dose to workers from high airborne radioactivity	S.R.04, Carbon bed radionuclide retention failure
7.2.1.4, 7.2.1.7, 7.2.1.8, 7.2.1.9, 7.2.1.13, 7.2.1.14, 7.2.1.17, and 7.2.1.22	Loss of the negative air balance between zones (a confinement feature that prevents migration of radionuclides from areas of high dose and high concentration to areas of low concentration)	High dose to workers from high airborne radioactivity	S.R.26, Failed negative air balance from zone to zone or failure to exhaust a radionuclide buildup in an area
7.2.1.12 and 7.2.1.17	During an extended power outage, some solution systems freeze and cause failure of the piping system, leading to radiological spills	High dose to workers from high airborne radioactivity	S.R.27, Extended outage of heat, leading to freezing, pipe failure, and release of radionuclides from liquid process systems

HEPA = high-efficiency particulate air. PHA = process hazards analysis.

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
8.2.1.5	Large leak leads to localized low oxygen levels that adversely impact worker performance and may lead to death	Standard industrial hazard – Localized asphyxiant	Nitrogen storage or distribution system leak
8.5.1.1 and 8.5.1.5	Operator double-batches allotted amount of material (fresh U, scrap U, [Proprietary Information], target batch) into one location or container during handling	Accidental criticality issue – Too much fissile mass in one location may become critical	S.C.02, Failure of AC on mass (batch limit) during handling of fresh U, scrap U, [Proprietary Information], and targets
8.5.1.3 and 8.5.1.5	Operator handling various containers of uranium or batches of uranium components brings two containers or batches closer together than the approved interaction control distance	Accidental criticality issue – Too much uranium mass in one location	S.C.03, Failure of AC on interaction limit during handling of fresh U, scrap U, [Proprietary Information], and targets
8.6.1.7	A liquid spill of recycle uranium or target dissolution solution occurs within the hot cell boundary	Criticality issue – Fissile solution may collect in unsafe geometry	S.C.04, A liquid spill of fissile solution occurs
8.6.1.9	Process solutions backflow through chemical addition lines to locations outside the hot cell boundary	Criticality issue – Fissile solution may collect in unsafe geometry	S.C.08, Fissile process solutions backflow through chemical addition lines
8.6.1.13	Improper installation of HEPA filters (and prefilters) leads to transfer of fissile uranium particulate into downstream sections of the ventilation system with uncontrolled geometries	Accidental nuclear criticality leads to high dose to worker and potential dose to public	S.C.24, High uranium content on HEPA filters
8.5.1.2 and 8.5.1.5	Operator handling enriched solutions pours solution into an unapproved container	Criticality hazard – Too much uranium mass in one place can lead to accidental nuclear criticality	S.C.27, Failure of AC on volume limit during sampling
8.4.1.8 and 8.6.1.12	Drop of a hot cell cover block or other heavy object damages SSCs relied on for safety	Criticality issue – Structural damage could adversely damage SSCs relied on for safety, leading to accidents with intermediate or high consequence	S.C.28, Crane drop accident over hot cell or other area with SSCs relied on for safety

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
8.1.2.7 and 8.1.2.12	A general facility fire (caused by vehicle accident inside or outside of the facility, wildfire, combustible fire in non-industrial areas, or fire in non-licensed material processing areas) spreads to areas in the building that contain licensed material	Uncontrolled fire can lead to damage to SSCs relied on for safety, resulting in chemical, radiological, or criticality hazards that represent intermediate to high consequence to workers, the public, and environment	S.F.08, General facility fire
8.2.1.7	Leak of hydrogen in the facility attains an explosive mixture and finds an ignition source, leading to detonation or deflagration of the mixture	May lead to an explosion (detonation or deflagration), depending on the location in the facility where the hydrogen leaks from. Explosion may compromise SSCs to various degrees and may lead to intermediate or high consequence events.	S.F.09, Hydrogen explosion in the facility due to a leak from the hydrogen storage or distribution system
8.6.1.11	Electrical fire sparks larger combustible fire in one of the hot cells	Radiological and criticality issue – Depending on the location and quantity of combustibles or flammables left in the area, a fire in the hot cell area could rupture systems with high-dose fission products and/or high uranium content, leading to spills and airborne releases	S.F.10, Combustible fire occurs in hot cell area
8.1.2.9 and 8.4.1.9	A natural gas leak develops in the steam generator room and finds an ignition source, resulting in a detonation or deflagration that damages SSCs	Potential explosion that could catastrophically damage nearby SSCs. Depending on the extent of the damage to SSCs, an accidental nuclear criticality or an intermediate or high consequence exposure to workers could occur.	S.F.11, Detonation or deflagration of natural gas leak in steam generator room
8.1.2.7, 8.3.1.2, and 8.6.1.5	Vehicle inside building strikes fresh uranium dissolution system component, leading to a spill or accidental criticality due to disruption of geometry and/or interaction	Accidental nuclear criticality leads to high dose to workers and potential dose to public	S.M.01, Vehicle strikes SSC relied on for safety and causes damage or leads to an accident sequence of intermediate or high consequence
8.4.1.6	TBD (impact must be evaluated after determining all IROFS that rely on personnel action)	TBD (impact must be evaluated after determining all IROFS that rely on personnel action)	S.M.02, Facility evacuation impacts on operation

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
8.1.2.13	Flooding from external events and internal events compromises the safe geometry slab area under certain tanks. Depending on the liquid level, interspersed moderation of components may be impacted. Floor storage arrays are subject to stored containers floating (loss of interaction control).	Criticality issue – Water accumulation under safe geometry storage vessels or in safe interaction storage arrays, causing interspersed moderation. Flooding could compromise safe-geometry storage capacity for subsequent spills of fissile solution. Either event could compromise criticality safety.	S.M.03, Flooding occurs in building due to internal system leak or fire suppression system activation (likely)
8.1.1.1	Large tornado strikes the facility	Radiological, chemical, and criticality issue – Structural damage could adversely damage SSCs relied on for safety. Facility could lose all electrical distribution. Facility could lose chilled water system function (cooling tower outside of building).	S.N.01, Tornado impact on facility and SSCs
8.1.1.2	Straight-line winds strike the facility	Radiological, chemical, and criticality issue – Structural damage could adversely damage SSCs relied on for safety. Facility could lose all electrical distribution. Facility could lose chilled water system function (cooling tower outside of building).	S.N.02, High straight-line wind impact on facility and SSCs
8.1.1.3	A 48-hr probable maximum precipitation event strikes the facility	Radiological, chemical, and criticality issue – Structural damage from roof collapse could adversely damage SSCs relied on for safety	S.N.03, Heavy rain impact on facility and SSCs
8.1.1.4	Flooding occurs in the area in excess of 500-year return frequency	Radiological issue – Minor structural damage is not anticipated to impact SSCs relied on for safety except that the facility could lose all electrical distribution and/or chilled water system function (cooling tower outside of building)	S.N.04, Flooding impact on facility and SSCs
8.1.1.6	Safe shutdown earthquake strikes – Seismic shaking can lead to damage of the facility and partial to complete collapse. This damage impacts SSCs inside and outside the hot cell boundary. Leaks of fissile solution, compromise of safe-geometry, and safe interaction storage in solid material storage arrays and pencil tanks or vessels containing enriched uranium solutions.	Radiological, chemical, and criticality issue – Structural damage could adversely damage SSCs relied on for safety. Facility could lose all electrical distribution. Facility could lose chilled water system function (cooling tower outside of building).	S.N.05, Seismic impact on facility and SSCs

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
8.1.1.9, 8.1.1.10	Heavy snowfall or ice buildup exceeds design loading of the roof, resulting in collapse of the roof and damage to SSCs (e.g., those outside of the hot cells)	Radiological, chemical, and criticality issue – Structural damage from roof collapse could adversely damage SSCs relied on for safety. Loss of site electrical power is highly likely in heavy ice storm event.	S.N.06, Heavy snowfall or ice buildup on facility and SSCs
8.6.1.8	Any stored high-dose product solution spills within the hot cell boundary	Radiological issue – High-dose solution is unconfined or uncontrolled and can cause exposures to workers, the public, and environment	S.R.01, A liquid spill of high-dose fission product solution occurs
8.5.1.5	Operator spills diluted sample outside of the hot cell area	Radiological issue – Potential spray or vaporization of radionuclide containing vapor-causing adverse worker exposure (based on typical low quantities handled in the laboratory, this is postulated to be an intermediate consequence event)	S.R.01, Spill of product solution in laboratory
8.6.1.10	Recycle uranium transferred out before lag storage decay complete or with significant high-dose radionuclide contaminants	Radiological issue – High radiation may occur in non-hot cell areas, impacting workers with higher than normal external doses	S.R.05, High-dose solution exits hot cell shielding boundary (destined for UN blending and storage tank)
8.6.1.9	Process solutions backflow through chemical addition lines to locations outside the hot cell boundary	Radiological issue – High radiation may occur in non-hot cell areas, impacting workers with higher than normal external doses	S.R.16, High-dose process solutions backflow through chemical addition lines
8.6.1.2 and 8.6.1.3	An improperly sealed cover block or transport door (e.g., for cask transfers) offer large opening potentials for radiation streaming	Radiological issue – Depending on location of damage, some streaming of high radiation may occur, impacting workers with higher than normal external doses	S.R.21, Damage to the hot cell wall penetration, compromising shielding
8.6.1.1	The seal on the bagless transport door fails and leads to high-dose radionuclides escaping the hot cell confinement boundary	Radiological issue – Degraded or loss of cascading negative air pressure between zones may allow high radiological airborne contamination to release without proper filtration and adsorption, leading to higher than allowed exposure rates to workers and the public	S.R.24, Bagless transport door failure
8.6.1.13	Following process upsets and over long periods of operation, contamination levels in downstream components leads to high dose during maintenance and to uncontrolled accumulation of fissile material	Radiological and criticality issue – Following process upsets and over long periods of operation, contamination levels in downstream components can lead to high dose during maintenance and to uncontrolled accumulation of fissile material	S.R.25, HEPA filter failure

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
8.6.1.2, 8.6.1.3, and 8.6.1.6	An improperly sealed cover block or transport door (e.g., for cask transfers) compromises negative air pressure balance	Radiological issue – Degraded or loss of cascading negative air pressure between zones may allow high radiological airborne contamination to release without proper filtration and adsorption, leading to higher than allowed exposure rates to workers and the public	S.R.26, Failed negative air balance from zone to zone or failure to exhaust a radionuclide buildup in an area
8.5.1.7 and 8.5.1.8	Laboratory technician is burned by solutions containing radiological isotopes during sample analysis activities	Radiological issue – Burns may lead to intermediate consequence events if eyes are involved	S.R.31, Chemical burns from contaminated solutions during sample analysis
8.4.1.8, 8.6.1.4, and 8.6.1.12	Drop of a hot cell cover block or other heavy object damages SSCs relied on for safety	Radiological and criticality issue – Structural damage could adversely damage SSCs relied on for safety, leading to accidents with intermediate or high consequence	S.R.32, Crane drop accident over hot cell or other area with SSCs relied on for safety
8.2.1.1	All nitric acid from a nitric acid storage tank is released in 1 hr from the chemical preparation and storage room	Standard industrial accident with potential to impact SSCs or cause additional accidents of concern	S.CS.01, Nitric acid fume release

AC = administrative control.
HEPA = high efficiency particulate air.
IROFS = items relied on for safety.
PHA = process hazards analysis.
SSC = structures, systems, and components.
TBD = to be determined.
U = uranium.
UN = uranyl nitrate.

The identified accident sequences are further evaluated in QRAs to continue the accident analysis and to identify IROFS for those accident sequences that exceed the performance criteria as specified in NWMI-2014-051, *Integrated Safety Analysis Plan for the Radioisotope Production Facility*.

13.2 ANALYSIS OF ACCIDENTS WITH RADIOLOGICAL AND CRITICALITY SAFETY CONSEQUENCES

This section presents an analysis of accident sequences with radiological and criticality safety consequences. In Section 13.1.3, a number of the hazards and accident sequences identified in the PHA that require further evaluation are grouped and identified. These accident sequences were evaluated using both qualitative and quantitative techniques. Accidents for operations with SNM (including irradiated target processing, target material recycle, waste handling, and target fabrication), radiochemical, and hazardous chemicals were analyzed. Initiating events for the analyzed sequences include operator error, loss of power, external events, and critical equipment malfunctions or failures. Shielded and unshielded criticality accidents are assumed to have high consequences to the worker if not prevented.

Most of the quantitative consequence estimates presented in these accident analyses were for releases to an uncontrolled area (public). The worker safety consequence estimates are primarily qualitative. As the design matures, quantitative worker safety consequence analyses will be performed. Updated frequency (likelihood) and the worker and public quantitative safety consequences will be provided in the Operating License Application.

Sections 13.2.2 through 13.2.5 present key representative sequences for radiological and criticality accidents.

- Section 13.2.2 discusses spills and spray accidents with both radiological and criticality safety consequences
- Section 13.2.3 discusses dissolver offgas accidents with radiological consequences
- Section 13.2.4 discusses leaks into auxiliary system accidents with both radiological and criticality safety consequences
- Section 13.2.5 discusses loss of electrical power

These accidents cover failure of primary vessels and piping in the processing areas, loss of fission product gas removal efficiency, leaks into auxiliary systems, and loss of power to the RPF.

Section 13.2.6 briefly presents evaluations of natural phenomena events. The stringent design criteria and requirements for the RPF structure, as discussed in Chapter 3.0, "Design of Structures, Systems, and Components," will require the RPF design to survive certain low-return frequency events. Therefore, the return frequency of most of the external events that the RPF will be designed to withstand are highly unlikely per Table 13-1.

The remainder of the accident sequences, identified in the PHA as requiring further evaluation, are summarized in Section 13.2.7. Each sequence is identified and the associated IROFS (if any) listed. The IROFS not discussed in Sections 13.2.2 through 13.2.6 are also discussed in this section. Numerous accident sequences with both radiological and criticality safety consequences have been evaluated. Some accident sequences are bounded or covered in the preceding accident analysis; others, on further evaluation, have an unmitigated likelihood or consequence that does not require IROFS-level controls.

The discussions that follow form the basis for evaluating the accident sequences at this point in the RPF project development. The additional required information will be provided in the Operating License Application.

13.2.1 Reserved

13.2.2 Liquid Spills and Sprays with Radiological and Criticality Safety Consequences

Liquid solution spill and spray events causing a radiological exposure hazard were identified by the PHA that represent a hazard to workers from direct exposure or inhalation and an inhalation exposure hazard to the public in the unmitigated scenario. The PHA also identified fissile solution leaks with worker safety concerns from a solution-type accidental nuclear criticality. This analysis addresses both of these hazards and identifies controls (in additional to the double-contingency controls identified in Chapter 6.0, "Engineered Safety Features," Section 6.3) to prevent an accidental criticality and reduce exposure from a spray or spill.

13.2.2.1 Initial Conditions

Initial conditions of the process are described by a tank filled with process solution. Multiple vessels are projected to be at initial conditions throughout the process, and the PHA reduced the variety of conditions to the following three configurations that span the range of potential initial conditions:

- A process tank containing low-dose uranium solutions, with no or trace quantities of fission product radionuclides located in a contact maintenance-type of enclosure typical of the target fabrication systems
- A process tank containing high-dose uranium solutions located in a hot cell-type of enclosure typical of the irradiated target dissolution system
- A process tank containing ^{99}Mo product solution located in a hot cell-type of enclosure typical of the molybdenum (Mo) purification system (this condition does not lead to a criticality safety concern)

In each case, a vessel is assumed to be filled with process solution appropriate to the process location with the process offgas ventilation system operating. A level monitoring system is available to monitor tank transfers and stagnant storage volumes on all tanks processing LEU or fission product solutions.

Bounding radionuclide concentrations in liquid streams were developed for five regions of the process in NWMI-2013-CALC-011, *Source Term Calculations*: (1) target dissolution, (2) Mo recovery and purification, (3) uranium recovery and recycle, (4) high-dose liquid waste handling, and (5) low-dose liquid waste handling. The bounding radionuclide concentrations are based on material balances during the processing of MURR targets, which represent the highest target inventory of fission products entering the RPF due to a combination of high target exposure power and short decay time after end of irradiation (EOI). The predicted radionuclide concentrations are increased by 10 percent to address truncating the radioisotope list tracked by material balance calculations for calculation simplification. Predicted batch isotope quantities were further increased by 20 percent as a margin for the radionuclide concentration estimates. This adds a 1.32 margin to the radionuclides stream compositions presented in Chapter 4.0, "Radioisotope Production Facility Description."

Two high-dose uranium solutions located in hot cell enclosures have been evaluated for the Construction Permit Application:

- **Dissolver product in the target dissolution system** – Based on a minimum radionuclide decay time of [Proprietary Information], representing the minimum time for receipt of targets at the RPF
- **Uranium separation feed in the uranium recovery and recycle system** – Based on a radionuclide decay time of [Proprietary Information], representing the minimum lag storage time required for impure uranium solution prior to starting separation of uranium from fission products

The source term used in this analysis is from NWMI-2013-CALC-011. The breakdown of the radionuclide inventory used in NWMI-2013-CALC-011 is extracted from NWMI-2013-CALC-006, *Overall Summary Material Balance – MURR Target Batch*, using the reduced set of 123 radioisotopes. NWMI-2014-CALC-014, *Selection of Dominant Target Isotopes for NWMI Material Balances*, identifies the 123 dominant radioisotopes included in the MURR material balance (NWMI-2013-CALC-006). NWMI-2014-CALC-014 provides the basis for using the 123 radioisotopes from the total list of 660 radioisotopes potentially present in irradiated targets. The majority of omitted radioisotopes exist in trace quantities and/or decay swiftly to stable nuclides. The reduced set of 123 radioisotopes consists of those that dominate the radioactivity and decay heat of irradiated targets.

Bounding solution concentrations from NWMI-2013-CALC-011 are summarized in Table 13-17. Additional conservatism has been incorporated in the dissolver product radionuclide concentrations. The nominal diluted dissolver product volume is [Proprietary Information] dissolver batch. Predicted dissolver product concentrations are increased by a factor of 2.4, to approximate a dissolver product volume of [Proprietary Information] in a dissolver prior to dilution, producing a uranium concentration of [Proprietary Information] (creating a maximum radioactive liquid source term for the RPF). The criticality evaluations also bound the [Proprietary Information] batch size. The uranium separation feed composition reflects planned processing adjustments that reduce the solution uranium concentration to [Proprietary Information]. Note that while most of the radioisotopes concentration are noticeably lower in the uranium separation feed stream of Table 13-17, some daughter isotopes (e.g., americium-241 [^{241}Am]) have increased due to parent decay.

Table 13-17. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{241}Am	[Proprietary Information]	[Proprietary Information]
^{136m}Ba	[Proprietary Information]	[Proprietary Information]
^{137m}Ba	[Proprietary Information]	[Proprietary Information]
^{139}Ba	[Proprietary Information]	[Proprietary Information]
^{140}Ba	[Proprietary Information]	[Proprietary Information]
^{141}Ce	[Proprietary Information]	[Proprietary Information]
^{143}Ce	[Proprietary Information]	[Proprietary Information]
^{144}Ce	[Proprietary Information]	[Proprietary Information]
^{242}Cm	[Proprietary Information]	[Proprietary Information]
^{243}Cm	[Proprietary Information]	[Proprietary Information]
^{244}Cm	[Proprietary Information]	[Proprietary Information]
^{134}Cs	[Proprietary Information]	[Proprietary Information]
^{134m}Cs	[Proprietary Information]	[Proprietary Information]
^{136}Cs	[Proprietary Information]	[Proprietary Information]
^{137}Cs	[Proprietary Information]	[Proprietary Information]
^{155}Eu	[Proprietary Information]	[Proprietary Information]
^{156}Eu	[Proprietary Information]	[Proprietary Information]

Table 13-17. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{157}Eu	[Proprietary Information]	[Proprietary Information]
^{129}I	[Proprietary Information]	[Proprietary Information]
^{130}I	[Proprietary Information]	[Proprietary Information]
^{131}I	[Proprietary Information]	[Proprietary Information]
^{132}I	[Proprietary Information]	[Proprietary Information]
^{132m}I	[Proprietary Information]	[Proprietary Information]
^{133}I	[Proprietary Information]	[Proprietary Information]
^{133m}I	[Proprietary Information]	[Proprietary Information]
^{134}I	[Proprietary Information]	[Proprietary Information]
^{135}I	[Proprietary Information]	[Proprietary Information]
^{83m}Kr	[Proprietary Information]	[Proprietary Information]
^{85}Kr	[Proprietary Information]	[Proprietary Information]
^{85m}Kr	[Proprietary Information]	[Proprietary Information]
^{87}Kr	[Proprietary Information]	[Proprietary Information]
^{88}Kr	[Proprietary Information]	[Proprietary Information]
^{140}La	[Proprietary Information]	[Proprietary Information]
^{141}La	[Proprietary Information]	[Proprietary Information]
^{142}La	[Proprietary Information]	[Proprietary Information]
^{99}Mo	[Proprietary Information]	[Proprietary Information]
^{95}Nb	[Proprietary Information]	[Proprietary Information]
^{95m}Nb	[Proprietary Information]	[Proprietary Information]
^{96}Nb	[Proprietary Information]	[Proprietary Information]
^{97}Nb	[Proprietary Information]	[Proprietary Information]
^{97m}Nb	[Proprietary Information]	[Proprietary Information]
^{147}Nd	[Proprietary Information]	[Proprietary Information]
^{236m}Np	[Proprietary Information]	[Proprietary Information]
^{237}Np	[Proprietary Information]	[Proprietary Information]
^{238}Np	[Proprietary Information]	[Proprietary Information]
^{239}Np	[Proprietary Information]	[Proprietary Information]
^{233}Pa	[Proprietary Information]	[Proprietary Information]
^{234}Pa	[Proprietary Information]	[Proprietary Information]
^{234m}Pa	[Proprietary Information]	[Proprietary Information]
^{112}Pd	[Proprietary Information]	[Proprietary Information]
^{147}Pm	[Proprietary Information]	[Proprietary Information]
^{148}Pm	[Proprietary Information]	[Proprietary Information]
^{148m}Pm	[Proprietary Information]	[Proprietary Information]

Table 13-17. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{149}Pm	[Proprietary Information]	[Proprietary Information]
^{150}Pm	[Proprietary Information]	[Proprietary Information]
^{151}Pm	[Proprietary Information]	[Proprietary Information]
^{142}Pr	[Proprietary Information]	[Proprietary Information]
^{143}Pr	[Proprietary Information]	[Proprietary Information]
^{144}Pr	[Proprietary Information]	[Proprietary Information]
^{144m}Pr	[Proprietary Information]	[Proprietary Information]
^{145}Pr	[Proprietary Information]	[Proprietary Information]
^{238}Pu	[Proprietary Information]	[Proprietary Information]
^{239}Pu	[Proprietary Information]	[Proprietary Information]
^{240}Pu	[Proprietary Information]	[Proprietary Information]
^{241}Pu	[Proprietary Information]	[Proprietary Information]
^{103m}Rh	[Proprietary Information]	[Proprietary Information]
^{105}Rh	[Proprietary Information]	[Proprietary Information]
^{106}Rh	[Proprietary Information]	[Proprietary Information]
^{106m}Rh	[Proprietary Information]	[Proprietary Information]
^{103}Ru	[Proprietary Information]	[Proprietary Information]
^{105}Ru	[Proprietary Information]	[Proprietary Information]
^{106}Ru	[Proprietary Information]	[Proprietary Information]
^{122}Sb	[Proprietary Information]	[Proprietary Information]
^{124}Sb	[Proprietary Information]	[Proprietary Information]
^{125}Sb	[Proprietary Information]	[Proprietary Information]
^{126}Sb	[Proprietary Information]	[Proprietary Information]
^{127}Sb	[Proprietary Information]	[Proprietary Information]
^{128}Sb	[Proprietary Information]	[Proprietary Information]
^{128m}Sb	[Proprietary Information]	[Proprietary Information]
^{129}Sb	[Proprietary Information]	[Proprietary Information]
^{151}Sm	[Proprietary Information]	[Proprietary Information]
^{153}Sm	[Proprietary Information]	[Proprietary Information]
^{156}Sm	[Proprietary Information]	[Proprietary Information]
^{89}Sr	[Proprietary Information]	[Proprietary Information]
^{90}Sr	[Proprietary Information]	[Proprietary Information]
^{91}Sr	[Proprietary Information]	[Proprietary Information]
^{92}Sr	[Proprietary Information]	[Proprietary Information]
^{99}Tc	[Proprietary Information]	[Proprietary Information]
^{99m}Tc	[Proprietary Information]	[Proprietary Information]

