Chapter 8.0 – Electrical Power Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1
June 2017

Prepared by:
Northwest Medical Isotopes, LLC
815 NW 9th Ave, Suite 256
Corvallis, OR 97330

This page intentionally left blank.

Chapter 8.0 – Electrical Power Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1

Date Published:
June 26, 2017

***Document Number*: NWMI-2013-021**	***Revision Number*: 1**
***Title*: Chapter 8.0 – Electrical Power Systems Construction Permit Application for Radioisotope Production Facility**	
***Approved by*: Carolyn Haass**	***Signature*:** Carolyn C Haass

This page intentionally left blank.

REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required
1	6/26/2017	Incorporate changes based on responses to NRC Requests for Additional Information	C Haass

This page intentionally left blank.

CONTENTS

8.0 ELECTRICAL POWER SYSTEMS 8-1
8.1 Normal Electrical Power Systems 8-2
8.1.1 Design Basis of the Normal Electric Power System 8-4
8.1.2 Design for Safe Shutdown 8-5
8.1.3 Ranges of Electrical Power Required 8-5
8.1.4 Use of Substations Devoted Exclusively to the Radioisotope Production Facility 8-6
8.1.5 Special Processing of Electrical Service 8-6
8.1.6 Design and Performance Specification 8-6
8.1.7 Special Routing or Isolation 8-6
8.1.8 Deviations from National Codes 8-6
8.1.9 Technical Specifications 8-6
8.2 Emergency Electrical Power Systems 8-7
8.2.1 Design Basis of the