Table 13-17. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{125m}Te	[Proprietary Information]	[Proprietary Information]
^{127}Te	[Proprietary Information]	[Proprietary Information]
^{127m}Te	[Proprietary Information]	[Proprietary Information]
^{129}Te	[Proprietary Information]	[Proprietary Information]
^{129m}Te	[Proprietary Information]	[Proprietary Information]
^{131}Te	[Proprietary Information]	[Proprietary Information]
^{131m}Te	[Proprietary Information]	[Proprietary Information]
^{132}Te	[Proprietary Information]	[Proprietary Information]
^{133}Te	[Proprietary Information]	[Proprietary Information]
^{133m}Te	[Proprietary Information]	[Proprietary Information]
^{134}Te	[Proprietary Information]	[Proprietary Information]
^{231}Th	[Proprietary Information]	[Proprietary Information]
^{234}Th	[Proprietary Information]	[Proprietary Information]
^{232}U	[Proprietary Information]	[Proprietary Information]
^{234}U	[Proprietary Information]	[Proprietary Information]
^{235}U	[Proprietary Information]	[Proprietary Information]
^{236}U	[Proprietary Information]	[Proprietary Information]
^{237}U	[Proprietary Information]	[Proprietary Information]
^{238}U	[Proprietary Information]	[Proprietary Information]
^{131m}Xe	[Proprietary Information]	[Proprietary Information]
^{133}Xe	[Proprietary Information]	[Proprietary Information]
^{133m}Xe	[Proprietary Information]	[Proprietary Information]
^{135}Xe	[Proprietary Information]	[Proprietary Information]
^{135m}Xe	[Proprietary Information]	[Proprietary Information]
^{89m}Y	[Proprietary Information]	[Proprietary Information]
^{90}Y	[Proprietary Information]	[Proprietary Information]
^{90m}Y	[Proprietary Information]	[Proprietary Information]
^{91}Y	[Proprietary Information]	[Proprietary Information]
^{91m}Y	[Proprietary Information]	[Proprietary Information]
^{92}Y	[Proprietary Information]	[Proprietary Information]
^{93}Y	[Proprietary Information]	[Proprietary Information]
^{93}Zr	[Proprietary Information]	[Proprietary Information]
^{95}Zr	[Proprietary Information]	[Proprietary Information]
^{97}Zr	[Proprietary Information]	[Proprietary Information]
Totals	[Proprietary Information]	[Proprietary Information]

Source: Table 2-1 of NWMI-2013-CALC-011, *Source Term Calculations*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2015.

EOI = end of irradiation.

13.2.2.2 Identification of Event Initiating Conditions

The accident initiating event is generally described as a process equipment failure, but also could be operator error or initiated by a fire/explosion. Multiple mechanisms were identified during the PHA that resulted in the equivalent of a failure that spills or sprays the tank contents, resulting in rapid and complete draining of a single tank to the enclosure in the vicinity of the tank location.

13.2.2.3 Description of Accident Sequences

The accident sequence for a tank leak is described as follows.

1. Process vessel fail or personnel error causes the tank contents to be emptied to the vessel enclosure floor in the vicinity of the leaking tank.
2. Tank liquid level monitoring and liquid level detection in the enclosure floor sump region alarms, informing operators that a tank leak has occurred.
3. Processing activities in the affected system are suspended based on location of the sump alarm.
4. Operators identify the location of the leaking vessel and take actions to stop additions to the leaking tank.
5. A final stable condition is achieved when solution accumulated in the sump has been transferred to a vessel available for the particular sump material and removed from the enclosure floor.

The accident sequence for a spray leak is similar to that of a tank leak and is described as follows.

1. The process line, containing pressurized liquid, ruptures or develops a leak during a transfer, spraying solution into the source or receiver tank enclosure and transferring leaked material to an enclosure floor in the vicinity of the leak.
2. Transfer liquid level monitoring and liquid level detection in the enclosure floor sump region alarms, informing operators that a leak has occurred.
3. Processing activities in the affected system are suspended based on location of the sump alarm.
4. Operators identify the location of the leaking vessel and take actions to ensure that the motive force of the leaking transfer line has been deactivated.
5. A final stable condition is achieved when solution accumulated in the sump has been transferred to a vessel available for the particular sump material and removed from the enclosure floor.

Maintenance activities to repair the cause of a tank or spray leak are initiated after achieving the final stable condition.

13.2.2.4 Function of Components or Barriers

The process vessel enclosure floor, walls, and ceiling will provide a barrier that prevents transfer of radioactive material to an uncontrolled area during a liquid spill or spray accident. For accidents involving high-dose uranium solutions and ^{99}Mo product solution, the process vessel enclosure floor, walls, and ceiling will provide shielding for the worker. The enclosure structure barriers are to function throughout the accident until (and after) a stable condition has been achieved.

The process enclosure secondary confinement (or ventilation) system will provide a barrier to prevent transfer of radioactive material to an uncontrolled area during a liquid spill or spray accident from radioactive material in the airborne particulate and aerosols generated by the event. The secondary confinement system is to function throughout the accident until a stable condition has been achieved.

The process enclosure sump system represents a component credited (part of the double-contingency analysis) for preventing the occurrence of a solution-type accidental nuclear criticality due to spills or sprays of fissile material. The sump system is to function throughout the accident until a stable condition has been achieved.

13.2.2.5 Unmitigated Likelihood

A spill or spray can be initiated by operations or maintenance personnel error or equipment failures. Failure rates for tanks, vessels, pipes, and pumps are estimated from WSRC-TR-93-262, *Savannah River Site Generic Data Base Development*. Table 13-2 (Section 13.1.1.1) shows qualitative guidelines for applying the likelihood categories. Operator error and tank failure as initiating events are estimated to have an unmitigated likelihood of "not unlikely."

Additional detailed information describing a quantitative evaluation, including assumptions, methodology, uncertainties, and other data, will be developed for the Operating License Application.

13.2.2.6 Radiation Source Term

The following source term descriptions are based on information developed for the Construction Permit Application. Additional detailed information describing source terms will be developed for the Operating License Application.

13.2.2.6.1 Direct Exposure Source Terms

Liquid spill source terms are dependent on the vessel location in the process system. The following source terms describe the three configurations used to span the range of initial conditions:

- **Low-dose uranium solutions** were bounded by the maximum projected uranium concentration solution in the target fabrication system. The primary attribute of low-dose uranium solutions used for consideration of direct exposure consequences is that fission products have been separated from recycled uranium to allow contact operation and maintenance of the target fabrication system within ALARA (as low as reasonably achievable) guidelines. Chapter 4.0, Section 4.2, shows that a pencil tank of this material would be less than 1 millirem (mrem)/hr; therefore, no radiological IROFS are required for this stream.
- **High-dose uranium solutions** were bounded by a spill from the irradiated target dissolver after dissolution is complete. Dissolution of the targets produces an aqueous solution containing uranyl nitrate, nitric acid, and fission products. The primary attribute of high-dose uranium solutions used for consideration of direct exposure consequences is that equipment operation and maintenance must be conducted in a shielded hot cell environment due to the presence of fission products.
- **^{99}Mo product solution** was bounded by a small solution volume (less than 1 L) containing the weekly inventory of product from processing MURR targets. The product is an aqueous solution containing ~0.2 M sodium hydroxide (NaOH) with a total inventory of 1.3×10^4 curies (Ci) ^{99}Mo.

13.2.2.6.2 Confinement Release Source Terms

Confinement release source terms are based on the five-factor algebraic formula for calculating source terms for airborne release accidents from NUREG/CR-6410, as shown by Equation 13-1.

$$ST = MAR \times DR \times ARF \times RF \times LPF \qquad \text{Equation 13-1}$$

where,

ST	=	Source term (activity)
MAR	=	Material at risk (activity)

DR	=	Damage ratio (dimensionless)
ARF	=	Airborne release fraction (dimensionless)
RF	=	Respirable fraction (dimensionless)
LPF	=	Leak path factor (dimensionless)

Confinement release source terms for spray used the source term parameters listed in Table 13-18. Four source term cases were developed for evaluation based on the two bounding liquid concentrations shown in Table 13-17 and the source term parameter alternatives.

Table 13-18. Source Term Parameters

Parameter[a]	Unmitigated spray release	Mitigated spray release
Material at risk (MAR)	Table 13-17	Table 13-17
Damage ratio (DR)	1.0	1.0
Airborne release fraction (ARF)	0.0001 (1.0 for Kr, Xe, and iodine)[b]	0.0001 (1.0 for Kr, Xe, and iodine)[b]
Respirable fraction (RF)	1.0	1.0
Leak path factor (LPF)	1.0	0.0005 (1.0 for Kr, Xe; 0.1 for iodine)

Source: Table 2-1 of NWMI-2015-RPT-009, *Fission Product Release Evaluation*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2015.

[a] Parameter definitions derived from NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, D.C., March 1998.

[b] Accident dose consequences were found to be sensitive to iodine source term parameters. Further work may allow for a lower iodine ARF.

Kr = krypton. Xe = xenon.

The DR was set to 1.0 for all cases. The assumed volume was 100 L of solution contained by a vessel being affected by the spill or spray release.

The ARF and RF values are functions of the release mechanism and do not enter into consideration for a mitigated versus unmitigated release. Thus, for both the unmitigated and mitigated cases, the ARF and RF were set to representative values based on the guidance in NUREG/CR-6410 and DOE-HDBK-3010, *DOE Handbook – Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*. A spray release due to rupture of a pressurized pipe (transfer line) is modeled as depressurization of a liquid through a leak below the liquid surface level. Both NUREG/CR-6410 and DOE-HDBK-3010 report an ARF of 1×10^{-4} and a RF of 1.0 for a spray leak involving a low temperature aqueous liquid.

These values take into consideration upstream pressures as high as 200 pounds (lb)/square inch (in.^{2}) gauge. The spray mechanism is also bounded by a droplet size distribution produced from commercial spray nozzles. This approach is conservative, as the effective nozzle created by a pipe failure is unlikely to be optimized to the extent of a manufactured spray nozzle. Therefore, an ARF of 1×10^{-4} and a RF of 1.0 were used for all isotopes, except iodine and the noble gas fission products Kr and Xe. Radioisotopes of Kr, Xe, and iodine were assigned an ARF of 1.0 for all cases.

For the unmitigated evaluations, the LPF was set to 1.0, since the unmitigated release scenario credits no confinement measures (i.e., no credit was taken for any aspect of the facility design or equipment performance). The gravitational settling associated with flow throughout the facility and the removal action of high-efficiency particulate air (HEPA) filtration may be lumped into an effective value for LPF. The performance of different filtration systems is presented in Appendix F of DOE-HDBK-3010. For scoping purposes, a HEPA filtration efficiency of 99.95 percent was selected for all mitigated cases, which corresponds to an LPF of 0.0005.

The HEPA filter LPF was applied to all isotopes except Kr, Xe, and iodine. An LPF of 1.0 was selected for Kr and Xe in the mitigated spray release evaluation, assuming these isotopes behave as a gas when airborne and are not removed by HEPA filtration or sufficiently retained on the high-efficiency gas adsorption (HEGA) modules. The mitigated analysis credits an iodine removal capability in the facility ventilation exhaust gas equipment, with an iodine removal efficiency of 90 percent. The credited removal efficiency corresponds to an LPF of 0.1 for iodine due to the HEGA modules co-located with the HEPA filters.

13.2.2.7 Evaluation of Potential Radiological Consequences

Confinement release consequence estimates for the Construction Permit Application are based on NUREG-1940, *RASCAL 4: Description of Models and Methods*, and Radiological Safety Analysis Code (RSAC), Version 6.2 (RSAC 6.2). Additional detailed information describing validation of models, codes, assumptions, and approximations will be developed for the Operating License Application.

13.2.2.7.1 Direct Exposure Consequences

The potential radiological exposure hazard of liquid spills depends on the vessel location in the process system. Low-dose uranium solutions are generally contact-handled, and direct radiation exposure to the worker is expected to be slightly elevated but well within ALARA guidelines. Therefore, no IROFS are required to control radiation exposure from spilled low-dose uranium solutions.

Vessels located within hot cells require shielding to control worker radiation exposure independent of whether process solution is contained in the vessel or spilled to the enclosure floor. High-dose uranium solutions are assumed to require hot cell shielding. Spills of ^{99}Mo solution from the Mo recovery and purification processes, and during handling prior to shipment of the product, involve product solution that contains high-dose ^{99}Mo. The direct whole-body exposure from radiation does not change from the normal case and must always be shielded to reduce the dose rate for workers to ALARA. As a preliminary estimate using a point-source dose rate conversion factor for ^{99}Mo of 0.112924 roentgen equivalent man (rem)/hr at 1 meter (m) per Ci ^{99}Mo, the unshielded dose rate for the product is: MAR = 1.3×10^4 Ci ^{99}Mo.

$$^{99}\text{Mo dose rate at 1 m} = 1.30\times10^4 \text{ Ci } ^{99}\text{Mo} \times 0.1129 \text{ rem/hr/Ci } ^{99}\text{Mo} = 1.5\times10^3 \text{ rem/hr}$$

In a very short period of time, a worker can receive a significant intermediate or high consequence dose. Therefore, both high-dose uranium and ^{99}Mo product solution vessels must be located in hot cells for normal operations to control the direct exposure to workers.

Based on the analysis of several accidental nuclear criticalities in industry, LA-13638, *A Review of Criticality Accidents*, identifies that a uranium solution criticality can yield between 10^{16} to 10^{17} fissions. Dose rates for anyone in the target fabrication area can have high consequences. Consequences for a shielded hot cell criticality will be developed for the Operating License Application.

13.2.2.7.2 Confinement Release Consequence

Receptor dose consequences were originally evaluated in NWMI-2015-RPT-009, *Fission Product Release Evaluation*, using the RASCAL code. Since the submission of the application, NWMI has selected RSAC 6.2 for off-site accident consequence modeling. For the liquid spills and spray accident, NWMI has rerun the dissolver product off-site dose calculations using RSAC 6.2. Four release consequence estimates were prepared to support the Construction Permit Application based on unmitigated and mitigated spray release events using the two liquid radionuclide concentrations shown in Table 13-18. The RSAC inputs for the dissolver product accident are listed below, and the RASCAL inputs for the high dose uranium solution are listed in Table 13-19. The uranium feed modeling will be rerun using RSAC 6.2 as part of the Operation License Application.

Table 13-19. Release Consequence Evaluation RASCAL Code Inputs

Input	Description
Primary tool	STDose – Source term to dose option selected as the primary tool in RASCAL for all cases.
Event type	Other release – RASCAL includes separate models for nuclear power plant accidents involving spent fuel, accidents involving fuel cycle activities, and other radioactive material releases at non-reactor facilities. The other radioactive material releases option was selected for all cases.
Facility location[a]	Columbia, Missouri
County	Boone
Time zone	Central
Latitude/longitude	38.9520° N/92.3290° W
Elevation	231 m
Plume rise	None – For scoping purposes, the enthalpy and momentum of the RPF stack exhaust was assumed negligible.
Meteorology	Summer-night-calm – Selected for scoping purposes and features wind speed of 6.4 km/hr (4 mi/hr), Pasquill Class F stability, no precipitation, relative humidity of 80%, and ambient temperature of 12.8°C (55°F). Low wind speed and stable conditions selected to provide maximum dose to near-field receptors.
Receptor distance	100 m – Selected to approximate site boundary. Input represents minimum value for RASCAL input.
Dose conversion factors	ICRP-72[b] – Selected as the most current and authoritative set of dose conversion factors available.

Source: Table 2-1 of NWMI-2015-RPT-009, *Fission Product Release Evaluation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2015.

[a] Location information obtained from Wikipedia.

[b] ICRP-72, *Age-Dependent Doses to the Members of the Public from Intake of Radionuclides – Part 5 Compilation of Ingestion and Inhalation Coefficients*, International Commission on Radiological Protection, Ottawa, Canada, 1995.

RASCAL = Radiological Assessment System for Consequence Analysis.
RPF = Radioisotope Production Facility.

RSAC 6.2 was used to model the dispersion resulting from a spray leak. The following parameters were used for model runs:

- Mixing depth: 400 m (1,312 feet [ft]) (default)
- Air density: 1,240 g/cubic meter [m^3] (1.24 ounce [oz]/cubic feet [ft^3]) (sea level)
- Pasquill-Gifford σ (NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*)
- No plume rise (i.e., buoyancy or stack momentum effects)
- No plume depletion (wet or dry deposition)
- 1-hr release (constant release of all activity)
- 1-hr exposure
- ICRP-30, *Limits for Intakes of Radionuclides by Workers*, inhalation model
- Finite cloud immersion model
- Breathing rate: 3.42E-4 m^3/second (sec) (1.2E-2 ft^3/sec) (ICRP-30 heavy activity)

Consequence evaluation results are shown in Figure 13-2 and Table 13-20 for a 100 L (26.4 gal) spray release event. NWMI is considering the unmitigated spray release of dissolver product solution as an off-site public intermediate consequence event (pending completion of the final safety analysis). The nearest permanent resident, at 432 m (0.27 miles [mi]), dissolver product spray unmitigated dose estimate is 300 mrem, while the maximum receptor location (1,100 m [0.68 mi]) has a total effective dose equivalent (TEDE) of 1.8 rem. The mitigated consequences are an order of magnitude lower due to the credited IROFS in the Zone I exhaust system. Therefore, the nearest permanent resident (432 m [0.27 mi]) dissolver product spray mitigated dose estimate is 30 mrem, while the maximum receptor location (1,100 m [0.68 mi]) has a TEDE of 0.18 rem.

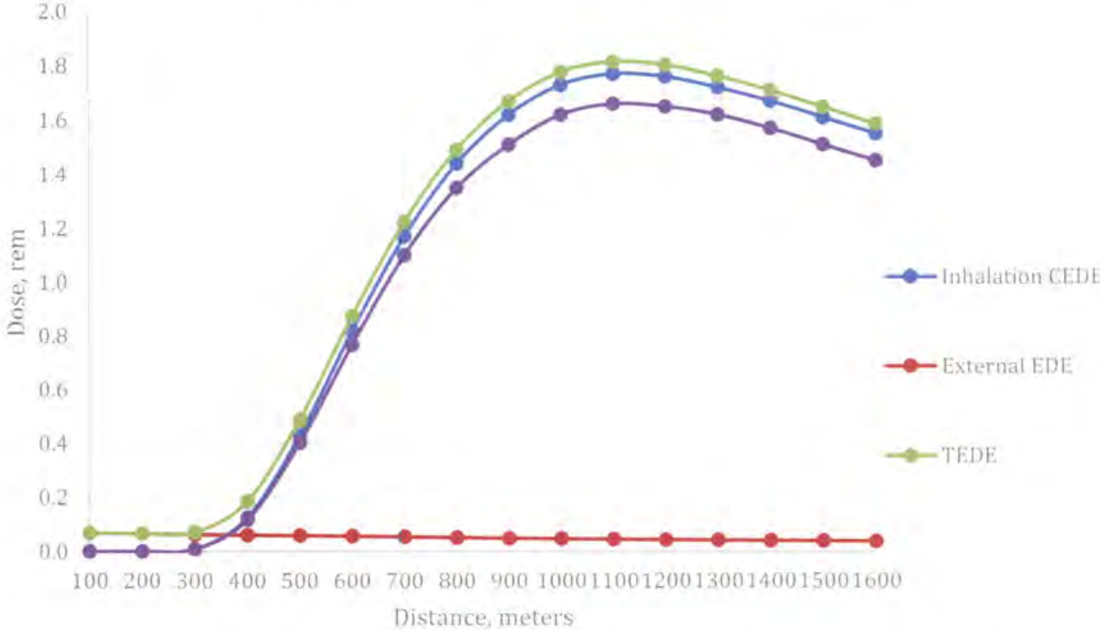

Figure 13-2. Unmitigated Off-Site Dose of Dissolver Product Spray Leak Accident

Table 13-20 shows that the uranium separation feed solution spray release unmitigated dose is below the immediate consequences thresholds of 10 CFR 70.61. Even though this receptor dose is at 100 m, the uranium feed modeling will be rerun using RSAC 6.2 as part of the Operation License Application.

Table 13-20. Uranium Separations Feed Spray Release Consequence Summary at 100 Meters

Process stream	Uranium separations feed	
Case	3	4
Mitigation	Unmitigated	Mitigated
Receptor dose, total EDE	**0.078 rem**	**0.006 rem**
Stack height	10 m (33 ft)[a]	23 m (75 ft)
Release mechanism	Spray leak, 100 L	
Release duration	1 hr	

Source: Table 2-1 and Table 2-7 of NWMI-2015-RPT-009, *Fission Product Release Evaluation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2015.

[a] Lowest value for plume height available as input to RASCAL and recommended by help file as input modeling a ground-level release.

EDE = effective dose equivalent. RASCAL = Radiological Assessment System for Consequence Analysis.

13.2.2.8 Identification of Items Relied on for Safety and Associated Functions

Unmitigated spill and spray releases have the potential to produce direct exposure and confinement releases with high consequence to workers and the public. Hot cell shielding is designed to provide protection from uncontrolled liquid spills and sprays that result in redistribution of high-dose uranium and ^{99}Mo product solution in the hot cell. From a direct exposure perspective, a liquid spill does not represent a failure or adverse challenge to the hot cell shielding boundary function. However, the hot cell shielding boundary must also function to prevent migration of liquid spills to uncontrolled areas outside the shielding boundary.

Liquid spill and spray-type releases occur as a result of the partial failure of process vessels to contain either the fissile solution (for areas outside of the hot cell) or to contain fissile or high-dose radiological solutions (for areas inside the hot cell). In either case, the process vessel spray release results in an event that carries with it a higher airborne radionuclide release magnitude than a simple liquid spill. The spray-type release also carries the extra hazard of potential chemical burns to eyes and skin, with the complication of radiological contamination. Consequently, spray protection is a secondary safety function needed to satisfy performance criteria. The liquid spill and spray confinement safety function of the hot cell liquid confinement boundary is then credited for confining the spray to the hot cell and protecting the worker from sprays of radioactive caustic or acidic solution with the potential to cause intermediate or high consequences. The airborne filtering safety feature of the hot cell secondary confinement boundary is credited with reducing airborne concentrations in the hot cells to levels outside the hot cell boundary, which are below intermediate consequence levels for workers and the public during the event.

Three IROFS are identified to control liquid spill and spray accidents from process vessels.

- IROFS RS-01, "Hot Cell Liquid Confinement Boundary"
- IROFS RS-03, "Hot Cell Secondary Confinement Boundary"
- IROFS RS-04, "Hot Cell Shielding Boundary"

Liquid spill and spray events involving solutions containing fissile material have the potential for producing liquid nuclear criticalities that must be prevented. The following IROFS are identified to control nuclear criticality aspects of the liquid spill and spray events.

- IROFS CS-07, "Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels"
- IROFS CS-08, "Floor and Sump Geometry Control on Slab Depth, Sump Diameter or Depth for Floor Spill Containment Berms"
- IROFS CS-09, "Double-Wall Piping"

Functions of the identified IROFS are described in the following sections.

13.2.2.8.1 IROFS RS-01, Hot Cell Liquid Confinement Boundary

IROFS RS-01 functions to mitigate the impact of liquid spills from process vessels in the hot cells. As a passive engineered control (PEC) and safety feature, the hot cell liquid confinement boundary will provide an integrated system of features that protects workers and the public from the high-dose radiation generated during primary confinement releases of primarily liquid solutions during the ^{99}Mo recovery process. The hot cell liquid confinement boundary will also protect the environment from releases of product solution from the primary confinement of the processing vessels. In addition, the barrier will provide a function of confining spills of irradiated LEU target solid material in some of the irradiated target handling hot cells.

The primary safety function of the hot cell liquid confinement boundary is to capture and contain liquid releases and to prevent those releases from exiting the boundary, causing high dose to workers or the public, or contaminating the environment. A secondary function of the liquid confinement boundary is to prevent contact chemical exposure to workers from acidic or caustic solutions contaminated with licensed material that exceeds the performance criteria established by NWMI for the RPF.

As a PEC to contain spills and sprays of high-dose product solution, the hot cell liquid confinement boundary will consist of sealed flooring with multiple layers of protection from release to the environment. Various areas will be diked to contain specific releases, and sumps of appropriate design will be provided with remote-operated pumps to mitigate liquid spills by capturing the liquid in appropriate safe-geometry tanks. Additional IROFS apply to the flooring and sumps for criticality safety double-contingency controls in some areas. In the ^{99}Mo purification product and sample hot cell, smaller confinement catch basins will be provided under points of credible spill potential in addition to use of a sealed floor. Entryway doors into a designated liquid confinement area will be sealed against credible liquid leaks to outside the boundary. This continuous barrier is also credited to prevent spills or sprays of high-dose product solutions that are acidic or caustic from causing adverse exposure to personnel through direct contact with skin, eyes, and mucus membranes, where the combination of the chemical exposure and the radiological contamination would lead to serious injury and long-lasting effects or even death.

Specific design features of the liquid confinement barrier, a liquid barrier to uncontrolled areas and worker radiation exposure from leaked solution, include:

- Continuous, impervious floor with an acid- or caustic-resistant surface finish
- Hot cell walls and ceiling designed to control worker dose from liquids accumulated in sumps
- Monitors with alarms to indicate a liquid release has occurred
- Sealed penetrations designed to prevent liquid leaks through the barrier to uncontrolled areas
- Sump solution collection vessels for accumulating leaked process solution

13.2.2.8.2 IROFS RS-03, Hot Cell Secondary Confinement Boundary

IROFS RS-03 functions to mitigate the impact of liquid spills and sprays from process vessels in the hot cells. As a system of PECs and AECs, the hot cell secondary confinement boundary safety feature is engineered to provide backup to credible upsets in the primary confinement system using the following safety functions:

- Provide negative air pressure in the hot cell (Zone I) relative to lower zones outside the hot cell using exhaust fans equipped with HEPA filters and HEGA modules to remove the release of radionuclides (both particulate and gaseous) to outside the primary confinement boundary to below 10 CFR 20 release limits during normal and abnormal operations.
- Components credited include:
  - Zone I Inlet HEPA filters to provide an efficiency of 99.97 percent for removal of radiological particulates from the air that may reverse flow from Zone I to Zone II
  - Zone I ducting to ensure that negative air pressure can be maintained by conveying exhaust air to the stack
  - Zone I exhaust train HEPA filters to provide 99.97 percent removal of radiological particulates from the air that flows to the stack
  - Zone I exhaust train HEGA modules to provide 90 percent removal of iodine gas from the air that flows to the stack
  - Zone I exhaust stack to provide dispersion of radionuclides in normal and abnormal releases at a discharge point of 22.9 m (75 ft) above the building ground level

- Stack monitoring and interlocks to monitor discharge and signal changing on service filter trains during normal and abnormal operations

As a system of PECs and AECs, the purpose of this IROFS is to mitigate high-dose radionuclide releases to maintain exposure to acceptable levels to both the worker and the public in a highly reliable and available manner. The hot cell secondary confinement boundary will perform this function using the following engineered features to ensure a high level of reliability and availability.

- As a PEC, the hot cell floor, walls, ceilings, and penetrations are designed to provide an air intrusion barrier sufficient to allow the exhaust system to maintain negative air pressure under normal and credible abnormal conditions. This barrier is not required to be air-tight, but must be controlled to the extent that the design capacity of the exhaust fans can maintain negative pressure. Design features associated with this function include airlocks for normal egress, cask and bagless transfer ports that can only open when the cask or container is properly sealed to the port, and appropriately sized ventilation ports between zones.
- Along with the AECs of the filtered ventilation system, this boundary will provide secondary confinement and prevent uncontrolled release of general radiological airborne gases and particulates that escape the primary confinement to reduce releases to the monitored stack to acceptable release levels during normal and abnormal operations.
- The Zone I exhaust system will serve the hot cell, high-integrity canister (HIC) loading area, and solid waste loading area. This exhaust system will maintain Zone I spaces at negative pressure with respect to atmosphere. All make-up air to Zone I spaces will be cascaded from Zone II spaces.
- HEPA filters will be included on both the inlet and outlet ducts to Zone I. The hot cell outlet HEPA filters will minimize the spread of contamination from the hot cell into the ductwork leading to the exhaust filter train but are not credited with reducing exposure to workers and/or the public. The hot cell inlet HEPA filters will prevent contamination spread during an upset condition that results in positive pressurization of Zone I spaces with respect to Zone II spaces.
- The process offgas subsystem will enter the Zone I exhaust subsystem just upstream of the filter train. The exhaust train outlet HEPA filters will prevent contamination from entering the stack. The stack will disperse radiological gases and particulate to levels below release limits in normal operations and below intermediate consequence levels during process upsets.
- As an AEC, the hot cell secondary confinement system will also serve as backup to the primary offgas treatment system by providing a backup stage of carbon retention bed removal (consisting of an iodine removal) capacity before exhausting into the ventilation system described above. This system will have limited availability for iodine adsorption if the primary system fails.

13.2.2.8.3 IROFS RS-04, Hot Cell Shielding Boundary

IROFS RS-04 functions to prevent worker dose rates from exceeding exposure criteria due to the presence of radioactive materials in the hot cell vessels before or after a liquid spill accident. As a PEC and safety feature, the hot cell shielding boundary will provide an integrated system of features that protect workers from the high-dose radiation generated during the ^{99}Mo recovery process. The primary safety function of the hot cell shielding boundary will be to reduce the radiation dose at the worker/hot cell interface to ALARA. The shield will also protect workers and the public at the controlled area or exclusion area boundary.

The hot cell shielding boundary will provide shielding for workers and the public during normal operations to reduce worker exposure to an average of 0.5 mrem/hr, or less, in normally accessible workstations and occupied areas outside of the hot cell. The hot cell shielding boundary will provide shielding for workers and the public during process upsets to reduce the worker exposure to a TEDE of 5 rem, or less, at workstations and occupied areas outside of the hot cell.

As a PEC, shielding will be provided by a thick concrete, steel-reinforced wall with steel cladding that reduces the normally expected operational exposures from within the boundary to an average of 0.5 mrem/hr, or less, outside of the boundary. Where direct visual access is required, leaded-glass windows with appropriate thicknesses will be used to reduce normally expected operational exposures from within the boundary to an average of 0.5 mrem/hr, or less, outside of the boundary. Some shielding will be movable, such as around the high-dose waste cask loading area. Where penetrations are required, the engineered design provides for access-controlled, non-occupied corridors or airlocks where potential radiation streaming is safely mitigated by multiple layers of shielding or through a torturous path. The shielding is also designed to reduce the exposure from postulated upsets within the hot cell shielding boundary to less than a low-consequence exposure to workers and the public of 5 rem, or less, per incident. These incidents include spills, sprays, fires, and other releases of radionuclides contained within the boundary. The shield may be divided into protection areas for the purposes of applying limiting conditions of operation. Each shielded protected area will be operable when the equipment in that area is in the operating or standby modes.

13.2.2.8.4 IROFS CS-07, Pencil Tank and Vessel Spacing Control using Fixed Interaction Spacing of Individual Tanks or Vessels

IROFS CS-07 functions to ensure that potential interactions between full vessels and a sump filled by a liquid spill or spray have been considered to prevent a nuclear criticality event. As a PEC, pencil tanks and other standalone vessels (controlled with safe geometry or volume constraints) are designed and fabricated with a fixed interaction spacing for safe storage and processing of the fissile solutions. The safety function of fixed interaction spacing of individual barrels in pencil tanks and between other single processing vessels or components is designed to minimize interaction of neutrons between vessels such that under normal and credible abnormal process upsets, the systems will remain subcritical. The fixed interaction control of tanks, vessels, or components containing fissile solutions will prevent accidental nuclear criticality, a high consequence event. The fixed interaction control distance from the safe slab depth spill containment berm is specified where applicable.

13.2.2.8.5 IROFS CS-08, Floor and Sump Geometry Control on Slab Depth, Sump Diameter or Depth for Floor Spill Containment Berms

IROFS CS-08 functions to ensure that sump designs have been considered to prevent a nuclear criticality event by geometry if filled with liquid from a spill or spray release. As a PEC, the floor under designated tanks, vessels, and workstations will be constructed with a spill containment berm that maintains a safe-geometry slab depth to be determined with final design, and one or more collection sumps with diameters or depths to be determined in final design. The safety function of this spill containment berm is to safely contain spilled fissile solution from systems overhead and prevent an accidental nuclear criticality if one of the tanks or related piping leaks, ruptures, or overflows (if so equipped with overflows to the floor). Each spill containment berm will be sized for the largest single credible leak associated with the overhead systems. The interaction distance for the spill containment area is provided in IROFS CS-07. The sump will have a monitoring system to alert the operator that the IROFS has been used and may not be available for a follow-on event. A spill containment berm will be operable if it contains reserve volume for the largest single credible spill. Spill containment berm sizes and locations will be determined by the final design.

13.2.2.8.6 IROFS CS-09, Double-Wall Piping

IROFS CS-09 functions to control liquid spills or sprays in a similar manner to IROFS CS-08. As a PEC, the piping system conveying fissile solution between credited locations will be provided with a double-wall barrier to contain any spills that may occur from the primary confinement piping. IROFS CS-09 is used at locations that pass through the facility where creating a spill containment berm (IROFS CS-08) under the piping is neither practical nor desirable for personnel chemical protection purposes. The double-wall piping arrangement is designed to gravity drain to a safe-geometry set of tanks or to a safe-geometry containment berm. The safety function of this PEC is to safely contain spilled fissile solution from system piping and prevent an accidental nuclear criticality if the primary confinement piping leaks or ruptures. The double-wall piping arrangement will maintain the safe-geometry diameter of the solution. The secondary safety function of the double-wall piping is to prevent personnel injury from exposure to acidic or caustic licensed material solutions that are conveyed in the piping.

Defensive-in-Depth

The following defense-in-depth features were identified by the liquid spill and spray accident evaluations.

- Alarming radiation area monitors will provide continuous monitoring of the dose rate in occupied areas, and alarm at an appropriate setpoint above background.
- Continuous air monitoring will be provided to alert operators of high airborne radiation levels that exceed derived air concentration (DAC) limits.
- HEPA filters on hot cell outlets are not credited and will reduce the impact of spills or sprays to the public.
- Most product solution and uranium solution processing systems will operate at or slightly below atmospheric pressure, or solutions will be pumped between tanks that are at atmospheric pressure to reduce the likelihood of system breach at high pressure.
- Tanks, vessels, components, and piping are designed for high reliability with materials that will minimize corrosion rates associated with the processed solutions.

13.2.2.9 Mitigated Estimates

The controls selected will mitigate both the frequency and consequences of this accident. The controls selected and described above will prevent a criticality associated with accidental spills and sprays of SNM. The selected IROFS have reduced the off-site consequences to acceptable levels (less than 500 mrem to the public). Section 13.2.2.7.2 provides the mitigated public dose estimates. Workers will be protected by the selected secondary confinement and shielding IROFSs. Additional detailed information, including worker dose and detailed frequency estimates, will be developed for the Operating License Application.

13.2.3 Target Dissolver Offgas Accidents with Radiological Consequences

The offgas accident discussed in Chapter 19.0, Environmental Report, is a complete release of the iodine (and noble gases) from a loaded dissolver offgas iodine removal unit (IRU). This accident is the loss of efficiency of the IRU due to a process upset (e.g., flooding of the nitrogen oxide [NO_x] scrubber) or equipment failure (e.g., loss of the IRU heater) during the dissolution of irradiated targets. The primary components of the dissolver offgas include:

- NO_x scrubbers (caustic and absorbers)
- IRUs
- Pressure-relief vessel
- Primary adsorbers (carbon media beds for 6 days noble gas holdup)

- Iodine guard beds (remove any iodine not trapped in the IRUs)
- Filters
- Vacuum receiver tanks
- Vacuum pumps (draw a downstream vacuum on the target dissolver offgas treatment train)
- Secondary adsorbers (additional carbon media beds to hold up noble gases for an additional 60 days)

The IRUs nominally removes about 99.9 percent of the iodine in the offgas stream after the NO_x scrubbers. NWMI expects the availability and operation of IRUs will become part of the technical specification to meet annual release limits. The iodine released from dissolution of the irradiated targets will have three primary pathways: (1) a fraction of the iodine will stay in the dissolver solution (this iodine is a key dose contributor to liquid spills and sprays accidents [see Section 13.2.2]), (2) a significant portion of the iodine gas exiting the dissolver will be captured in the caustic scrubber (and other NO_x treatment absorbers) and end up in the high dose liquid waste tanks, and (3) the remainder of the iodine will be captured in the IRUs.

These IRUs will remove the bulk of the radioactive iodine that passes through the dissolver scrubbers during the dissolution process. As demonstrated by the analysis discussed in Chapter 19.0, iodine will be the greatest contributor to the effective dose equivalent (EDE) for gaseous accident-related releases from the RPF.

The primary and secondary adsorbers will be important for delaying the release of radioactive noble gases (radioisotopes of Kr and Xe) until these isotopes have had time to decay. However, as shown in the analysis in Chapter 19.0, the dose impact of noble gases will be orders of magnitude below that of radioiodine. Therefore, this evaluation focuses on accidents or upsets negatively impacting the IRU performance as the bounding offgas accident.

13.2.3.1 Initial Conditions

The target dissolver and associated offgas treatment train are assumed to be operational and in service prior to the occurrence of any accident sequence that affects the IRUs. The IRUs are assumed to be loaded with the conservative bounding holdup inventory of iodine, as determined in NWMI-2013-CALC-011.

No credible event has been identified where the total captured inventory on the IRUs would be released. This accident evaluation is for the release of the iodine generated from a single dissolution of [Proprietary Information]. The maximum amount of iodine [Proprietary Information] is shown in Table 13-21. The mass balance projects about 20 percent of the iodine will stay in the dissolver solution and nearly 50 percent of the elemental iodine (I_2) that does volatize will be captured in the NO_x scrubbers (primary the caustic scrubber) and transferred to the high dose liquid waste system. However, for this analysis, all of the iodine is assumed to evolve and remain in the offgas stream going to the IRUs.

Table 13-21. Maximum Bounding Inventory of Radioiodine [Proprietary Information]

Isotope	Ci
^{129}I	[Proprietary Information]
^{130}I	[Proprietary Information]
^{131}I	[Proprietary Information]
^{132}I	[Proprietary Information]
^{132m}I	[Proprietary Information]
^{133}I	[Proprietary Information]
^{133m}I	[Proprietary Information]
^{134}I	[Proprietary Information]
^{135}I	[Proprietary Information]
Total I Ci	[Proprietary Information]

I = iodine.

Therefore, this evaluation focuses on accident sequences where the inventory at risk is that generated directly from the dissolution of [Proprietary Information].

13.2.3.2 Identification of Event Initiating Conditions

There are a number of events identified in the PHA that have the potential to impact the normal efficient operation of the target dissolver offgas treatment train. The three most likely sequences with the potential to impact efficient operation include: (1) excessive moisture carryover in the gas stream due to a process upset in the NO_x units, (2) high gas flow rates due to process conditions in the dissolver (e.g., excessive sweep air) or poor NO_x recovery, and (3) loss of temperature control (loss of power or failure of temperature controller) to the IRU. All three of these accidents have the potential to reduce the IRU efficiency.

13.2.3.3 Description of Accident Sequences

The accident sequences for loss of IRU efficiency include the following.

- [Proprietary Information] is being dissolved.
- A process upset occurs that reduces the IRU efficiency by an unspecified amount.
- The event is identified by the operator either from a process control alarm (e.g., low heater temperature) or a radiation alarm on the gas stream or piping exiting the hot cell.
- Following procedure, the operator turns the steam off to the dissolver (to slow down the dissolution process).
- The operator troubleshoots the upset condition and switches to the back IRU, if warranted, and/or manually opens the valve to the pressure-relief tank in the dissolver offgas system to capture the offgas stream.

If the initiator for the event is loss of power or the event creates a condition where vacuum in the dissolver offgas system is lost, the pressure-relief tank valve would automatically open to capture the offgas stream. This tank has been sized to contain the complete gas volume of a dissolution cycle.

13.2.3.4 Function of Components or Barriers

The IRUs will be the primary iodine capture devices; however, there will be iodine guard beds downstream of each of the primary noble gas adsorbers. The vent system piping will direct the dissolver offgas to the pressure-relief tank or through the guard beds and into the primary process vessel vent system. This system will also have iodine removal beds located downstream of the point where the target dissolver offgas treatment train discharges into the process vessel vent system. Thus, the system will provide a redundant iodine removal capacity that backs up the target dissolver offgas treatment train IRUs. The process vessel vent system will discharge to the Zone I exhaust header, which has a HEGA module that is a defense-in-depth component for this accident sequence.

13.2.3.5 Unmitigated Likelihood

Loss of iodine removal efficiency can be initiated by operations or maintenance personnel error or equipment failures. Failure rates for tanks, vessels, pipes, and pumps are estimated from WSRC-TR-93-262. Table 13-2 shows qualitative guidelines for applying the likelihood categories. Operator error and equipment failure as initiating events are estimated to have an unmitigated likelihood of "not unlikely."

Additional detailed information describing a quantitative evaluation, including assumptions, methodology, uncertainties, and other data, will be developed for the Operating License Application.

13.2.3.6 Radiation Source Term

The radioiodine inventory is given in Section 13.2.3.1. As discussed with regard to the analysis in Chapter 19.0, the dose consequences of noble gas radioisotopes are orders of magnitude less than that of iodine radioisotopes. Therefore, the iodine source term is the focus of this accident sequence evaluation. No credit is taken for any iodine removal in the dissolver scrubbers or residual iodine remaining in the dissolver solution. Conversely, in this accident, the previous capture iodine is not part of the source term. Therefore, the source term is 27,100 Ci. Additional detailed information describing the validation of models, codes, assumptions, and approximations will be developed for the Operating License Application.

The source term for this accident is based on a set of initial conditions that were designed to bound the credible offgas scenarios. These assumptions include:

- [Proprietary Information]
- All the iodine in the targets released into the offgas system, and no iodine or noble gases captured in the NO_x scrubbers or retained in the dissolver solution
- Iodine removal efficiency of the dissolver offgas IRU goes to zero
- Greater than expected release of material (e.g., no plating out of iodine, or subsequent iodine capture in downstream of unit operations)

The bounding iodine value includes the 1.32 safety factor used in NWMI-2013-CALC-011. The breakdown of the radionuclide inventory used in NWMI-2013-CALC-011 is extracted from NWMI-2013-CALC-006 using the reduced set of 123 radioisotopes. NWMI-2014-CALC-014 identifies the 123 dominant radioisotopes included in the MURR material balance (NWMI-2013-CALC-006). NWMI-2014-CALC-014 provides the basis for using the 123 radioisotopes from the total list of 660 radioisotopes potentially present in irradiated targets. The majority of omitted radioisotopes will exist in trace quantities and/or decay swiftly to stable nuclides. The reduced set of 123 radioisotopes consists of those that dominate the radioactivity and decay heat of irradiated targets.

13.2.3.7 Evaluation of Potential Radiological Consequences

Radiological consequences are bounded by those evaluated in the Section 19.4 analysis. The unmitigated dose consequences should be about 3.4 times less than the results for the public, based on the source term ratio. Realistic radiological consequences are negligible due to the presence of defense-in-depth iodine capabilities in the dissolver offgas system and in the process vessel vent system that backs up the performance of the target dissolver offgas treatment train IRUs. Additional detailed information describing validation of the models, codes, assumptions, and approximations will be developed for the Operating License Application.

Assuming this accident has similar release characteristics as Section 19.4, the radiological dose consequences can be estimated using the ratio of source terms. This is reasonable since a dissolution takes 1 to 2 hr. The entire inventory would also be released over a 2-hr period directly to the 22.9 m (75-ft) stack and into the environment. RSAC 6.2 was used to model the dispersion, and the following parameters were used for model runs:

- Mixing depth: 400 m (1,312 ft) (default)
- Air density: 1,240 g/m^3 (1.24 oz/ft^3) (sea level)
- Pasquill-Gifford σ (NRC Regulatory Guide 1.145)
- No plume rise (i.e., buoyancy or stack momentum effects)

- No plume depletion (wet or dry deposition)
- 2-hr release (constant release of all activity)
- 2-hr exposure
- ICRP-30 inhalation model
- Finite cloud immersion model
- Breathing rate: 3.42E-4 m^3/sec (1.2E-2 ft^3/sec) (ICRP-30 heavy activity)
- Respiratory fraction: 1.0

Table 13-22 shows the distance-dependent total receptor accident doses versus distance from the RPF stack for 2-hr exposure. This table was developed using the results from the Section 19.4 dose consequences and dividing by a ratio of the accident source terms. The maximum public dose is 6.65 rem at 1,100 m.

Table 13-22. Target Dissolver Offgas Accident Total Effective Dose Equivalent

	TEDE (rem)
Distance (m)	Total
100	2.05E-01
200	1.98E-01
300	2.21E-01
400	6.41E-01
500	1.76E+00
600	3.18E+00
700	4.50E+00
800	5.47E+00
1,000	6.50E+00
1,100	***6.65E+00***
1,200	6.62E+00
1,300	6.50E+00
1,400	6.29E+00
1,500	6.06E+00
1,600	5.82E+00
1,700	2.05E-01

Peak total dose is bolded and italicized.

TEDE = total effective dose equivalent.

RSAC 6.2 calculates inhalation doses using the ICRP-30 model with Federal Guidance Report No. 11 dose conversion factors (EPA 520/1-88-020, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*). The committed dose equivalent (CDE) is calculated for individual organs and tissues over a 50-year period after inhalation.

The CDE for each organ or tissue is multiplied by the appropriate ICRP-26, *Recommendations of the International Commission on Radiological Protection*, weighting factor and then summed to calculate the committed effective dose equivalent (CEDE).

The RSAC 6.2 gamma dose from the cloud is the EDE (the person may or may not be immersed in the cloud depending on the plume position in relation to the ground surface), which is the sum of the products of the dose equivalent to the organ or tissue and the weighting factors applicable to each of the body organs or tissues that is irradiated.

The summation of the two RSAC 6.2 doses is the TEDE, which is the sum of the EDE (for external exposures) and the CEDE (for inhalation exposures).

The RSAC 6.2 dose calculations and dose terminology are consistent with 10 CFR 20 terminology based on ICRP-26/30. The doses and dose commitments (~6.65 rem) are within intermediate consequences severity categories (<25 rem).

13.2.3.8 Identification of Items Relied on for Safety and Associated Functions

IROFS RS-03, Hot Cell Secondary Confinement Boundary

The applicable part of IROFS RS-03 that specifically mitigates target dissolver offgas treatment train IRU failures is the process vessel vent iodine removal beds. These beds are located downstream of where the target dissolver offgas treatment train discharges into the process vessel vent system; hence, the beds provide a backup to the target dissolver offgas treatment train IRUs. IROFS RS-03 is categorized as an AEC.

IROFS RS-09, Primary Offgas Relief System

As an AEC, a relief device will be provided that relieves pressure from the system to an on-service receiver tank maintained at vacuum, with the capacity to hold the gases generated by the dissolution of one batch of targets in the target dissolution tank. The safety function of this system is to prevent failure of the primary confinement system by capturing gaseous effluents in a vacuum receiver. To perform this function, a relief device will relieve into a vacuum receiver that is sized and maintained at a vacuum consistent with containing the capacity of one batch of targets in dissolution.

Defensive-in-Depth

The following defense-in-depth features preventing target dissolver offgas accidents were identified by the accident evaluations.

- Releases at the stack will be monitored for radionuclide emissions to ensure that the overall removal efficiency of the system is reducing emissions to design levels and well below regulator limits.
- A spare dissolver offgas IRU will be available if the online IRU unit loses efficiency.
- The primary carbon retention bed will include an iodine adsorption stage that reduces iodine as a normal backup to the IRU.

13.2.3.9 Mitigated Estimates

The controls selected do not affect the frequency of this accident but mitigate the consequences. The process vessel vent iodine removal bed and the HEGA module in the Zone I exhaust system will mitigate the dose consequences by a factor of 100. The selected IROFS have reduced the off-site consequences to acceptable levels (less than 66 mrem to the public). Additional detailed information, including worker dose estimates and detailed frequency, will be developed for the Operating License Application.

13.2.4 Leaks into Auxiliary Services or Systems with Radiological and Criticality Safety Consequences

In the unmitigated scenario, liquid solution leaks into secondary containment (e.g., cooling water jackets) were identified by the PHA to represent a hazard to workers from direct radiological exposure or inhalation and an inhalation exposure hazard to the public. The PHA also identified fissile solution leaks into secondary containment as an event that could lead to an accidental nuclear criticality. The accidents covered by this analysis bound the family of accidents where highly radioactive or fissile solution leaves the hot cell or other shielded areas via auxiliary systems and creates a worker safety or criticality concern.

13.2.4.1 Initial Conditions

Initial conditions are described as a tank or vessel (with a heating or cooling jacket) filled with process solution. Multiple vessels are projected to be at this initial condition throughout the process. The second primary configuration of concern is the hot cell and target fabrication condensers associated with the four concentrator or evaporator systems. The evaporator(s) initial conditions are normal operations, in which boiling solutions generate an overhead stream that needs to be condensed. The bounding source term is expected to be the dissolvers or the feed tanks in the Mo recovery and purification system. Table 13-23 lists the radionuclide liquid concentration for [Proprietary Information]. The [Proprietary Information] stream is used to represent and bound the uranium recovery and recycle and target fabrication evaporators feed streams.

Table 13-23. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{241}Am	[Proprietary Information]	[Proprietary Information]
^{136m}Ba	[Proprietary Information]	[Proprietary Information]
^{137m}Ba	[Proprietary Information]	[Proprietary Information]
^{139}Ba	[Proprietary Information]	[Proprietary Information]
^{140}Ba	[Proprietary Information]	[Proprietary Information]
^{141}Ce	[Proprietary Information]	[Proprietary Information]
^{143}Ce	[Proprietary Information]	[Proprietary Information]
^{144}Ce	[Proprietary Information]	[Proprietary Information]
^{242}Cm	[Proprietary Information]	[Proprietary Information]
^{243}Cm	[Proprietary Information]	[Proprietary Information]
^{244}Cm	[Proprietary Information]	[Proprietary Information]
^{134}Cs	[Proprietary Information]	[Proprietary Information]
^{134m}Cs	[Proprietary Information]	[Proprietary Information]
^{136}Cs	[Proprietary Information]	[Proprietary Information]
^{137}Cs	[Proprietary Information]	[Proprietary Information]
^{155}Eu	[Proprietary Information]	[Proprietary Information]
^{156}Eu	[Proprietary Information]	[Proprietary Information]
^{157}Eu	[Proprietary Information]	[Proprietary Information]
^{129}I	[Proprietary Information]	[Proprietary Information]
^{130}I	[Proprietary Information]	[Proprietary Information]
^{131}I	[Proprietary Information]	[Proprietary Information]
^{132}I	[Proprietary Information]	[Proprietary Information]
^{132m}I	[Proprietary Information]	[Proprietary Information]
^{133}I	[Proprietary Information]	[Proprietary Information]
^{133m}I	[Proprietary Information]	[Proprietary Information]
^{134}I	[Proprietary Information]	[Proprietary Information]
^{135}I	[Proprietary Information]	[Proprietary Information]
^{83m}Kr	[Proprietary Information]	[Proprietary Information]
^{85}Kr	[Proprietary Information]	[Proprietary Information]
^{85m}Kr	[Proprietary Information]	[Proprietary Information]
^{87}Kr	[Proprietary Information]	[Proprietary Information]
^{88}Kr	[Proprietary Information]	[Proprietary Information]
^{140}La	[Proprietary Information]	[Proprietary Information]
^{141}La	[Proprietary Information]	[Proprietary Information]
^{142}La	[Proprietary Information]	[Proprietary Information]
^{99}Mo	[Proprietary Information]	[Proprietary Information]
^{95}Nb	[Proprietary Information]	[Proprietary Information]
^{95m}Nb	[Proprietary Information]	[Proprietary Information]
^{96}Nb	[Proprietary Information]	[Proprietary Information]

Table 13-23. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{97}Nb	[Proprietary Information]	[Proprietary Information]
^{97m}Nb	[Proprietary Information]	[Proprietary Information]
^{147}Nd	[Proprietary Information]	[Proprietary Information]
^{236m}Np	[Proprietary Information]	[Proprietary Information]
^{237}Np	[Proprietary Information]	[Proprietary Information]
^{238}Np	[Proprietary Information]	[Proprietary Information]
^{239}Np	[Proprietary Information]	[Proprietary Information]
^{233}Pa	[Proprietary Information]	[Proprietary Information]
^{234}Pa	[Proprietary Information]	[Proprietary Information]
^{234m} Pa	[Proprietary Information]	[Proprietary Information]
^{112}Pd	[Proprietary Information]	[Proprietary Information]
^{147}Pm	[Proprietary Information]	[Proprietary Information]
^{148}Pm	[Proprietary Information]	[Proprietary Information]
^{148m}Pm	[Proprietary Information]	[Proprietary Information]
^{149}Pm	[Proprietary Information]	[Proprietary Information]
^{150}Pm	[Proprietary Information]	[Proprietary Information]
^{151}Pm	[Proprietary Information]	[Proprietary Information]
^{142}Pr	[Proprietary Information]	[Proprietary Information]
^{143}Pr	[Proprietary Information]	[Proprietary Information]
^{144}Pr	[Proprietary Information]	[Proprietary Information]
^{144m}Pr	[Proprietary Information]	[Proprietary Information]
^{145}Pr	[Proprietary Information]	[Proprietary Information]
^{238}Pu	[Proprietary Information]	[Proprietary Information]
^{239}Pu	[Proprietary Information]	[Proprietary Information]
^{240}Pu	[Proprietary Information]	[Proprietary Information]
^{241}Pu	[Proprietary Information]	[Proprietary Information]
^{103m}Rh	[Proprietary Information]	[Proprietary Information]
^{105}Rh	[Proprietary Information]	[Proprietary Information]
^{106}Rh	[Proprietary Information]	[Proprietary Information]
^{106m}Rh	[Proprietary Information]	[Proprietary Information]
^{103}Ru	[Proprietary Information]	[Proprietary Information]
^{105}Ru	[Proprietary Information]	[Proprietary Information]
^{106}Ru	[Proprietary Information]	[Proprietary Information]
^{122}Sb	[Proprietary Information]	[Proprietary Information]
^{124}Sb	[Proprietary Information]	[Proprietary Information]
^{125}Sb	[Proprietary Information]	[Proprietary Information]
^{126}Sb	[Proprietary Information]	[Proprietary Information]
^{127}Sb	[Proprietary Information]	[Proprietary Information]
^{128}Sb	[Proprietary Information]	[Proprietary Information]

Table 13-23. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{128m}Sb	[Proprietary Information]	[Proprietary Information]
^{129}Sb	[Proprietary Information]	[Proprietary Information]
^{151}Sm	[Proprietary Information]	[Proprietary Information]
^{153}Sm	[Proprietary Information]	[Proprietary Information]
^{156}Sm	[Proprietary Information]	[Proprietary Information]
^{89}Sr	[Proprietary Information]	[Proprietary Information]
^{90}Sr	[Proprietary Information]	[Proprietary Information]
^{91}Sr	[Proprietary Information]	[Proprietary Information]
^{92}Sr	[Proprietary Information]	[Proprietary Information]
^{99}Tc	[Proprietary Information]	[Proprietary Information]
^{99m}Tc	[Proprietary Information]	[Proprietary Information]
^{125m}Te	[Proprietary Information]	[Proprietary Information]
^{127}Te	[Proprietary Information]	[Proprietary Information]
^{127m}Te	[Proprietary Information]	[Proprietary Information]
^{129}Te	[Proprietary Information]	[Proprietary Information]
^{129m}Te	[Proprietary Information]	[Proprietary Information]
^{131}Te	[Proprietary Information]	[Proprietary Information]
^{131m}Te	[Proprietary Information]	[Proprietary Information]
^{132}Te	[Proprietary Information]	[Proprietary Information]
^{133}Te	[Proprietary Information]	[Proprietary Information]
^{133m}Te	[Proprietary Information]	[Proprietary Information]
^{134}Te	[Proprietary Information]	[Proprietary Information]
^{231}Th	[Proprietary Information]	[Proprietary Information]
^{234}Th	[Proprietary Information]	[Proprietary Information]
^{232}U	[Proprietary Information]	[Proprietary Information]
^{234}U	[Proprietary Information]	[Proprietary Information]
^{235}U	[Proprietary Information]	[Proprietary Information]
^{236}U	[Proprietary Information]	[Proprietary Information]
^{237}U	[Proprietary Information]	[Proprietary Information]
^{238}U	[Proprietary Information]	[Proprietary Information]
^{131m}Xe	[Proprietary Information]	[Proprietary Information]
^{133}Xe	[Proprietary Information]	[Proprietary Information]
^{133m}Xe	[Proprietary Information]	[Proprietary Information]
^{135}Xe	[Proprietary Information]	[Proprietary Information]
^{135m}Xe	[Proprietary Information]	[Proprietary Information]
^{89m}Y	[Proprietary Information]	[Proprietary Information]
^{90}Y	[Proprietary Information]	[Proprietary Information]
^{90m}Y	[Proprietary Information]	[Proprietary Information]
^{91}Y	[Proprietary Information]	[Proprietary Information]

Table 13-23. Bounding Radionuclide Liquid Stream Concentrations (4 pages)

Unit operation	Target dissolution	Uranium recovery and recycle
Decay, hours after EOI	[Proprietary Information]	[Proprietary Information]
Stream description	Dissolver product	Uranium separation feed
Isotope	**Bounding concentration (Ci/L)**	**Bounding concentration (Ci/L)**
^{91m}Y	[Proprietary Information]	[Proprietary Information]
^{92}Y	[Proprietary Information]	[Proprietary Information]
^{93}Y	[Proprietary Information]	[Proprietary Information]
^{93}Zr	[Proprietary Information]	[Proprietary Information]
^{95}Zr	[Proprietary Information]	[Proprietary Information]
^{97}Zr	[Proprietary Information]	[Proprietary Information]
Totals	[Proprietary Information]	[Proprietary Information]

Source: Table 2-1 of NWMI-2013-CALC-011, *Source Term Calculations*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2015.

EOI = end of irradiation.

In each case, a jacketed vessel is assumed to be filled with process solution appropriate to the process location, with the process offgas ventilation system operating. A level monitoring system will be available to monitor tank transfers and stagnant store volumes on all tanks processing LEU or fission product solutions.

The source term used in this analysis is from NWMI-2013-CALC-011. The breakdown of the radionuclide inventory used in NWMI-2013-CALC-011 is extracted from NWMI-2013-CALC-006 using the reduced set of 123 radioisotopes. NWMI-2014-CALC-014 identifies the 123 dominant radioisotopes included in the MURR material balance (NWMI-2013-CALC-006). NWMI-2014-CALC-014 provides the basis for using the 123 radioisotopes from the total list of 660 radioisotopes potentially present in irradiated targets. The majority of omitted radioisotopes exist in trace quantities and/or decay swiftly to stable nuclides. The reduced set of 123 radioisotopes consists of those that dominate the radioactivity and decay heat of irradiated targets.

13.2.4.2 Identification of Event Initiating Conditions

The accident initiating event is generally described as a process equipment failure. The PHA identified similar accident sequences in four nodes associated with leaks of enriched uranium solution into heating and/or cooling coils surrounding safe-geometry tanks or vessels. The PHA identified predominately corrosive degradation of the tank or overpressure of the tank as potential causes that might damage this interface and allow enriched uranium solution to leak into the cooling system media or into the steam condensate for the heating system.

The primary containment fails, which allows radioactive or fissile solutions to enter an auxiliary system. Radioactive or fissile solution leaks across the mechanical boundary between a process vessel and associated heating/cooling jacket into the heating/cooling media. Where heating/cooling jackets or heat exchangers are used to heat or cool a fissile and/or high-dose process solution, the potential exists for the barrier between the two to fail and allow fissile and/or high-dose process solution to enter the auxiliary system. If the auxiliary system is not designed with a safe-geometry configuration, or if this system exits the hot cell containment, confinement, or shielding boundary in an uncontrolled manner, either an accidental criticality is possible or a high-dose to workers or the public can occur.

Where auxiliary services enter process solution tanks, the potential exists for backflow of high-dose radiological and/or fissile process solution into the auxiliary service systems (purge air, chemical addition line, water addition line, etc.). Since these systems are not designed for process solutions, this event can lead to either accidental nuclear criticality or to high-dose radioactive exposures to workers occupying areas outside the hot cell confinement boundary.

13.2.4.3 Description of Accident Sequences

The PHA made no assumption about the geometry or the extent of the heating/cooling subsystem. Consequently, an assumption is made that without additional control, a credible accidental nuclear criticality could occur, as the fissile solution enters into the heating/cooling system not designed for fissile solution, or as the solution exits the shielded area and creates a high worker dose consequence. If the system is not a closed loop, a direct release to the atmosphere can also occur. Either of these potential outcomes can exceed the performance criteria of one process upset, leading to accidental nuclear criticality or a release that exceeds intermediate or high consequence levels for dose to workers, the public, or environment.

The accident sequence for a tank leak into the cooling water (or heating) system includes the following.

- The process vessel wall fails and the tank contents leak into the cooling jacket and medium, or the process medium leaks into the vessel.
- Tank liquid level monitoring and liquid level instrumentation are functional; however, depending on the size of the leak, the tank level instrumentation may or may not detect that a tank has leaked.
- The cooling water system monitor (conductivity or pH) detects a change in the cooling water, and an alarm notifies the operator.
- The operator places the system in a safe configuration and troubleshoots the source of the leak.
- Maintenance activities to identify, repair, or replace the cause of the leak are initiated after achieving the final stable condition.

Additional PHA accident sequences include the backflow (siphon) or backup of process solutions into the chemical or water addition systems. The controls for these accidents are described in Section 13.2.4.8.

13.2.4.4 Function of Components or Barriers

This accident sequence requires the failure of the primary confinement in a safe-geometry vessel or tank, the normal condition criticality safety control for the process. This same barrier will provide primary containment of the high-dose process solution to maintain the solution within the hot cell containment, confinement, and shielding boundary. The heating and cooling systems will have secondary loops (closed loops), so a second failure is required for the fissile solution to enter into a non-geometric-safe auxiliary system or into a non-shielded auxiliary system out of the hot cells.

13.2.4.5 Unmitigated Likelihood

Leaks into auxiliary services can be initiated by mechanical failure of equipment boundaries between the process solutions and auxiliary system fluids, or backflow of high-dose radiological or fissile solution to a chemical supply system. Failure rates for tanks, vessels, pipes, and pumps are estimated from WSRC-TR-93-262. Table 13-2 shows qualitative guidelines for applying the likelihood categories. Failures resulting in leaks or backflows as initiating events are estimated to have an unmitigated likelihood of "not unlikely."

Additional detailed information describing a quantitative evaluation, including assumptions, methodology, uncertainties, and other data, will be developed for the Operating License Application.

13.2.4.6 Radiation Source Term

The following source term descriptions are based on information developed for the Construction Permit Application. Additional detailed information describing source terms will be developed for the Operating License Application.

Source terms associated with leaks and backflows into auxiliary system are dependent on vessel location in the process system. The high-dose uranium solution source term bounds this analysis. Solution leaks into the cooling or heating system were bounded by the irradiated target dissolver after dissolution is complete. The target dissolution process produces an aqueous solution containing uranyl nitrate, nitric acid, and fission products. The fission product inventory is bounded by dissolution of a batch of MURR targets that is decayed [Proprietary Information], with an equivalent uranium concentration of 283 g U/L. The primary attribute of high-dose uranium solutions used for consideration of direct exposure consequences is that equipment operation and maintenance must be conducted in a shielded hot cell environment due to the presence of fission products.

13.2.4.7 Evaluation of Potential Radiological Consequences

The following evaluations are based on information developed for the Construction Permit Application. Additional detailed information describing radiological consequences will be developed for the Operating License Application.

13.2.4.7.1 Direct Exposure Consequences

The potential radiological exposure hazard of liquid spills discussed in Section 13.2.2 bound the consequences from radiation exposure for these accident sequences. Even the low-dose uranium solutions, while generally contact-handled, have similar exposure consequences due to the criticality hazard. Auxiliary systems located within hot cells will require shielding to control worker radiation exposure independent of whether process solution is contained in the vessel or leaked into the auxiliary system. Thus, in a very short period of time, a worker can receive a significant intermediate or high consequence dose rate.

Based on the analysis of several accidental nuclear criticalities in industry, LA-13638 identifies that a uranium solution criticality can yield between 10^{16} to 10^{17} fissions. Dose rates for anyone in the target fabrication area can have high consequences. Consequences for a shielded hot cell criticality will be developed for the Operating License Application.

13.2.4.7.2 Confinement Release Consequences

Not applicable to this accident sequence.

13.2.4.8 Identification of Items Relied on for Safety and Associated Functions

Hot cell shielding is designed to provide protection from leaks into the heating and cooling closed loop auxiliary systems that result in redistribution of high-dose uranium solutions in the hot cell. From a direct exposure perspective, this type of accident does not represent a failure or adverse challenge to the hot cell shielding boundary function.

13.2.4.8.1 IROFS RS-04, Hot Cell Shielding Boundary

IROFS RS-04 functions to prevent worker dose rates from exceeding exposure criteria due to the presence of radioactive materials in the hot cell vessels before or after a leak to the cooling and heating auxiliary systems.

As a PEC and safety feature, the hot cell shielding boundary will provide an integrated system of features that protect workers from the high-dose radiation generated during radioisotope processing. A primary safety function of the hot cell shielding boundary will be to reduce the radiation dose at the worker/hot cell interface to ALARA. While protecting workers, the shield will also protect the public at the controlled area boundary. The hot cell shielding boundary will provide shielding for workers and the public during normal operations to reduce worker exposure to an average of 0.5 mrem/hr, or less, in normally accessible workstations and occupied areas outside of the hot cell.[1] The hot cell shielding boundary will also provide shielding for workers and the public during process upsets to reduce worker exposure to a TEDE of 5 rem, or less, at workstations and occupied areas outside of the hot cell.[2]

As a PEC, shielding will be provided by a thick concrete, steel-reinforced wall with steel cladding that reduces the normally expected operational exposures from within the boundary to an average of 0.5 mrem/hr, or less, outside of the boundary. Where direct visual access is required, leaded-glass windows with appropriate thicknesses will be used to reduce normally expected operational exposures from within the boundary to an average of 0.5 mrem/hr, or less, outside of the boundary. Some shielding will be movable, such as around the high-dose waste cask loading area. Where penetrations are required, the engineered design provides for access-controlled, non-occupied corridors or airlocks where potential radiation streaming is safely mitigated by multiple layers of shielding or through a torturous path. The shielding is also designed to reduce the exposure from postulated upsets within the hot cell shielding boundary to less than a low consequence exposure to workers and the public of 5 rem, or less, per incident. These incidents include spills, sprays, fires, and other releases of radionuclides contained within the boundary. The shield may be divided into protection areas for the purposes of applying limiting conditions of operation. Each shielded protected area will be operable when the equipment in that area is in the operating or standby modes.

13.2.4.8.2 IROFS CS-06, Pencil Tank and Vessel Spacing Control using the Diameter of the Tanks, Vessels, or Piping

All tanks, vessels, or piping systems involved in a process upset will be controlled with a safe-geometry confinement IROFS that consists of IROFS CS-06 to provide a diameter of the vessels confinement or IROFS CS-26 to provide safe volume confinement.

13.2.4.8.3 IROFS CS-10, Closed Safe Geometry Heating or Cooling Loop with Monitoring and Alarm

As a PEC, a closed-loop safe-geometry heating or cooling loop with monitoring for uranium process solution or high-dose process solution will be provided to safely contain fissile process solution that leaks across this boundary, if the primary boundary fails. The dual-purpose safety function of this closed loop is to prevent fissile process solution from causing accidental nuclear criticality and to prevent high-dose process solution from exiting the hot cell containment, confinement, or shielded boundary (or, for systems located outside of the hot cell containment, confinement, or shielded boundary, to prevent low-dose solution from exiting the facility), causing excessive dose to workers and the public, and/or release to the environment.

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[1] Some operations may have higher doses during short periods of the operation. The average worker exposure rate is designed to be 0.5 mrem/hr, or less. Areas not normally accessible by the worker may have higher dose rates (e.g., streaming radiation around normally inaccessible remote manipulator penetrations well above the worker's head).

[2] The shielding is not credited for mitigating dose rates during an accidental nuclear criticality; instead, additional IROFS are identified to provide double-contingency protection to prevent (reduce the likelihood of) an accidental nuclear criticality.

The heat exchanger materials will be compatible with the harsh chemical environment of the tank or vessel process (this may vary from application to application). Sampling of the heating or cooling media (e.g., steam condensate conductivity, cooling water radiological activity, uranium concentration, etc.) will be conducted to alert the operator that a breach has occurred and that additional corrective actions are required to identify and isolate the failed component and restore the closed loop integrity. Discharged solutions from this system will be handled as potentially fissile and sampled according to IROFS CS-16 and CS-17 prior to discharge to a non-safe geometry.

13.2.4.8.4 IROFS CS-27, Closed Heating or Cooling Loop with Monitoring and Alarm

As a PEC, on the evaporator or concentrator condensers, a closed cooling loop with monitoring for breakthrough of process solution will be provided to contain process solution that leaks across this boundary, if the boundary fails. IROFS CS-27 is applied to those high-heat capacity cooling jackets (requiring very large loop heat exchangers) servicing condensers where the leakage is always from the cooling loop to the condenser, reducing back-leakage, and the risk of product solutions entering the condenser is very low by evaporator or concentrator design.

The purpose of this safety function is to monitor the condition of the condenser cooling jacket to ensure that in the unlikely event that a condenser overflow occurs, fissile and/or high-dose process solution will not flow into this non-safe geometry cooling loop and cause nuclear criticality. The closed loop will also isolate any high-dose fissile product solids (from the same event) from penetrating the hot cell shielding boundary, and any high-dose fission gases from penetrating the hot cell shielding boundary during normal operations. The heat exchanger materials will be compatible with the harsh chemical environment of the tank or vessel process (this may vary from application to application). Sampling of the cooling media (e.g., cooling water radiological activity, uranium concentration, etc.) will be conducted to alert the operator that a breach has occurred and that additional corrective actions are required to identify and isolate the failed component and to restore the closed loop integrity. Closed loop pressure will also be monitored to identify a leak from the closed loop to the process system. Discharged solutions from this system will be handled as potentially fissile and sampled according to IROFS CS-16 and CS-17 prior to discharge to a non-safe geometry.

13.2.4.8.5 IROFS CS-20, Evaporator or Concentrator Condensate Monitoring

As an AEC, the condensate tanks will use a continuously active uranium detection system to detect high carryover of uranium that shuts down the evaporator feeding the tank. The purpose of this system is to (1) detect an anomaly in the evaporator or concentrator indicating high uranium content in the condenser (due to flooding or excessive foaming), and (2) prevent high concentration uranium solution from being available in the condensate tank for discharge to a non-favorable geometry system or in the condenser for leaking to the non-safe geometry cooling loop. The safety function of this IROFS is to prevent an accidental nuclear criticality. The detection system will work by continuously monitoring condensate uranium content and detecting high uranium concentration, and then shutting down the evaporator to isolate the condensate from the condenser and condensate tank. At a limiting setpoint, the uranium monitor-detecting device will close an isolation valve in the inlet to the evaporator (or otherwise secure the evaporator) to stop the discharge of high-uranium content solution into the condenser and condensate collection tank.

The uranium monitor is designed to produce a valve-open permissive signal that fails to an open state, closing the valve on loss of electrical power. The isolation valve is designed to fail-closed on loss of instrument air, and the solenoid is designed to fail-closed on loss of signal. The locations where this IROFS is used will be determined during final design.

13.2.4.8.6 IROFS CS-18, Backflow Prevention Device

As a PEC or AEC, chemical and gas addition ports to fissile process solution systems will enter through a backflow prevention device. This device may be an anti-siphon break, an overloop seal, or other active engineering feature that addresses the conditions of backflow and prevents fissile solution from entering non-safe geometry systems or high-dose solutions from exiting the hot cell shielding boundary in an uncontrolled manner. The safety function of this IROFS is to prevent fissile solutions and/or high-dose solutions from backflowing from the tank into systems that are not designed for fissile solutions that could lead to accidental nuclear criticality or to locations outside of the hot cell shielding boundaries that might lead to high exposures to the worker. Each hazardous location will be provided an engineered backflow prevention device that provides high reliability and availability for that location.

The backflow prevention device features for high-dose product solutions will be located inside the hot cell shielding and confinement boundaries of IROFS RS-04 and RS-01, respectively. The feature is designed such that spills from overflow are directed to a safe geometry confinement berm controlled by IROFS CS-08 (described in NWMI-2015-SAFETY-004, *Quantitative Risk Analysis of Process Upsets Associated with Passive Engineering Controls Leading to Criticality Accident Sequences*, Section 3.1.6.3) or to safe-geometry tanks controlled by IROFS CS-11.

13.2.4.8.7 IROFS CS-19, Safe Geometry Day Tanks

As a PEC, safe-geometry day tanks will be provided where the first barrier cannot be a backflow prevention device. The safety function of this PEC is to prevent accidental nuclear criticality by providing a safe-geometry tank if a fissile solution backs-up into an auxiliary chemical addition system. IROFS CS-19 will be used where conventional backflow prevention in pressurized systems is not reliable. The safe-geometry day tank will be provided for those chemical addition activities where the reagent cannot be added via an anti-siphon break since the tank or vessel is not vented and operates under some backpressure conditions. The feature works by providing a safe-geometry vessel that is filled with chemical reagent using the conventional backflow prevention devices, and then provides a pump to add the reagents to the respective process system under pressure. Safe-geometry day tanks servicing high-dose product solutions systems will be located in the hot cell shielding or confinement boundaries of IROFS RS-04 and RS-01, respectively.

Defensive-in-Depth

The following defense-in-depth features preventing leaks into auxiliary services or systems were identified by the accident evaluations.

- All tanks will be vented and unpressurized under normal use.
- The heating and cooling systems will operate at pressures that are higher than the processing systems that they heat or cool. The majority of system leakage would typically be in the direction of the heat transfer media to the processing system.
- All vented tanks are designed with level indicators that are available to the operator to detect the level of solution in a tank remotely. Operating procedures will identify an operational high-level fill operating limit for each tank. As part of the level detector, a high-level audible alarm and light will be provided to indicate a high level above this operating limit so that the operator can take action to correct conditions leading to failure of the operating limit. With batch-type operation with typically low volume transfers, the sizing of the tanks will include sufficient overcapacity to handle reasonable perturbations in operations caused by variations in chemical concentrations and operator errors (adding too much).

- Tank and vessel walls will be made of corrosion-resistant materials and have wall thicknesses that are rated for long service with harsh acid or basic chemicals.
- Purge and gas reagent addition lines (air, nitrogen, and oxygen) will be equipped with check valves to prevent flow of process solutions back into uncontrolled geometry portions (tanks, receivers, dryers, etc.) of the delivery system.

13.2.4.9 Mitigated Estimates

The controls selected will mitigate both the frequency and consequences of this accident. The controls selected and described above will prevent a criticality associated with SNM leaks into auxiliary systems. The selected IROFS have reduced the potential worker safety consequences to acceptable levels. Additional detailed information, including worker dose and detailed frequency estimates, will be developed for the Operating License Application.

13.2.5 Loss of Power

13.2.5.1 Initial Conditions

Initial conditions of the event are described by normal operation of all process systems and equipment.

13.2.5.2 Identification of Event Initiating Conditions

Multiple initiating events were identified by the PHA that could result in the loss of normal electric power.

13.2.5.3 Description of Accident Sequences

The loss of power event sequence includes the following.

1. Electrical power to the RPF is lost due to an initiating event.
2. The uninterruptible power supply automatically provides power to systems that support safety functions, protecting RPF personnel and the public. The following systems are supported with an uninterruptible power supply:
    - Process and facility monitoring and control systems
    - Facility communication and security systems
    - Emergency lighting
    - Fire alarms
    - Criticality accident alarm systems
    - Radiation protection systems
3. Upon loss of power, the following actions occur:
    - Inlet bubble-tight isolation dampers within the Zone I ventilation system close, and the heating, ventilation, and air conditioning (HVAC) system is automatically placed into the passive ventilation mode of operation.
    - Process vessel vent system is automatically placed into the passive ventilation mode of operation, and all electrical heaters cease operation as part of the passive operation mode.
    - Pressure-relief confinement system for the target dissolver offgas system is activated on reaching the system relief setpoint, and dissolver offgas is confined in the offgas piping, vessels, and pressure-relief tank (IROFS RS-09).

- Process vessel emergency purge system is activated for hydrogen concentration control in tank vapor spaces (IROFS FS-03).
- Uranium concentrator condensate transfer line valves are automatically configured to return condensate to the feed tank due to residual heating or cooling potential for transfer of process fluids to waste tanks (IROFS CS-14/CS-15).
- All equipment providing a motive force for process activities cease, including:
  - Pumps performing liquid transfers of process solutions
  - Pumps supporting operation of the steam and cooling utility heat transfer fluids
  - Equipment supporting physical transfer of items (primarily cranes)

4. Operators follow alarm response procedures.
5. The facility is now in a stable condition.

13.2.5.4 Function of Components or Barriers

All facility structural components of the hot cell secondary confinement boundary (in a passive ventilation mode) and hot cell shielding boundary (walls, floors, and ceilings) will remain intact and functional. The engineered safety features requiring power will activate or go to their fail-safe configuration.

13.2.5.5 Unmitigated Likelihood

Loss of power can be initiated by off-site events or mechanical failures of equipment. Failures resulting in loss of power as initiating events are estimated to have an unmitigated likelihood of "not unlikely."

Additional detailed information describing a quantitative evaluation, including assumptions, methodology, uncertainties, and other data, will be developed for the Operating License Application.

13.2.5.6 Radiation Source Term

The loss of power evaluation is based on information developed for the Construction Permit Application. Detailed information describing radiation source terms for the loss of power event will be developed for the Operating License Application.

13.2.5.7 Evaluation of Potential Radiological Consequences

The loss of power evaluation is based on information developed for the Construction Permit Application. A detailed evaluation of potential radiological consequences, including a summary of radiological consequences from the analysis of other accidents where loss of power was an initiator, will be provided in the Operating License Application.

13.2.5.8 Identification of Items Relied on for Safety and Associated Functions

No additional IROFS have been identified specific to this event other than maintain operability of the IROFS listed in Section 13.2.5.3. The loss of normal electric power will not result in unsafe conditions for either workers or the public in uncontrolled areas.

Defensive-in-Depth

The following defense-in-depth feature, minimizing the impact of a loss of power event, was identified by the accident evaluations.

- A standby diesel generator will be available at the RPF.

13.2.6 Natural Phenomena Events

Chapter 2.0, "Site Characteristics," and Chapter 3.0 discuss the design of SSCs to withstand external events. The RPF is designed to withstand the effects of natural phenomena events. Consequences of natural phenomena accident sequences have been evaluated. Sections 13.2.6.1 through 13.2.6.6 provide event descriptions and identify any additional controls required to protect the health and safety of workers, the public, and environment.

13.2.6.1 Tornado Impact on Facility and Structures, Systems, and Components

The adverse impact of a tornado on facility operations has a number of facets that must be evaluated. This evaluation addresses the facility design as impacted by the maximum-sized tornado with a return frequency of 10^{-5}/year (yr).

- High winds can lead to significant damage to the facility structure. Damage to the structure is a function of the strength of the tornado winds, duration, debris carried by the winds, direction of impact, and facility design. This evaluation determines the impact of tornado winds on the facility from a design basis perspective to ensure that the design prevents impact to SSCs in the building.
- The local area impact may result in loss of utilities (e.g., electrical power) and reduced access by local emergency responders. Loss of power is evaluated (Section 13.2.5) as a potential cause for all adverse events. The individual PHA nodes evaluate the loss of site power and loss of power distribution to each subsystem.
- High winds may directly impact SSCs important to safety (e.g., components of the fire protection system are located in areas adjacent to the building) and reduce the reliability of those SSCs to respond to additional events (like loss of electrical power) that can be initiated concurrently with the tornado (either as an indirect result or as an additional random failure). This evaluation analyzes the impact of tornado winds on these SSCs.

Tornado impact on the facility structure – High wind pressures could cause a partial or complete collapse of the facility structure, which may cause damage to SSCs important to safety or impact the availability and reliability of those SSCs. A partial or complete structural collapse may also lead directly to a radiological or chemical release or a potential nuclear criticality, if damage caused by the collapse creates a violation of criticality spacing requirements. Tornado wind-driven missiles could penetrate the facility building envelope (walls and roof), impacting the availability and reliability of SSCs important to safety, or may lead directly to a radiological or chemical release.

Tornado impact on SSCs important to safety located outside the main facility – High wind pressures and tornado wind-driven missiles could damage SSCs important to safety located outside the RPF building envelope. The damage sustained may impact the availability and reliability of the SSCs important to safety. Loss of site power may affect the ability of SSCs important to safety located within the facility building envelope to respond to additional events.

A partial or complete collapse of the facility structure could also lead directly to an accident with adverse intermediate or high consequences. The only IROFS located outside the RPF building envelope is the exhaust stack. Buckling or toppling of the exhaust stack would affect its ability and availability to mitigate other events with intermediate consequences. The return frequency of the design basis tornado is 10^{-5}/yr, making the initiating event highly unlikely.

No additional IROFS are required.

13.2.6.2 High Straight-Line Winds Impact the Facility and Structures, Systems, and Components

Similar to the tornado, high straight-line winds can also damage the facility structure, which in turn can lead to damage to SSCs relied on for safety. This evaluation demonstrates how the facility design addressed straight-line winds with a return interval of 100 years or more, as required by building codes.

Buckling or toppling of the exhaust stack would affect the ability and availability to mitigate other events with intermediate consequences. A partial or complete collapse of the facility structure may also lead directly to an accident with adverse intermediate or high consequences.

The facility is designed as a Risk Category IV structure, a standard industrial facility with equivalent chemical hazards, in accordance with American Society of Civil Engineers (ASCE) 7, *Minimum Design Loads for Buildings and Other Structures*. The return frequency of the basic (design) wind speed for Risk Category IV structures is 5.88×10^{-4}/yr (mean return interval, MRI = 1,700 yr). At this return frequency, the straight-line wind event is not likely but credible during the design life of the facility and is considered in the structural design as a severe weather event. The provisions of the ASCE 7 standard, when used with companion standards such as American Concrete Institute (ACI) 318, *Building Code Requirements for Structural Concrete*, and American Institute of Steel Construction (AISC) 360, *Specification for Structural Steel Buildings*, are written to achieve the target maximum annual probabilities of established in ASCE 7. The highest maximum probability of failure, which is for a failure that is not sudden and does not lead to a wide-spread progression of collapse, targeted for Risk Category IV structures is 5.0×10^{-6}. Therefore, the likelihood of failure of the structure when subjected to the design basis straight-line wind in conjunction with other loads, as required by ASCE 7, is highly unlikely.

No additional IROFS are required.

13.2.6.3 Heavy Rain Impact on Facility and Structures, Systems, and Components

Localized heavy rain can overwhelm the structural integrity of the RPF roofing system. This evaluation determines the impact of probable maximum precipitation (PMP) in the form of rain on the roofing structure. The PMP is defined as "theoretical greatest depth of precipitation for a given duration that is physically possible over a particular drainage area at a certain time of year." In other words, the PMP represents the theoretical worst-case of the most precipitation the atmosphere is capable of discharging to a particular area over a selected period of time. The PMP is based on an empirical methodology with no defined annual exceedance probability.

For impact on the facility, the PMP for 25.9 square kilometers (km^2) (10 square miles [mi^2]) is evaluated. Large amounts of rain water accumulating on the roof could lead to collapse of the roof. A partial or complete collapse of the facility roof may impact the availability or reliability of SSCs relied on for safety located within the RPF building envelope to respond to other events of high consequence.

From the National Weather Service (NWS)/National Oceanic and Atmospheric Administration (NOAA) Hydrometeorological Report No. 51, *Probable Maximum Precipitation Estimates, United States East of the 105th Meridian*, the PMP is defined as "theoretical greatest depth of precipitation for a given duration that is physically possible over a particular drainage area at a certain time of year." In other words, the PMP represents the theoretical worst-case of the most precipitation the atmosphere is capable of discharging to a particular area over a selected period of time. The PMP is based on an empirical methodology with no defined annual exceedance probability. Although the NWS/NOAA has historically stated that it is not possible to assign an exceedance probability to the PMP (NOAA Technical Report NWS 25, *Comparison of Generalized Estimates of Probable Maximum Precipitation with Greatest Observed Rainfalls*), several academic studies and papers have undertaken the exercise to determine the annual exceedance probability for PMP using modern probabilistic techniques and storm modeling and have found that the exceedance probability varies by location but is quite low (NAP, 1994). As such, the PMP event has been determined to be highly unlikely.

No additional IROFS are required.

The roof of the RPF is designed to prevent rainwater from accumulating on the roof. In accordance with 2012 International Building Code (IBC) and ASCE 7, the roof structure is designed to safely support the weight of rainwater accumulation with the primary drainage system blocked and the secondary drainage system at its design flow rate when subjected to a rainfall intensity based on the 100-yr hourly rainfall rate. Deflections of roof members are limited to prevent rainwater ponding on the roof. The roof structure is also designed to support the extreme winter precipitation load discussed in Section 13.2.6.6.

13.2.6.4 Flooding Impact to the Facility and Structures, Systems, and Components

Regional flooding from large precipitation events raising the water levels of local streams and rivers to above the 500-yr flood level can have an adverse impact on the structure and SSCs within. These impacts include the structural damage from water and the damage to power supplies and instrument control systems for SSCs relied on for safety. The infiltration of flood water into the facility could cause the failure of moderation control requirements and lead to an accidental nuclear criticality. Direct damage or impairment of SSCs could also be caused by flooding in the facility.

The site will be graded to direct the stormwater from localized downpours with a rainfall intensity for the 100-yr storm for a 1-hr duration around and away from the RPF. Thus, no flooding from local downpours is expected based on standard industrial design. Rainwater that falls on the waste management truck ramp and accumulates in the trench drain has low to no consequence for radiological, chemical, and criticality hazards.

Situated on a ridge, the RPF will be located above the 500-yr flood plain according to the flood insurance rate map for Boone County, Missouri, Panel 295 (FEMA, 2011). The site is above the elevation of the nearest bodies of water (two small ponds and a lake), and no dams are located upstream on the local streams. This data conservatively provides a 2×10^{-3} year return frequency flood, which can be considered an unlikely event according to performance criteria. However, the site is located at an elevation of 248.4 m (815 ft), and the 500-year flood plain starts at an elevation of 242 m (795 ft), or 6.1 m (20 ft) below the site. Since the site is located only 6.1 m (20 ft) below the nearest high point on a ridge (relative to the local topography), is well above the beginning of the 500-yr flood plain, and is considered a dry site, the probable maximum flood from regional flooding is considered highly unlikely, without further evaluation.[3]

No additional IROFS are required.

13.2.6.5 Seismic Impact to the Facility and Structures, Systems, and Components

Beyond the impact on the facility structure and the potential for falling facility components impacting SSCs or direct damage to SSCs causing adverse events, other activities were identified as sensitive to seismic events. During the irradiated target shipping cask unloading preparations, the shield plug fasteners will be removed from an upright cask before mating the cask to the cask docking port. During the short period between that activity and installing the cask, a seismic event could dislodge the lift/cask combination and result in dislodging the shield plug in the presence of personnel. This event would result in potentially lethal doses to workers in a short period of time.

Seismic ground shaking can directly damage SSCs relied on for safety or lead to damage of the facility, including partial or complete collapse, which could impact SSCs relied on for safety inside and outside the RPF. Damage to the facility could also impact SSCs, causing radiological and chemical releases of intermediate consequence.

[3] The recommended standard for determining the probably maximum flood, ANS 2.8, *Determining Design Basis Flooding at Power Reactor Sites*, has been withdrawn.

Leaks of fissile solution, compromising the safe-geometry and safe interaction storage in solid material storage arrays and pencil tanks or vessels containing enriched uranium solutions, could lead to a criticality accident, a high consequence accident. NWMI-2015-SAFETY-004, Section 3.1, identifies IROFS to prevent and mitigate this accident scenario.

Dislodging the irradiated target shipping cask during unloading preparations could expose workers to a potentially lethal radiation dose. This event is considered a high consequence accident.

The safe-shutdown earthquake the RPF will be evaluated using Regulatory Guide 1.60, *Design Response Spectra for Seismic Design of Nuclear Power Plants*, for radioisotopes production facilities (e.g., 10 CFR 50). NWMI is currently using a spectrum anchored to 0.20 g peak ground acceleration for the RPF design basis. Regulatory Guide 1.60 is not indexed to any specific soil type, with its frequency content sufficiently broad to cover all soil types.

This peak ground acceleration matches that of MURR (Adams, 2016) and the Calloway Nuclear Generating Station, which both are within 80.5 km (50 mi) of the RPF, as suggested by the NRC staff during the November 10, 2016 public meeting. The analysis procedure develops ground motion acceleration time histories that match or exceed the Regulatory Guide 1.60 spectrum as input to the building finite element model. Structural damping will follow the recommendations of Regulatory Guide 1.61, *Damping Values for Seismic Design of Nuclear Power Plants*, which range from about 3 to 7 percent.

Response spectra corresponding to the recommended damping values of Regulatory Guide 1.61 will be used to derive seismic loads. Damping varies depending on the type of SSC. Structural damping will follow Regulatory Guide 1.61 guidance (ranging from about 3 to 7 percent). Plotting response spectra at 5 percent damping for purposes of illustration is a convention within the nuclear industry, but for analysis loads, damping will vary depending on the earthquake level (operating basis earthquake or safe-shutdown earthquake) and the type of SSC.

Using Regulatory Guide 1.60 and 1.61, the failure of the facility subject to the maximum-considered earthquake ground-shaking is considered highly unlikely.

No credit can be taken for physical features of the irradiated target cask lifting fixture for the unmitigated case; therefore, the unmitigated likelihood is equal to the annual probability of exceedance for the safe shutdown earthquake, $f_{earthquake} = 4 \times 10^{-4}$.

13.2.6.5.1 IROFS FS-04, Irradiated Target Cask Lifting Fixture

As a PEC, the irradiated target cask lifting fixture will be designed to prevent the cask from tipping within the fixture and prevent the fixture itself from toppling during a seismic event.

13.2.6.6 Heavy Snow Fall or Ice Buildup on Facility and Structures, Systems, and Components

This evaluation addresses snow loading on the facility structure. The facility protects the SSCs, and an extreme snow-loading event may cause failure of the roof, impacting the SSCs' ability to perform associated safety functions. NRC DC/COL ISG-07, *Interim Staff Guidance on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures*, provides guidance on the design of Category I structures for snow load that conservatively bounds the RPF. The normal snow load as defined in the NRC ISG is the 100-yr snowpack, which is equivalent to the design snow load for Risk Category IV structures determined in accordance with ASCE 7.

Collapse of the roof may damage SSCs that are relied on for safety, leading to accident sequences such as accidental nuclear criticality (e.g., a pencil tank was crushed and interaction controls violated) or a radiological release (e.g., if a hot cell confinement boundary was breached and a primary confinement boundary damaged), or may prevent an SSC from being available to perform its function.

The extreme winter precipitation load, as defined in the NRC ISG, is a combination of the 100-yr snowpack and the liquid weight of the probable maximum winter precipitation. The probable maximum winter precipitation is based on the seasonal variation of the PMP, given in NWS/NOAA Hydrometeorological Report 53, *Seasonal Variation of 10-Square Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian*, for winter months. The PMP is defined in Section 13.2.6.3. Considering the extreme winter precipitation load is a combination of the 100-yr snowpack and the theoretical worst-case precipitation event, the extreme winter precipitation load is highly unlikely.

The normal snow load is the 100-yr snowpack, which is equivalent to the design snow load for a Risk Category IV structure determined in accordance with ASCE 7. The return frequency of the normal snow load is relatively high and expected to be likely to occur during the design life of the facility. The provisions of the ASCE 7 standard, when used with companion standards such as ACI 318 and AISC 360, are written to achieve the target maximum annual probabilities of failure established in ASCE 7. The highest probability of failure, which is for a failure that is not sudden and does not lead to a wide-spread progression of collapse, targeted for Risk Category IV structures is 5.0×10^{-6}. Therefore, the likelihood of failure of the structure when subjected to the normal design snow load in conjunction with other loads as required by ASCE 7 is highly unlikely.

No additional IROFS are required.

13.2.7 Other Accidents Analyzed

A total of 75 accident sequences identified for further evaluation by the PHA were analyzed for the Construction Permit Application. A summary of all accidents analyzed is provided in Table 13-24. This table includes the accidents evaluated in Section 13.2.2 to 13.2.6 for completeness. Table 13-24 lists each accident sequence number, a descriptive title of the accident, and IROFS identified (if needed) to prevent or mitigate the consequences of the accident sequence.

The preliminary IROFS for each sequence are listed in the far right column of Table 13-24. The IROFS number and title are provided. If the accident sequence is bounded by the accidents discussed in Section 13.2.2 to 13.2.6, a pointer to the bounding accident sequence is listed. After further analysis, if the IROFS level controls were determined to not be required either due to reduced consequences or reduced frequency, this is stated. Other accident sequences have IROFS identified, and a pointer is included to the section where the control is discussed in more detail.

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.R.01	High-dose solution or enriched uranium solution spill causing a radiological exposure hazard	• IROFS RS-01, Hot Cell Liquid Confinement Boundary • IROFS RS-03, Hot Cell Secondary Confinement Boundary • IROFS RS-04, Hot Cell Shielding Boundary • IROFS CS-07, Pencil Tank and Vessel Spacing Control using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-08, Floor and Sum Geometry Control on Slab Depth, Sump Diameter or Depth for Floor Spill Containment Berms • IROFS CS-09, Double-Wall Piping • See Section 13.2.2.8
S.R.02	Spray release of solutions spilled from primary offgas treatment solutions, resulting in radiological consequences	• Bounded by S.R.01
S.R.03	Spray release of high-dose or enriched uranium-containing product solution, resulting in radiological consequences	• Bounded by S.R.01
S.R.04	Liquid enters process vessel ventilation system damaging IRU or retention beds, releasing retained radionuclides	• IROFS RS-09, Primary Offgas Relief System • IROFS RS-03, Hot Cell Secondary Confinement Boundary • See Section 13.2.3.8
S.R.05	High-dose solution enters the UN blending and storage tank	• Not credible or low consequence
S.R.06	High flow through IRU causing premature release of high-dose iodine gas	• Bounded by S.R.04
S.R.07	Loss of temperature control on the IRU leading to release of high-dose iodine	• Bounded by S.R.04
S.R.08	Loss of vacuum pumps	• Bounded by S.R.04
S.R.09	Loss of IRU or carbon bed media to downstream part of the system	• Bounded by S.R.04
S.R.10	Wrong retention media added to bed or saturated retention media	• Event unlikely with intermediate consequence

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.R.12	Mo product cask removed from the hot cell boundary with improper shield plug installation	• Event unlikely with intermediate consequence
S.R.13	High-dose containing solution leaks to chilled water or steam condensate system	• IROFS RS-04, Hot Cell Shielding Boundary • IROFS CS-06, Pencil Tank and Vessel Spacing Control using the Diameter of the Tanks, Vessels, or Piping • IROFS CS-10, Closed Safe-Geometry Heating or Cooling Loop with Monitoring and Alarm • IROFS CS-27, Closed Heating or Cooling Loop with Monitoring and Alarm • IROFS CS-20, Evaporator or Concentrator Condensate Monitoring • IROFS CS-18, Backflow Prevention Device • IROFS CS-19, Safe-Geometry Day Tanks • See Section 13.2.4.8
S.R.14	IX resin failure due to wrong reagent or high temperature	• Bounded by S.R.01
S.R.16	Backflow of high-dose radiological and/or fissile solution into auxiliary system (purge air, chemical addition line, water addition line, etc.)	• Bounded by S.R.13
S.R.17	Carryover of high-dose solution into condensate (a low-dose waste stream)	• IROFS RS-08, Sample and Analysis of Low Dose Waste Tank Dose Rate Prior to Transfer Outside the Hot Cell Shielded Boundary • IROFS RS-10, Active Radiation Monitoring and Isolation of Low-Dose Waste Transfer • See Section 13.2.7.1
S.R.18	High-dose solution flows into the solidification media hopper	• Low consequence event that does not challenge IROFS RS-04
S.R.19	High target basket retrieval dose rate	• Design evolved after PHA, accident sequence eliminated
S.R.20	Radiological spill of irradiated LEU target material in the hot cell area	• Bounded by S.R.01
S.R.21	Damage to the hot cell wall providing shielding	• Low consequence event that does not damage shielding function of IROFS RS-04
S.R.22	Decay heat buildup in unprocessed LEU target material removed from targets leads to higher-dose radionuclide offgassing	• Low consequence event

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.R.23	Offgassing from irradiated target dissolution tank occurs when the upper valve is opened	• IROFS RS-03, Hot Cell Secondary Confinement Boundary • See Section 13.2.2.8
S.R.24	Bagless transport door failure	• IROFS RS-03, Hot Cell Secondary Confinement Boundary • IROFS RS-04, Hot Cell Shielding Boundary • See Section 13.2.2.8
S.R.25	HEPA filter failure	• IROFS RS-03, Hot Cell Secondary Confinement Boundary • See Section 13.2.2.8
S.R.26	Failed negative air balance from zone-to-zone or failure to exhaust a radionuclide buildup in an area	• IROFS RS-03, Hot Cell Secondary Confinement Boundary • See Section 13.2.2.8
S.R.27	Extended outage of heat leading to freezing, pipe failure, and release of radionuclides from liquid process systems	• Highly unlikely event for process solutions containing fission products • Bounded by S.C.04 for target fabrication systems
S.R.28	Target or waste shipping cask or container not loaded or secured according to procedure, leading to personnel exposure	• Information will be provided in the Operating License Application
S.R.29	High dose to worker from release of gaseous radionuclides during cask receipt inspection and preparation for target basket removal	• IROFS RS-12, Cask Containment Sampling Prior to Closure Lid Removal • IROFS RS-13, Cask Local Ventilation During Closure Lid Removal and Docking Preparations • See Section 13.2.7.1
S.R.30	Cask docking port failures lead to high-dose to worker due to streaming radiation and/or high airborne radioactivity	• IROFS RS-04, Hot Cell Shielding Boundary • IROFS RS-15, Cask Docking Port Enabling Sensor • See Sections 13.2.2.8 and 13.2.7.1
S.R.31	Chemical burns from contaminated solutions during sample analysis	• Judged unlikely event with intermediate consequence
S.R.32	Crane load drop accidents	• IROFS FS-01, Enhanced Lift Procedure • IROFS FS-02, Overhead Cranes • See Section 13.2.7.1

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.C.01	Failure of facility enrichment limit	• Judged highly unlikely based on supplier's checks and balances
S.C.02	Failure of administrative control on mass (batch limit) during handling of fresh U, scrap U, LEU target material, targets, and samples	• IROFS CS-02, Mass and Batch Handling Limits for Uranium Metal, [Proprietary Information], Targets, and Laboratory Sample Outside Process Systems • IROFS CS-03, Interaction Control Spacing Provided by Administrative Control • IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • See Section 13.2.7.2
S.C.03	Failure of interaction limit during handling of fresh U, scrap U, LEU target material, targets, containers, and samples	• IROFS CS-02, Mass and Batch Handling Limits for Uranium Metal, [Proprietary Information], Targets, and Laboratory Sample Outside Process Systems • IROFS CS-03, Interaction Control Spacing Provided by Administrative Control • IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • See Section 13.2.7.2
S.C.04	Spill of process solution from a tank or process vessel leading to accidental criticality	• IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-08, Floor and Sump Geometry Control of Slab Depth, Sump Diameter or Depth for Floor Spill Containment Berms • IROFS CS-09, Double-Wall Piping • IROFS CS-26, Processing Component Safe Volume Confinement • See Section 13.2.7.2
S.C.05	Leak of fissile solution into the heating or cooling jacket on the tank or vessel	• Bounded by S.R.13
S.C.06	System overflow to process ventilation involving fissile material	• IROFS CS-11, Simple Overflow to Normally Empty Safe Geometry Tank with Level Alarm • IROFS CS-12, Condensing Pot or Seal Pot in Ventilation Vent Line • IROFS CS-13, Simple Overflow to Normally Empty Safe Geometry Floor with Level Alarm in the Hot Cell Containment Boundary • See Section 13.2.7.2

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.C.07	Fissile solution leaks across mechanical boundary between process vessels and heating/cooling jackets into heating/cooling media	• Bounded by S.R.13
S.C.08	Backflow of high-dose radiological and/or fissile solution into auxiliary system (purge air, chemical addition line, water addition line, etc.)	• Bounded by S.R.13
S.C.09	High concentrations of uranium enter the concentrator or evaporator condensates	• IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-26, Processing Component Safe Volume Confinement • See Section 13.2.7.2
S.C.10	High concentrations of uranium enter the low-dose or high-dose waste collection tanks	• IROFS CS-14, Active Discharge Monitoring and Isolation • IROFS CS-15, Independent Active Discharge Monitoring and Isolation • IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.11	High concentrations of uranium in contactor solvent regeneration aqueous waste	• Bounded by S.C.04 and S.C.10
S.C.12	High concentrations of uranium in the LEU target material wash solution	• IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • See Section 13.2.7.2

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.C.13	High concentrations of uranium in the nitrous oxide scrubber	• IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.14	High concentrations of uranium in the IX waste collection tanks effluent	• IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.15	High concentrations of uranium in the IX resin waste	• IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.17	High concentrations of uranium in the solid waste encapsulation process	• IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • IROFS CS-21, Visual Inspection of Accessible Surfaces for Foreign Debris • IROFS CS-22, Gram Estimator Survey of Accessible Surfaces for Gamma Activity • IROFS CS-23, Nondestructive Assay of Items with Inaccessible Surfaces • IROFS CS-24, Independent Nondestructive Assay of Items with Inaccessible Surfaces • IROFS CS-25, Target Housing Weighing Prior to Disposal • See Section 13.2.7.2
S.C.19	Failure of PEC – Component safe geometry dimension or safe volume	• IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-26, Processing Component Safe Volume Confinement • See Section 13.2.7.2
S.C.20	Failure of concentration limits	• No credible path leading to criticality identified or not credible by design

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.C.21	Target basket passive design control failure on fixed interaction spacing	• IROFS CS-02, Mass and Batch Handling Limits for Uranium Metal, [Proprietary Information], Targets, and Laboratory Sample Outside Process Systems • IROFS CS-03, Interaction Control Spacing Provided by Administrative Control • See Section 13.2.7.2
S.C.22	High concentration of uranium in the TCE evaporator residue	• IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement Using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.23	High concentration in the spent silicone oil waste	• IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • IROFS CS-05, Container Batch Volume Limit • IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement Using the Diameter of Tanks, Vessels, or Piping • IROFS CS-07, Pencil Tank and Vessel Spacing Control Using Fixed Interaction Spacing of Individual Tanks or Vessels • IROFS CS-16, Sampling and Analysis of Uranium Mass or Concentration Prior to Discharge or Disposal • IROFS CS-17, Independent Sampling and Analysis of Uranium Concentration Prior to Discharge or Disposal • See Section 13.2.7.2
S.C.24	High uranium content on HEPA filters and subsequent failure	• Bounded by S.C.17
S.C.27	Failure of administratively controlled container volume limits	• IROFS CS-03, Interaction Control Spacing Provided by Administrative Control • IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement • IROFS CS-05, Container Batch Volume Limit • See Section 13.2.7.2
S.C.28	Crane load drop accidents	• IROFS FS-01, Enhanced Lift Procedure • IROFS FS-02, Overhead Cranes • See Section 13.2.7.2
S.F.01	Pyrophoric fire in uranium metal	• Event highly unlikely based on credible physical conditions
S.F.02	Accumulation and ignition of flammable gas in tanks or systems	• IROFS FS-03, Process Vessel Emergency Purge System • See Section 13.2.7.3

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.F.03	Hydrogen detonation in reduction furnace	• Judged highly unlikely based on credible physical conditions
S.F.04	Fire in reduction furnace	• Judged unlikely based on event frequency
S.F.05	Fire in a carbon retention bed	• IROFS FS-05, Exhaust Stack Height • See Section 13.2.7.3
S.F.06	Accumulation of flammable gas in ventilation system components	• Bounded by S.F.02
S.F.07	Fire in nitrate extraction system - combustible solvent with uranium	• Event unlikely with intermediate or low consequences
S.F.08	General facility fire	• Information will be provided in the Operating License Application
S.F.09	Hydrogen explosion in the facility due to a leak from the hydrogen storage or distribution system	• Information will be provided in the Operating License Application
S.F.10	Combustible fire occurs in hot cell area	• Information will be provided in the Operating License Application
S.F.11	Detonation or deflagration of natural gas leak in steam generator room	• Information will be provided in the Operating License Application
S.N.01	Tornado impact on facility and SSCs important to safety	• Judged highly unlikely event based on return frequency
S.N.02	High straight-line winds impact the facility and SSCs important to safety	• Judged highly unlikely to result in structure failure
S.N.03	Heavy rain impact on facility and SSCs important to safety	• Bounded by S.N.06
S.N.04	Flooding impact to the facility and SSCs important to safety	• Judged highly unlikely event based on facility location above the 500-year flood plain
S.N.05	Seismic impact to the facility and SSCs important to safety	• Judged highly unlikely to result in structure failure • IROFS FS-04, Irradiated Target Cask Lifting Fixture • See Section 13.2.6.5
S.N.06	Heavy snowfall or ice buildup on facility and SSCs important to safety	• Judged highly unlikely to result in structure failure
S.M.01	Vehicle strikes SSC important to safety and causes damage or leads to an accident sequence of intermediate or high consequence	• Judged likely event with low consequence

Table 13-24. Analyzed Accidents Sequences (9 pages)

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
S.M.02	Facility evacuation impacts on operations	• Judged likely event with low consequence
S.M.03	Localized flooding due to internal system leakage or fire suppression sprinkler activation	• IROFS CS-08, Floor and Sump Geometry Control of Slab Depth, Sump Diameter or Depth for Floor Spill Containment Berms • See Section 13.2.7.2
S.CS.01	Nitric acid fume release	• No IROFS currently identified

HEPA = high-efficiency particulate air.
IROFS = items relied on for safety.
IRU = iodine removal unit.
IX = ion exchange.
LEU = low-enriched uranium.
Mo = molybdenum.
PEC = passive engineered control.
PHA = preliminary hazards analysis.
SSC = structures, systems, and components.
TCE = trichloroethylene
U = uranium.
UN = uranyl nitrate.

Table 13-25 provides a summary of all IROFS identified by the accident analyses performed for the Construction Permit Application. Table 13-25 also identifies whether the IROFS were considered engineered safety features or administrative controls. Engineered safety features are described in Chapter 6.0, and the administrative controls are discussed in Chapter 14.0, "Technical Specifications." Additional IROFS are anticipated to be identified (or the current IROFS modified) by additional design detail developed for the Operating License Application.

Table 13-25. Summary of Items Relied on for Safety Identified by Accident Analyses (3 pages)

IROFS designator	Descriptor	Engineered safety feature	Administrative control
RS-01	Hot cell liquid confinement boundary	✓	
RS-02	Reserved		
RS-03	Hot cell secondary confinement boundary	✓	
RS-04	Hot cell shielding boundary	✓	
RS-05	Reserved		
RS-06	Reserved		
RS-07	Reserved		
RS-08	Sample and analysis of low-dose waste tank dose rate prior to transfer outside the hot cell shielded boundary		✓
RS-09	Primary offgas relief system	✓	
RS-10	Active radiation monitoring and isolation of low-dose waste transfer	✓	
RS-11	Reserved		
RS-12	Cask containment sampling prior to closure lid removal		✓
RS-13	Cask local ventilation during closure lid removal and docking preparations	✓	
RS-14	Reserved		

Table 13-25. Summary of Items Relied on for Safety Identified by Accident Analyses (3 pages)

IROFS designator	Descriptor	Engineered safety feature	Administrative control
RS-15	Cask docking port enabling sensor	✓	
CS-01	Reserved		
CS-02	Mass and batch handling limits for uranium metal, [Proprietary Information], targets, and laboratory sample outside process systems		✓
CS-03	Interaction control spacing provided by administrative control		✓
CS-04	Interaction control spacing provided by passively designed fixtures and workstation placement	✓	
CS-05	Container batch volume limit		✓
CS-06	Pencil tank, vessel, or piping safe geometry confinement using the diameter of tanks, vessels, or piping	✓	
CS-07	Pencil tank and vessel spacing control using fixed interaction spacing of individual tanks or vessels	✓	
CS-08	Floor and sump geometry control of slab depth, sump diameter or depth for floor spill containment berms	✓	
CS-09	Double-wall piping	✓	
CS-10	Closed safe geometry heating or cooling loop with monitoring and alarm	✓	
CS-11	Simple overflow to normally empty safe geometry tank with level alarm	✓	
CS-12	Condensing pot or seal pot in ventilation vent line	✓	
CS-13	Simple overflow to normally empty safe geometry floor with level alarm in the hot cell containment boundary	✓	
CS-14	Active discharge monitoring and isolation	✓	
CS-15	Independent active discharge monitoring and isolation	✓	
CS-16	Sampling and analysis of uranium mass or concentration prior to discharge or disposal		✓
CS-17	Independent sampling and analysis of uranium concentration prior to discharge or disposal		✓
CS-18	Backflow prevention device	✓	
CS-19	Safe-geometry day tanks	✓	
CS-20	Evaporator or concentrator condensate monitoring	✓	
CS-21	Visual inspection of accessible surfaces for foreign debris		✓
CS-22	Gram estimator survey of accessible surfaces for gamma activity		✓
CS-23	Nondestructive assay of items with inaccessible surfaces		✓
CS-24	Independent nondestructive assay of items with inaccessible surfaces		✓
CS-25	Target housing weighing prior to disposal		✓
CS-26	Processing component safe volume confinement	✓	
CS-27	Closed heating or cooling loop with monitoring and alarm	✓	

Table 13-25. Summary of Items Relied on for Safety Identified by Accident Analyses (3 pages)

IROFS designator	Descriptor	Engineered safety feature	Administrative control
FS-01	Enhanced lift procedure		✓
FS-02	Overhead cranes		✓
FS-03	Process vessel emergency purge system	✓	
FS-04	Irradiated target cask lifting fixture	✓	
FS-05	Exhaust stack height	✓	

IROFS = items relied on for safety.

The following subsections describe the IROFS that are not previously discussed elsewhere in this chapter. The IROFS are grouped according to their respective accident sequence categories, as shown in Table 13-26.

Table 13-26. Accident Sequence Category Definitions

Accident sequence category	Definition	Section containing related IROFS description
S.R.	Radiological	13.2.7.1
S.C.	Criticality	13.2.7.2
S.F.	Fire or explosion	13.2.7.3
S.N.	Natural phenomena	13.2.7.4
S.M.	Man-made	13.2.7.5
S.CS.	Chemical safety	13.2.7.6

IROFS = items relied on for safety.

13.2.7.1 Items Relied on for Safety for Radiological Accident Sequences (S.R.)

The following IROFS fall under the radiological accident sequence category and are not discussed elsewhere in this chapter.

13.2.7.1.1 IROFS RS-08, Sample and Analysis of Low Dose Waste Tank Dose Rate Prior to Transfer Outside the Hot Cell Shielded Boundary

As an augmented administrative control (AAC), prior to transferring the solution from the low-dose waste tank to the low-dose waste encapsulation system outside of the hot cell shielded boundary, the low-dose waste tank will be administratively locked out, sampled, and the sample analyzed for high radiation. Batches that satisfy the sample criteria can be transferred to the low-dose waste encapsulation system. The safety function of this AAC is to prevent transfer of low-dose solution to outside the shielded boundary at radiation dose rates that would lead to intermediate- or high-dose consequences to workers.

13.2.7.1.2 IROFS RS-10, Active Radiation Monitoring and Isolation of Low Dose Waste Transfer

As an AEC, the recirculating stream and discharge stream of the low-dose waste tank will be simultaneously monitored in a background shielded trunk outside of the hot cell shielded cavity. The continuous gamma-ray instrument monitoring the recirculation line and the transfer line will provide an open permissive signal to a dedicated isolation valve in the transfer line. The safety function of the system is to prevent transfer of low-dose waste solutions with exposure rates in excess of approved limits (safety limits and limiting safety system settings to be determined later) to outside the shielded boundary at radiation dose rates that would lead to intermediate- or high-dose consequences to workers or the public.

The system functions by monitoring both the recirculation line for the low-dose waste collection tank and the transfer line to the low-dose waste encapsulation system outside of the hot cell shielded boundary. Monitoring will be performed in a shielded trunk, which reduces the background from the normally shielded hot cell areas to acceptable levels for monitoring. In this closed-loop system, the gamma monitor will provide an open permissive signal to a fail-closed isolation valve in the transfer line, allowing the isolation valve to open.

If the radiation levels exceed a safety limit setpoint during recirculation for sampling or during transfers, the isolation valve will be closed. The isolation valve will also fail closed on loss of power and loss of instrument air.

13.2.7.1.3 IROFS RS-12, Cask Containment Sampling Prior to Closure Lid Removal

As an AEC, a sampling system will be connected to the cask vent to sample the atmosphere within the cask prior to closure lid removal. The system will sample the contents of the cask and have the ability to remediate the atmosphere using a vacuum system if dose rates are too high (safety limits to be determined). The safety function of IROFS RS-12 is to prevent personnel exposure to high-dose gaseous radionuclides.

The system will identify a hazardous concentration of high-dose gases in the cask, and if a high dose is identified, will remediate the situation through evacuation to a safe processing system. The system works by evacuating a sample of the gas and analyzing the sample as it passes by a detector. If high activity is detected, the system will alarm. The operator will use the system to evacuate and backfill the cask with fresh air (from a protected pressurized source such as a compressed bottle) until the atmospheres are within approved safety limits.

13.2.7.1.4 IROFS RS-13, Cask Local Ventilation During Closure Lid Removal and Docking Preparations

As an AEC, a local capture ventilation system will be used over the closure lid to remove any escaped gases from the breathing zone of the worker during removal of the closure lid, removal of the shielding block bolts, and installation of the lifting lugs. The safety function of IROFS RS-13 is to prevent exposure to the worker by evacuating any high-dose gaseous radionuclides from the worker's breathing zone and preventing immersion of the worker in a high-dose environment. The system will use a dedicated evacuation hood over the top of the cask during containment closure lid removal. The gases will be removed to the Zone 1 secondary containment system for processing.

13.2.7.1.5 IROFS RS-15, Cask Docking Port Enabling Sensor

As an AEC, the cask docking port will be equipped with sensors that detect when a cask is mated with the cask docking port door. The sensors feed an enabling circuit that will prevent the door from being opened when no cask is present. The safety function of IROFS RS-15 is to prevent the cask docking port door from being opened, allowing a streaming radiation path to an accessible area and to prevent Zone II to Zone I air pressure imbalances that would allow air to migrate into the Zone II airlock. The system will also prevent a high streaming dose to workers from targets inside the hot cell, if the cask lift fails following mating. The system is designed to provide an enabling contact signal and positive closure signal when the sensor does not sense a cask mated to the door, causing the door to close.

13.2.7.1.6 IROFS FS-01, Enhanced Lift Procedure

As an Administrative Control (AC), lifts of high-dose rate containers or casks or of heavy objects (weight limit to be determined in final design) that move over hot cells in the standby or operating modes will use an enhanced lift procedure to reduce the likelihood of an upset. Enhancements will use the guidelines in DOE-STD-1090-2011, *Hoisting and Rigging*, for critical lifts (for nonroutine cover block lifts) and pre-engineered production lifts (for routine container and cask lifts using pre-engineered fixtures). The safety function of IROFS FS-01 is to prevent (by reducing the likelihood) a dropped load or striking an SSC with a heavy load, causing damage that leads to an intermediate or high consequence event. The IROFS will be administered through the use of operating and maintenance procedures.

13.2.7.2 Items Relied on for Safety for Criticality Accident Sequences (S.C.)

The following IROFS fall under the criticality accident sequence category and are not discussed elsewhere in this chapter.

13.2.7.2.1 IROFS CS-02, Mass and Batch Handling Limits for Uranium Metal, [Proprietary Information], Targets, and Laboratory Samples Outside Process Systems

As a simple AC, mass and batch limits will be applied to handling, processing, and storage activities where uranium metal, [Proprietary Information] (LEU target material), targets, and/or samples are used. The mass or batch limits will be set such that the handled quantity can sustain double-batching or one interaction control failure with another approved quantity of fissile material, approved volume of fissile material, or an approved configuration for a tank, vessel, or IX column.

Where safe batches are allowed, fixtures will be used to ensure that the safe batch is not exceeded (e.g., where [Proprietary Information] are allowed as a safe batch, the operator will be provided with a carrying fixture that allows only [Proprietary Information]). For targets, the housing is credited for maintaining the contents dry. Final limits for each activity will be set in final design.

13.2.7.2.2 IROFS CS-03, Interaction Control Spacing Provided by Administrative Control

As a simple AC, while handling approved quantities of uranium metal, approved quantities of [Proprietary Information] (LEU target material), batches of targets, or batches of samples, an interaction control will be maintained between quantities being handled; fissile solution tanks, vessels, or IX columns; and safe-geometry ventilation housings. Interaction control spacing will be set in final design when all process upsets are evaluated.

13.2.7.2.3 IROFS CS-04, Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement

As a PEC, fixed interaction control fixtures or workstations will be provided for holding or processing approved containers with designated quantities of uranium metal, quantities of [Proprietary Information] (LEU target material), batches of targets, and batches of samples. The fixtures are designed to hold only the approved container or batch and are fixed with 61 centimeter (cm) (2-ft) edge-to-edge spacing from all other fissile material containers, workstations, or fissile solution tanks, vessels, or IX columns. Where LEU target material is handled in open containers, the design should prevent spills from readily spreading to an adjacent workstation or storage location. Final workstation and fixture spacing will be determined in final design when all process upsets are evaluated. Workstations with interaction controls will include the following (not an all-inclusive listing):

- LEU target material trichloroethylene (TCE) wash column workstation containing a safe-geometry funnel
- LEU target material ammonium hydroxide rinse column workstations containing safe-geometry funnels
- Target basket fixture that provides safe spacing of a batch of targets from another batch in the target receipt cell

13.2.7.2.4 IROFS CS-05, Container Batch Volume Limit

As a simple AC to address the activity of sampling and small quantity storage, a volumetric batch limit will be applied such that the total number of small sample or storage containers is controlled to a safe total volume. Many activities at the RPF will involve very high-dose solutions; only small quantities of a sample may be removed from the shielded area for analysis due to radiological reasons. As a result, sample bottles will be relatively small. The uranium content in these containers will often be unknown. To provide safe storage and handling in the laboratory environment, a safe volumetric batch limit on these small containers will be applied.

Some potentially contaminated uranium waste streams will also be generated at the RPF that require quantification of the uranium content prior to disposal. These waste streams will need a safe volume container for interim storage while the uranium content is being identified. The final set of approved containers and volumes will be provided during final design when all process upsets are evaluated.

13.2.7.2.5 IROFS CS-11, Simple Overflow to Normally Empty Safe Geometry Tank with Level Alarm

As a PEC, for each vented tank containing fissile or potentially fissile process solution for which IROFS CS-11 is assigned, a simple overflow line will be installed below the level of the process vessel ventilation port and any chemical addition ports (where an anti-siphon safety feature will be installed). The overflow drain will prevent the process solution from entering the respective non-geometrically favorable portions of the process ventilation system and any chemical addition ports (where the solutions will enter through anti-siphon devices). The safety function of this feature is to prevent accidental nuclear criticality in non-geometrically favorable portions of the process ventilation system. The overflow will be directed to a safe-geometry storage tank, which will normally be empty. The overflow storage tank will be equipped with a level alarm to inform the operator when use of the IROFS has been initiated so that actions may be taken to restore operability of the safety feature by emptying the tank. The locations where this IROFS is used will be determined during final design.

13.2.7.2.6 IROFS CS-12, Condensing Pot or Seal Pot in Ventilation Vent Line

As a PEC, downstream of each tank for which IROFS CS-12 is assigned, a safe geometry condensing pot or seal pot will be installed to capture and redirect liquids to a safe-geometry tank or flooring area with safe-geometry sumps. One such condensing or seal pot may service several related tanks within the safe-geometry boundary of the ventilation system. The condensing or seal pot will prevent fissile solution from flowing into the respective non-geometrically favorable process ventilation system by directing the solution to a safe-geometry tank or flooring area with safe-geometry sumps.

The safety function of IROFS CS-12 is to prevent accidental nuclear criticality in non-geometrically favorable portions of the process ventilation system. The safe-geometry tank or sumps will be equipped with a level alarm to inform the operator when use of the IROFS has been initiated. Each individual tank or vessel operation must be evaluated for required capacity for overflow to ensure that a suitable overflow volume is available.

A monitoring and alarm circuit will be provided so that common overflow tanks or safe slab flooring or sumps may be used for multiple tanks or vessels, and limiting conditions of operation will be defined to ensure that the IROFS is made available in a timely manner or operations are suspended following an overflow event of a single tank. Where independent seal or condensing pots are credited, the drains of the seal or condensing pots must be directed to independent locations to prevent a common clog or overcapacity condition from defeating both.

13.2.7.2.7 IROFS CS-13, Simple Overflow to Normally Empty Safe Geometry Floor with Level Alarm in the Hot Cell Containment Boundary

As a PEC for each vented tank containing fissile or potentially fissile process solution for which IROFS CS-13 is assigned, a simple overflow line will be installed above the high alarm setpoint. The overflow will be directed to one or more safe-geometry flooring configurations with safe-geometry sumps. IROFS CS-13 will prevent accidental criticality by ensuring that overflowing fissile solutions are captured in a safe-geometry slab configuration with safe-geometry sumps. These flooring areas (separated as needed to support operations in different hot cell areas) will normally be empty. The flooring areas will be equipped with a sump level alarm to inform the operator when use of the IROFS has been initiated.

13.2.7.2.8 IROFS CS-14, Active Discharge Monitoring and Isolation

As an AEC for discharges from safe-geometry systems to non-favorable geometry systems, an active uranium detection system will be used to close an isolation valve in the discharge line at a uranium concentration limit and/or cumulative mass limit (the limit[s] to be set sufficiently low to preclude follow-on process upsets and sufficiently high to maintain an operating limit setpoint below the safety setpoint). This system will prevent a high-concentration uranium solution from being discharged to a non-favorable geometry system.

The safety function of IROFS CS-14 is to prevent an accidental nuclear criticality. The closed-loop system is designed to isolate the discharge points listed below by actively monitoring the solution stream for uranium concentration using a suitable uranium monitor. At a limiting setpoint, the uranium monitor will close an isolation valve in the discharge line to stop the discharge. The uranium monitor is designed to produce a valve-open permissive signal that fails to an open state, closing the valve on loss of electrical power. The isolation valve is designed to fail-closed on loss of instrument air, and the solenoid is designed to fail-closed on loss of signal. The locations where this IROFS is used will be determined during final design.

13.2.7.2.9 IROFS CS-15, Independent Active Discharge Monitoring and Isolation

As an AEC for discharges from safe-geometry systems to non-favorable geometry systems, an independent active uranium detection system will be used to close an independent isolation valve in the discharge line at a uranium concentration limit and/or cumulative mass limit (the limit[s] to be set sufficiently low to preclude follow-on process upsets and sufficiently high to maintain an operating limit setpoint below the safety setpoint). This system will prevent a high concentration uranium solution from being discharged to a non-favorable geometry system.

The safety function of IROFS CS-15 is to prevent an accidental nuclear criticality. The closed-loop system is designed to isolate the discharge points listed below by actively monitoring the solution stream for uranium concentration using a suitable monitor to detect uranium. At a limiting setpoint, the monitor will close an isolation valve in the discharge line to stop the discharge. The monitor is designed using a different monitoring method and isolation valve than used in IROFS CS-14 to produce a valve-open permissive signal that fails to an open state, closing the valve on loss of electrical power.

The isolation valve is designed to fail-closed on loss of instrument air, and the solenoid is designed to fail-closed on loss of signal. The locations where this IROFS is used will be determined during final design.

13.2.7.2.10 IROFS CS-16, Sampling and Analysis of U Mass/Concentration Prior to Discharge/Disposal

As an AAC, prior to initiating discharge from the safe-geometry container, tanks, or vessels assigned IROFS CS-16 to non-favorable geometry systems, the container, tank, or vessel will be isolated and placed under administrative control, recirculated or otherwise uniformly mixed, sampled, and the sample analyzed for uranium content. The discharge or disposal will only be approved following independent review of the sample results to confirm that the uranium content is below a concentration or a mass limit (to be determined for each individual application based on expected volumes and follow-on processing needs) and under the independent oversight of a supervisor (who administratively controls the locks on the discharge system). Uranium mass in the disposal container or vessel will be tracked to ensure that the mass or concentration limit for the container is not exceeded.

The safety function of IROFS CS-16 is to prevent accidental nuclear criticality caused by discharging or disposing of high-concentration uranium to an uncontrolled system. The IROFS functions as described by ensuring, through physical sampling and analysis, that the uranium content of an isolated container, tank, or vessel (both inlets and outlets isolated, as applicable) is below a safe, single parameter limit on solution concentration or under a safe mass for the disposal container. Systems, tanks, or vessels for which IROFS CS-16 applies, include:

- TCE recycle tanks
- Spent silicone oil
- Condensate tanks (either as normal or backup controls)

13.2.7.2.11 IROFS CS-17, Independent Sampling and Analysis of U Concentration Prior to Discharge/Disposal

As an AAC, prior to initiating discharge from the safe-geometry tanks or vessels assigned IROFS CS-17 to non-favorable geometry systems, the tank or vessel will be isolated and placed under administrative control, recirculated, sampled, and the sample analyzed for uranium content. The recirculation or uniformly mixing, sampling, and analysis activities will be independent (performed at a different time, using different operators or laboratory technicians, and different analysis equipment, checked with independent standards) of that performed in IROFS CS-16.

The discharge or disposal will only be approved following independent review of the sample results to confirm the uranium content is below the limiting setpoint for uranium concentration or batch mass for the contents and under the independent oversight of a supervisor (who administratively controls the locks on the discharge system). Uranium mass in the disposal container or vessel will be tracked and independently verified to ensure that the mass or concentration limit for the container is not exceeded.

The safety function of IROFS CS-17 is to prevent accidental nuclear criticality caused by discharging high-concentration uranium to an uncontrolled system. The IROFS functions as described by ensuring, through physical sampling and analysis, that the uranium content of an isolated tank or vessel is below a safe, single parameter limit on solution concentration or mass for a disposal container. Systems, tanks, or vessels for which IROFS CS-17 applies include:

- TCE recycle tanks
- Spent silicone oil
- Condensate tanks (either as normal or backup controls)

13.2.7.2.12 IROFS CS-21, Visual Inspection of Accessible Surfaces for Foreign Debris

As a simple AC, a visual inspection will be performed to identify foreign matter on accessible surfaces of equipment and waste materials approved for this method prior to disposal. All visible foreign material is assumed to be uranium. All surfaces must be non-porous. Materials involved must be solids (no solutions or liquids present). All surfaces must be visually accessible either directly or through approved inspection devices. The inspection criterion is for no foreign material of discernible thickness to be visible (transparent films allowed). The safety function of this AC is to ensure that no significant uranium deposits exist on the item being disposed, to prevent an accumulation of a minimum subcritical mass of uranium in the disposal container. The control will be exercised at designated waste consolidation stations, holding specifically approved waste containers, and on the items approved by the Criticality Safety Manager. The waste will not be consolidated until independent measurements conducted according to IROFS CS-22 or IROFS CS-24 have been completed. The item will be controlled during the waste measurement analysis period. Items initially approved include disassembled irradiated or scrap target housing parts or pieces.

13.2.7.2.13 IROFS CS-22, Gram Estimator Survey of Accessible Surfaces for Gamma Activity

As an AAC, a gram estimator survey will be performed on all accessible surfaces of equipment and waste materials approved for this method prior to disposal. The survey will be performed on low-risk waste streams that have surfaces that are 100 percent accessible with the measurement instrument. The measurement setpoint is designed to detect activity from 15 g of ^{235}U uniformly spread over 30 kilograms (kg) of 4-mil (thousandth of an inch) thick polyethylene sheeting (both sides) as a bounding waste form for disposal at the U.S. Department of Transportation (DOT) fissile-excepted limit of 0.5 g ^{235}U/L kg non-fissile material.

The purpose of this IROFS is to provide a backup instrument AAC to visual inspection (IROFS CS-21) for bulking and disposal of low-risk waste to prevent accidental nuclear criticality. All surfaces will need to be accessible to the instrument used. The waste stream must not be contaminated with significant fission product radionuclides since all activity is attributed to uranium. This survey will be performed as backup to the visual inspection described in IROFS CS-21. An independent person from the one performing the visual inspection of IROFS CS-21 will perform the survey. The control will be exercised at designated waste consolidation stations, holding specifically approved waste containers, on the waste items using survey instrument(s) and setpoint(s) approved by the Criticality Safety Manager. Waste consolidation will be conducted after independent verification of the two methods of quantifying uranium mass has been performed. IROFS CS-22 is applicable to radiological waste generated outside the hot cell boundary that has had a low risk for direct contact with uranium-bearing materials.

13.2.7.2.14 IROFS CS-23, Non-Destructive Assay of Items with Inaccessible Surfaces

As an AAC, a nondestructive assay (NDA) method will be used on approved waste streams to quantify the uranium mass prior to disposal. An approved waste container with an approved uranium mass limit will receive the waste. A running inventory of items and uranium mass will be maintained with the waste disposal container.

The purpose of this IROFS is to prevent accidental nuclear criticality by controlling the mass of enriched uranium that is disposed in a non-safe geometry waste container. At designated waste consolidation stations holding specifically approved waste containers, the control will be exercised on the waste items using NDA techniques and mass or concentration limits approved by the Criticality Safety Manager. The waste will not be consolidated until independent measurements conducted according to IROFS CS-24 are completed. The item will be controlled during the waste measurement analysis period.

13.2.7.2.15 IROFS CS-24, Independent NDA of Items with Inaccessible Surfaces

As an AAC, an independent NDA method will be used on approved waste streams to quantify the uranium mass prior to disposal. An approved waste container with an approved uranium mass limit will receive the waste. A running inventory of items and uranium mass will be maintained with the waste disposal container.

The purpose of this IROFS is to prevent accidental nuclear criticality by controlling the mass of enriched uranium that is disposed in a non-safe geometry waste container. The control will be used as a backup to IROFS CS-16, IROFS CS-21 or IROFS CS-23, as approved by the Criticality Safety Manager for each waste stream. At designated waste consolidation stations holding specifically approved waste containers, the control will be exercised on the waste items using NDA techniques and mass or concentration limits approved by the Criticality Safety Manager. Waste consolidation will be conducted after independent verification of the two methods of quantifying uranium mass has been performed.

13.2.7.2.16 IROFS CS-25, Target Housing Weighing Prior to Disposal

As an AAC, on disposal of empty target housings, target housing pieces will be weighed and the weight compared to the original housing tare weight. The removed LEU target material will be weighed, and the weight compared to the original loading of LEU target material prior to disposal. The weights will agree within tolerances approved by the Criticality Safety Manager. Any differences will be attributed as [Proprietary Information] mass remaining in the wastes. An approved waste container with an approved uranium mass limit will receive the waste. A running inventory of items and uranium mass will be maintained with the waste disposal container.

The purpose of this IROFS is to prevent accidental nuclear criticality by controlling the mass of enriched uranium that is disposed in a non-safe geometry waste container. The control will be used as a backup to IROFS CS-16 for the disposal of target housings. At designated waste consolidation stations holding specifically approved waste containers, the control will be exercised on the waste items weighed on approved scales and at mass or concentration setpoint(s) approved by the Criticality Safety Manager. Waste consolidation will be conducted after independent verification of the two methods of quantifying uranium mass (the go/no-go method of IROFS CS-16, and the quantitative method of IROFS CS-25) have been performed.

13.2.7.2.17 IROFS CS-26, Processing Component Safe Volume Confinement

As a PEC, some processing components (e.g., pumps, filter housings, and IX columns) will be controlled to a safe volume for safe storage and processing of fissile solutions. The safety function of the safe volume component is also one of confinement of the contained solution. The safe volume confinement of fissile solutions will prevent accidental nuclear criticality, a high consequence event. The safe volume confinement conservatively includes the outside diameter of any heating or cooling jackets (or any other void spaces that may inadvertently capture fissile solution) on the component. Where insulation is used on the outside wall of the component, the insulation will be closed foam or encapsulated type (so as not to soak up solution during a leak) and will be compatible with the chemical nature of the contained solution.

13.2.7.3 Items Relied on for Safety for Fire or Explosion Accident Sequences (S.F.)

The following IROFS fall under the fire or explosion accident sequence category and are not discussed elsewhere in this chapter.

13.2.7.3.1 IROFS FS-05, Exhaust Stack Height

As a PEC, the exhaust stack is designed and fabricated with a fixed height for safe release of the gaseous effluents.

13.2.7.3.2 IROFS FS-02, Overhead Cranes

Overhead cranes will be designed, operated, and tested according to ASME B30.2, *Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)*. Lifting devices for shipping containers will be designed, operated, and tested according to ANSI N14.6, *Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More for Nuclear Materials*.

The safety function of IROFS FS-02 is to prevent (by reducing the likelihood) mechanical failure of cranes during heavy lift activities. This IROFS will be implemented through the facilities configuration management and management measures programs.

13.2.7.3.3 IROFS FS-03, Process Vessel Emergency Purge System

As an AEC, an emergency backup set of bottled nitrogen gas will be provided for tanks that have the potential to reach the hydrogen lower flammability limit either through the radiolytic decomposition of water or through reaction with the nitric acid (or other reagents added during processing). The system will monitor the pressure or flow going to the header and open an isolation valve on low pressure or flow (setpoint to be determined) to restore the sweep gas flow to the system using nitrogen. The system will be configured to provide more than 24 hr of sweep gas for the required tanks.

The safety function of IROFS FS-03 is to prevent a hydrogen-air mixture in the tanks from reaching lower flammability limit conditions to prevent the deflagration or detonation hazard. The purge gases will be exhausted through the dissolver offgas or the process vessel ventilation system. The system is designed to sense low pressure or flow on the normal sweep system and introduce a continuous purge of nitrogen from a reliable emergency backup station of bottled nitrogen into each affected vessel.

13.2.7.4 Items Relied on for Safety for Natural Phenomena Accident Sequences (S.N.)

The IROFS under the natural phenomena accident sequence category are discussed in Section 13.2.6.

13.2.7.5 Items Relied on for Safety for Man-Made Accident Sequences (S.M.)

There are no IROFS specifically identified for the man-made accident sequence category.

13.2.7.6 Items Relied on for Safety for Chemical Accident Sequences (S.CS.)

There are no IROFS specifically identified for the chemical accident sequence category.

13.3 ANALYSIS OF ACCIDENTS WITH HAZARDOUS CHEMICALS

This section analyzes the hazardous chemical-based accident sequences identified in the PHA.

13.3.1 Chemical Burns from Contaminated Solutions During Sample Analysis

13.3.1.1 Chemical Accident Description

This accident sequence occurs during sampling and analysis activities performed outside the hot cell confinement and shielding boundary where facility personnel (operators and/or technicians) may handle radioactively contaminated acidic or caustic solutions. There are two possible modes of occurrence for this accident.

- A sample container is dropped during handling activities outside a laboratory hood, resulting in a spill/splash event.
- A spill occurs during sample handling or analysis where the container is required to be opened.

13.3.1.2 Chemical Accident Consequences

Either of the modes described above can result in damage to skin and/or eye tissue on exposure to the acidic or caustic sample solution. This accident sequence may result in long-term or irreversible tissue damage, particularly to the eyes.

13.3.1.3 Chemical Process Controls

Facility personnel will be required to follow strict protocols for sampling and analysis activities at the RPF. Sampling locations, techniques, containers to be used, routes to take through the RPF when transporting a sample, analysis procedures, reagents, analytical equipment requirements, and sample material disposal protocols will all be specified per procedures and/or work plans prepared and discussed prior to sampling or analytical activities. Operators and technicians will be required to wear personal protective equipment, specifically for eye and skin protection.

Radiologically contaminated acidic and caustic solution samples will be handled in approved containers. Containers will be properly sealed when removed from sample locations and vent hoods during transport and/or storage.

Sample containers will also be opened only when securely located in an approved laboratory hood, with the hood lowered for spray protection. This process will provide an additional layer of protection for eyes and skin (e.g., protective eyewear/face shield, laboratory coat or apron, anti-contamination chemical resistant gloves, etc.).

13.3.1.4 Chemical Process Surveillance Requirements

Specific surveillance requirements will be identified in the Operating Permit Application. For this accident sequence, surveillance may consist of management auditing or oversight of sampling and analysis activities to ensure adherence to the specified protocol of procedures, personal protective equipment usage, approved container usage, and laboratory hood etiquette.

13.3.2 Nitric Acid Fume Release

13.3.2.1 Chemical Accident Description

This accident consists of a release of nitric acid fumes inside or outside of the RPF originating from one of the nitric acid storage tanks in the chemical storage and preparation room.

13.3.2.2 Chemical Accident Consequences

Chapter 19.0 identifies hazardous chemical release scenarios for the facility using several of the stored chemicals. A 1-hr release of the bounding RPF inventory of 5,000 L of nitric acid was shown to cause a concentration of 1,200 parts per million (ppm) at the controlled area fence line and 19.1 ppm at 434 m (1,425 ft) (nearest resident location) under dispersion conditions of moderate wind. Unmitigated exposure to a nearby worker would be much higher. The AEGL-2, 60-minute (min) exposure limit for nitric acid is 24 ppm, which is high consequence to the public. AEGL-3, the 10-min exposure limit, is 170 ppm for a high consequence exposure to the worker. These determinations were made using the ALOHA (Areal Locations of Hazardous Atmospheres) computer code for estimating the consequences of chemical releases. The use of ALOHA is recognized by the NRC in NUREG/CR-6410.

The impact and consequences of a chemical release on RPF operations would require personnel to either evacuate the facility or, under some circumstances, shelter in place depending on the location of the event.

13.3.2.3 Chemical Process Controls

The RPF will follow U.S. Environmental Protection Agency and Occupational Safety and Health Administration regulations for design, construction, and operation of chemical preparation and storage areas. Chemical handling procedures will be provided to operators to ensure safe handling of chemicals according to applicable regulatory requirements and consistent with the applicable material safety data sheets.

IROFS to prevent or mitigate events that could impact the chemical storage tanks in the RPF chemical storage and preparation room are addressed in Section 13.2.5.

13.3.2.4 Chemical Process Surveillance Requirements

Specific surveillance requirements for chemical use and storage at the RPF will be identified in the Operating Permit Application.

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NORTHWEST MEDICAL ISOTOPES

Chapter 14.0 – Technical Specifications

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 3
September 2017

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Chapter 14.0 – Technical Specifications

Construction Permit Application for Radioisotope Production Facility

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Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required
1	8/5/2017	Incorporate changes based on responses to NRC Requests for Additional Information	C. Haass
2	N/A		
3	9/5/2017	Incorporate final comments from NRC Staff and ACRS; full document revision	C. Haass

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TABLES

TERMS

Acronyms and Abbreviations

AC	administrative control
ANS	American Nuclear Society
ANSI	American National Standards Institute
CFR	Code of Federal Regulations
IROFS	items relied on for safety
ISA	integrated safety analysis
LCO	limiting condition of operation
LSSS	limiting safety system setting
NWMI	Northwest Medical Isotopes, LLC
RAM	radioactive material
RPF	Radioisotope Production Facility
SL	safety limit
SNM	special nuclear material
SSC	systems, structures, and components

14.0 TECHNICAL SPECIFICATIONS

This chapter describes the process by which the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) technical specifications will be developed and written. For the Construction Permit Application, NWMI has prepared the strategy and content of what will be required for technical specifications during RPF operations. No technical specifications were developed for the Construction Permit Application. The technical specifications will be included in the submission of the Operating License Application. The variables or conditions listed in Table 14-1 are probable subjects of technical specifications based on their involvement with preventing release of radioactive materials routinely or in the event of an accident.

Table 14-1. Potential Technical Specifications

Item or variable	Reason
Uranium mass limits on batches, samples, and approved containers[a]	Criticality control
Spacing requirements on targets and containers with SNM[a]	Criticality control
Floor and sump designs[a]	Criticality control
Hot cell liquid confinement[a]	Criticality control
Process tank size and spacing[a]	Criticality control
Evaporator condensate monitor	Criticality control
Criticality monitoring system	Criticality control
In-line uranium content monitoring	Criticality control
Air pressure differential between zones[a]	Control of airborne RAM
Ventilation system filtration[a]	Control of airborne RAM
Process offgas subsystem	Control of airborne RAM
Primary offgas relief system	Control of airborne RAM
Hot cell shield thickness and integrity[a]	Occupation and general public dose reduction
Hot cell secondary confinement boundary[a]	Control of airborne RAM
Double-wall piping	Control of liquid RAM/criticality control
Process closed heating and cooling loops	Control of both airborne and liquid RAM
System backflow prevention devices	Control of liquid RAM/criticality control
Stack height[a]	Control of airborne RAM
Area radiation monitoring system	Occupation and general public dose reduction

[a] Items that will significantly influence the final design.

RAM = radioactive material. SNM = special nuclear material.

The format and content of the technical specifications for the RPF will be based on the guidance provided in American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1, *The Development of Technical Specifications for Research Reactors*; NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*; and the final interim staff guidance augmenting NUREG-1537 (NRC, 2012). The technical specifications will be consistent with Title 10, *Code of Federal Regulations*, Part 50.34, "Contents of Applications; Technical Information," and will address the applicable paragraphs of 10 CFR 50.36, "Technical Specifications." However, the technical specifications will be written to address the differences between the RPF and either power or research reactors.

The proposed technical specifications will form a comprehensive set of parameters to ensure that normal RPF operations will not result in off-site radiation exposures in excess of the guidelines in 10 CFR 20, "Standards for Protection Against Radiation," and also reasonably ensure that the RPF will function as analyzed in the Operating License Application. Adherence to the technical specifications will limit the likelihood of malfunctions and mitigate the consequences to the public of off-normal or accident events.

The RPF integrated safety analysis (ISA) process identified systems, structures, or components (SSC) that are defined as items relied on for safety (IROFS). The importance of these SSCs will also need to be reflected in the technical specifications. Each IROFS will need to be examined and likely translated into a limiting condition of operation (LCO). This translation will involve identifying the most appropriate specification to ensure operability and a corresponding surveillance periodicity for the IROFS. An IROFS could potentially be translated into a design function but this seems less likely than translating it into a LCO.

The outline for the technical specifications that will be prepared during development of the Operating License Application is provided below.

14.1 OUTLINE

14.1.1 Introduction

The introductory section will identify the scope, purpose, and format of the technical specifications. A list of definitions will be identified to provide consistent language throughout the document.

Term	Definition
Actions	Actions are that part of a limiting condition for operation that prescribes Required Actions to be taken under designated conditions within specified completion times.
Administrative control (AC)	...(described in Section 14.1.6)
Design features	...(described in Section 14.1.5)
Limiting condition for operation (LCO)	...(described in Section 14.1.3)
Limiting safety system setting (LSSS)	...(described in Section 14.1.2)
Modes	Modes are used to (1) determine safety limits, limiting control settings, limiting conditions for operation, and administrative controls program applicability, (2) distinguish facility operational conditions, (3) determine minimum staffing requirements, and (4) provide an instant facility status report.

Term	Definition
Operable/ operability	A system, subsystem, component, or device shall be operable or have operability when it is capable of performing its specified safety function(s), and (1) setpoints are within limits, (2) operating parameters necessary for operability are within limits, and (3) when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, component, or device to perform its safety function(s) are also capable of performing their related safety support function(s).
Safety limit (SL)	...(described in Section 14.1.2)
Shall	Denotes a mandatory requirement that must be complied with to maintain the requirements, assumptions, or conditions of the facility safety basis.
Surveillance requirements	...(described in Section 14.1.4)
Verify/verification	A qualitative assessment to confirm or substantiate that specific plant conditions exist. This assessment may include collecting sample data or quantitative data; taking instrument readings; recording data and information on logs, datasheets, or electronic media; and evaluating data and information according to procedures.

14.1.2 Safety Limit and Limiting Safety System Setting

Safety limits (SL) will be established from basic physical conditions, as determined by appropriate process variables, to ensure that the integrity of the principal physical barrier is maintained if the SLs are not exceeded. Limiting safety system settings (LSSS) will be established for the operation of the RPF to defend the SL. The LSSS will be limiting values for setting instrumentation by which point protective action will be initiated. SLs for radiochemical and chemical processing will be developed to maintain operations within limits pursuant to 10 CFR 50.36 to protect workers and the public. As an example, the amount of radioactive material will be limited so as not to exceed the shielding and confinement capabilities of the systems and components in which the materials are processed or stored. Each SL and LSSS will have an identified applicability, objective, specification, and basis. Currently, neither the SL nor LSSS have been specifically identified but may be part of the Operating License Application.

14.1.3 Limiting Condition of Operation

Administratively established constraints on equipment and operational characteristics will be identified and described. These limits will be the lowest functional capability or performance level required for safe operation of the facility. Each LCO will have an identified applicability, objective, specification, and basis. The basis of each LCO will be provided and consistent with analysis provided in the Operating License Application. Anticipated systems covered in this section include containment, ventilation, effluent monitoring, and criticality monitoring. Windows, or short time periods, of approved inoperability will be established to create operational flexibility. The basis of these windows will be analyzed in the Operating License Application.

14.1.4 Surveillance Requirements

A set of requirements will provide maximum intervals for checks, tests, and calibrations for each system or component identified in Section 14.1.3 to verify a minimum performance or operability level. The basis for each will be identified and will be derived from either an analysis presented in the Operating License Application or experience, engineering judgment, or manufacturer recommendations.

14.1.5 Design Features

This section will establish the minimum design functions of safety-related SSCs, particularly construction or geometric arrangements. These design functions, if altered or modified, are implied to significantly affect safety and will not be identified in other sections. Anticipated areas covered in this section include the site and facility description, and fissionable material storage. Design features that will be provided in the technical specifications are the features of the RPF (e.g., materials of construction and geometric arrangements) that would have a significant effect on safety if those features were altered or modified. The requirements of 10 CFR 50.36(c)(4) are specified here as they pertain to the above referenced processes.

14.1.6 Administrative Controls

This section will establish the administrative structure and controls for the RPF and will identify the roles, responsibility, and reporting lines for NWMI management (e.g., Levels 1 through 4). Other requirements include:

- Identifying minimum staffing and supervisory functions
- Preparing and maintaining call lists
- Selecting and training personnel
- Developing a process for creating and modifying procedures
- Identifying actions to be taken in case of an SL violation (if applicable), exceeding an LCO, or release of radioactivity in excess of regulatory limits
- Developing reporting requirements for annual operating conditions or events
- Specifying records retention

This section will also identify the creation of a Review and Audit Committee and will address the establishment of a charter, review and audit functions, quorum requirements, membership expertise, and meeting frequency for the committee.

14.2 REFERENCES

10 CFR 20, "Standards for Protection Against Radiation," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*, Office of the Federal Register, as amended.

ANSI/ANS 15.1, *The Development of Technical Specifications for Research Reactors*, American National Standards Institute/American Nuclear Society, LaGrange Park Illinois, 2013.

NRC, 2012, *Final Interim Staff Guidance Augmenting NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Parts 1 and 2, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors*, Docket ID: NRC-2011-0135, U.S. Nuclear Regulatory Commission, Washington, D.C., October 30, 2012.

NUREG-1537 (Part 1), *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content*, U.S. Nuclear Regulatory Commission, Washington, D.C., February 1996.

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Chapter 15.0 – Financial Qualifications

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Chapter 15.0 – Financial Qualifications

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CONTENTS

TABLES

TERMS

Acronyms

^{99}Mo	molybdenum-99
CFR	Code of Federal Regulations
FOCD	foreign ownership, control, or domination
LEU	low-enriched uranium
LLC	limited liability corporation
NRC	U.S. Nuclear Regulatory Commission
NWMI	Northwest Medical Isotopes, LLC
RPF	radioisotope production facility
SRP	Standard Review Plan
U.S.	United States
U.S.C.	United States Code

15.0 FINANCIAL QUALIFICATIONS

Financial information for Northwest Medical Isotopes, LLC (NWMI) is presented in this chapter per Title 10, *Code of Federal Regulations*, Part 50, "Domestic Licensing of Production and Utilization Facilities," Subparts 50.33(d)(3)(iii), 50.33(f), and 50.33(k). Information regarding the Price-Anderson Act, Section 170 of the Atomic Energy Act of 1954 (42 U.S.C. § 2011 et seq.), as amended, is also provided. This information establishes NWMI's financial qualifications to design, construct, operate, and decommission, and to own a radioisotope production facility (RPF). The following information is presented in the following sections:

- Financial ability to construct an RPF authorized by the Construction Permit (Section 15.1)
- Financial ability to safely operate an RPF (Section 15.2)
- Financial ability to safely decommission an RPF (Section 15.3)
- Information regarding foreign ownership, control, or domination (FOCD) (Section 15.4)
- Information regarding nuclear insurance and indemnity (Section 0)

15.1 FINANCIAL ABILITY TO CONSTRUCT A FACILITY

The U.S. Nuclear Regulatory Commission (NRC) requires that an applicant for a construction permit submit sufficient financial information to demonstrate reasonable assurance that the applicant can obtain the necessary funds to cover the estimated design, construction, and startup costs for the RPF, and the related fuel-cycle costs (e.g., for low-enriched uranium [LEU] from the U.S. Department of Energy) pursuant to 10 CFR 50.33(f). In addition, the applicant is required to indicate source(s) of the funds to cover the costs.

The financial guidelines to be followed by the applicant are provided in 10 CFR 50, Appendix C, "A Guide for the Financial Data and Related Information Required to Establish Financial Qualifications for Construction Permits and Combined Licenses." This appendix (1) distinguishes between applicants that are established organizations and those that are newly formed entities organized primarily for the purpose of engaging in the activity for which the permit is sought, and (2) provides a guide for the financial data and related information required to establish financial qualifications for construction permits. NWMI is considered a newly formed entity per 10 CFR 50, Appendix C.

NWMI is submitting information that demonstrates the company possesses or has reasonable assurance of obtaining the necessary funds to cover estimated design, construction, and startup costs and the related fuel-cycle costs.

NWMI is submitting information demonstrating that the company possesses or has reasonable assurance of obtaining the necessary funds to cover estimated design, construction, and startup costs, and related fuel-cycle costs. The estimated NWMI costs to construct an RPF are summarized below. These estimates are based on NWMI's preliminary design of the RPF completed in May 2015. The estimated NWMI costs to construct an RPF are summarized below.

Total facility costs	[Proprietary Information]
Plant equipment	[Proprietary Information]
LEU costs for RPF startup and first year	[Proprietary Information]
Total estimated costs	**[Proprietary Information]**

NWMI prepared an RPF base estimate that covers all components of the project (e.g., scope, conditions, and characteristics), including engineering and construction equipment, materials, and labor. The estimate incorporates data from previous and similar projects and NWMI's preliminary RPF time-cycle logistical study that includes data for labor requirements, materials, operations, and maintenance. The base estimate also used inputs from the completed project file, project schedule, and knowledge of site conditions. The estimate was escalated to the year of construction dollars using a construction cost index and to the mid-point of construction. NWMI developed clear and concise documentation for traceability that will allow future updates, review, and validation of the estimate.

To date, NWMI has received [Proprietary Information] in equity financing and anticipates facility financing [Proprietary Information] for the final design and construction of the RPF using various sources of financing, including equity and debt to be completed in the 3rd quarter 2016. Total RPF estimated costs are [Proprietary Information]. NWMI research and development, preliminary design, regulatory, and permitting cost projections are fully funded through existing equity financing receipts and commitments. NWMI has established a wholly owned subsidiary for the RPF and expects construction to be debt-financed. The RPF site is located in the Discovery Ridge Research Park (Columbia, Missouri) and on land owned by the University of Missouri system and will be leased for [Proprietary Information].

15.2 FINANCIAL ABILITY TO SAFELY OPERATE A FACILITY

NWMI will be applying for a Class 103 license per 10 CFR 50.22, "Class 103 licenses; for Commercial and Industrial Facilities," and 10 CFR 70, "Domestic Licensing of Special Nuclear Material." Additional future applications will be applied for, including receipt, possession, and use of source material under 10 CFR 40, "Domestic Licensing of Source Material," and byproduct material under 10 CFR 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." NWMI expects to request an operating license for a term of 30 years.

NWMI is providing financial information that demonstrates the company possesses or has reasonable assurance of obtaining the funds necessary to cover estimated facility operational costs for the term of the operating license. Table 15-1 provides the estimated NWMI RPF operating costs and expected revenues for the first five years of RPF commercial operations.

Table 15-1. Estimated Radioisotope Production Facility's Operating Costs and Expected Revenues for Years 1–5

$000	2018	2019	2020	2021	2022
Revenue	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Cost of goods sold	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Gross profit	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
% Gross profit	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Operating expenses	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Income from operations	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Non-operating expenses	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Income taxes	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Net income	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Net income % of revenue	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

Pursuant to 10 CFR 50.33(f)(2), the sources of funds to cover these costs will be derived from the expected revenues associated with the sale of molybdenum-99.

NWMI prepared the RPF operations base estimate based on previous and similar projects and base cost estimating. The operations base estimate also used inputs from NWMI's preliminary RPF time-cycle logistical study that includes data for labor requirements, materials, operations, and maintenance. NWMI developed clear and concise documentation for traceability that will allow future updates, review, and validation of the estimate.

15.3 FINANCIAL ABILITY TO SAFELY DECOMMISSION A FACILITY

NWMI will provide financial information that demonstrates reasonable assurance that funds will be available to decommission the RPF in accordance with 10 CFR 50.33(f) as part of the Operating License application. In addition, the financial information will be submitted in accordance with 10 CFR 50.75(d).

In addition, pursuant to 10 CFR 50.75(e), the RPF decommissioning report will contain financial assurances, including a cost estimate for the RPF decommissioning, identification of which method(s) will be used to provide funds for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the operational life of the RPF to account for changes in labor, energy, and waste disposal.

Based on previous experience and discussions with nuclear industry experts, NWMI has developed a preliminary cost estimate for decommissioning the RPF to be [Proprietary Information]. NWMI's current business strategy anticipates that decommissioning of the RPF will be financed by an external escrow account in which deposits will be made annually, coupled with either a surety method, insurance, or some other form of guaranty. Financial projections assume that the annual escrow deposit will be approximately [Proprietary Information] and adjusted for inflation periodically, which provides reasonable assurance that decommissioning funds will be available for the RPF.

The NWMI RPF Decommissioning Plan, including detailed costs and associated financial assurances, will be provided in the Operating License application. The estimated costs of decommissioning will be developed using the analysis of the RPF design and analysis of estimates and actual costs of decommissioning similar facilities.

15.4 FOREIGN OWNERSHIP, CONTROL, OR DOMINATION

NWMI understands that the NRC will evaluate our application in a manner that is consistent with the guidance provided in the Standard Review Plan (SRP) regarding "Foreign Ownership, Control, or Domination of applicants for Reactor Licenses," June 1999, referred to as the "SRP on FOCD." This evaluation will determine whether NWMI is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

The NRC's position outlined in the SRP on FOCD states "the foreign control prohibition should be given an orientation toward safeguarding the national defense and security." Furthermore, the SRP on FOCD outlines how the effects of foreign ownership may be mitigated through implementation of a "negation action plan" to ensure that any foreign interest is effectively denied control or domination over the applicant.

NWMI fully understands that a financial analyst will review all of the information submitted by the company to determine whether there is FOCD. If it is determined that there is FOCD, additional action would be necessary to negate FOCD, and the applicant would be advised and requested to submit a Negative Action Plan.

NWMI is a limited liability company organized under the laws of the state of Oregon. NWMI is *not* owned, controlled, or dominated by alien, foreign corporation, or foreign government. In addition, NWMI is not acting as an agent or representative of another person or company in filing the Construction Permit Application.

NWMI is governed and managed by a six-member Board of Managers, all of whom are U.S. citizens. NWMI currently has 18 members. To the best of our knowledge, all members holding more than one percent of NWMI's membership interests are U.S. citizens or entities owned or controlled by U.S. citizens.

15.5 NUCLEAR INSURANCE AND INDEMNITY

The Price-Anderson Act provides a system to pay funds for claims by members of the public for personal injury and property damage resulting from any nuclear incident. The Price-Anderson Act provides coverage in varying degrees. The implementing regulations regarding the Price-Anderson Act are provided in 10 CFR 140, "Financial Protection Requirements and Indemnity Agreements."

NWMI understands the requirement to have and maintain financial protection and insurance requirements under the Price-Anderson Act. The NWMI RPF is planned to be licensed under both 10 CFR 50 for the processing of irradiated LEU to recover ^{99}Mo and recycle LEU, and 10 CFR 70 for the fabrication of LEU targets that will be irradiated in a network of domestic university reactors. Prior to the RPF becoming operational, NWMI plans to obtain and maintain financial protection in the form of nuclear liability insurance. The amount of insurance required will be developed and finalized during the Operations License Application.

NWMI will also execute and maintain an indemnification agreement with the NRC for the duration of the RPF Operating License. In addition, pursuant to 10 CFR 140.13, "Amount of Financial Protection Required of Certain Holders of Construction Permits and Combined Licenses under 10 CFR 52," NWMI will maintain financial protection of $1 million in insurance prior to fuel (or LEU) being accepted by NWMI at the RPF, and full financial protection prior to operation of the RPF.

NWMI will not purchase property insurance pursuant to 10 CFR 50.54(w).

15.6 REFERENCES

10 CFR 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 40, "Domestic Licensing of Source Material," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*, Appendix C, "A Guide for the Financial Data and Related Information Required To Establish Financial Qualifications for Construction Permits and Combined Licenses," Office of the Federal Register, as amended.

10 CFR 50.22, "Class 103 Licenses; for Commercial and Industrial Facilities," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 50.33, "Contents of Applications; General Information," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 50.54, "Conditions of Licenses," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 50.75, "Reporting and Recordkeeping for Decommissioning Planning," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 70, "Domestic Licensing of Special Nuclear Material," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 140, "Financial Protection Requirements and Indemnity Agreements," *Code of Federal Regulations*, Office of the Federal Register, as amended.

10 CFR 140.13, "Amount of Financial Protection Required of Certain Holders of Construction Permits and Combined Licenses under 10 CFR 52," *Code of Federal Regulations*, Office of the Federal Register, as amended.

42 U.S.C. § 2011 et seq., "Atomic Energy Act of 1946," *United States Code*, as amended.

64 FR 52355, "Final Standard Review Plan on Foreign Ownership, Control, or Domination," *Federal Register*, Volume 64, Issue 187, U.S. Nuclear Regulatory Commission, Washington, D.C., September 28, 1999.

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Chapter 16.0 – Other License Considerations

Construction Permit Application for Radioisotope Production Facility

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Chapter 16.0 – Other License Considerations

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CONTENTS

TERMS

Acronyms and Abbreviations

NWMI	Northwest Medical Isotopes, LLC
RPF	Radioisotope Production Facility

16.0 OTHER LICENSE CONSIDERATIONS

16.1 PRIOR USE OF FACILITY COMPONENTS

Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) will only use new and appropriately qualified components and systems to conduct all special nuclear material and radioisotope production processes. Thus, discussions involving used components and systems are not applicable to the NWMI RPF.

16.2 MEDICAL USE OF THE RADIOISOTOPE PRODUCTION FACILITY

NWMI RPF does not include equipment or facilities associated with direct medical administration of radioisotopes or other radiation-based therapies. Thus, discussions involving medical use of the RPF is not applicable for this Construction Permit Application.

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Chapter 17.0 – Decommissioning and Possession-Only License Amendments

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Chapter 17.0 – Decommissioning and Possession-Only License Amendments

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TERMS

Acronyms and Abbreviations

CFR	Code of Federal Regulations

17.0 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

17.1 DECOMMISSIONING

Per Title 10, *Code of Federal Regulations*, Subpart 50.34, "Contents of Applications; Technical Information," (10 CFR 50.34) paragraph (a)(1)(i), a construction permit applicant for a non-power reactor (or production facility) is required to submit in their Construction Permit Application the information prescribed in 10 CFR 50.34, paragraphs (a)(2) through (a)(8). Thus, the Construction Permit Application is not required to include a decommissioning plan. A decommissioning report will be submitted in accordance with 10 CFR 50.33(k)(1) with the Operating License Application.

17.2 POSSESSION-ONLY LICENSE AMENDMENTS

This section relates to a possession-only license and is not applicable to the Northwest Medical Isotopes, LLC Radioisotope Production Facility.

17.3 REFERENCES

10 CFR 50.34, "Contents of Applications; Technical Information," *Code of Federal Regulations*, Office of the Federal Register, as amended.

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NORTHWEST MEDICAL ISOTOPES

Chapter 18.0 – Highly Enriched Uranium to Low-Enriched Uranium Conversion

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Chapter 18.0 – Highly Enriched Uranium to Low-Enriched Uranium Conversion

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18.0 HIGHLY ENRICHED URANIUM TO LOW-ENRICHED URANIUM CONVERSION

This chapter of the Construction Permit Application addressing the conversion of highly enriched uranium to low-enriched uranium is not applicable to the Northwest Medical Isotopes, LLC Radioisotope Production Facility.

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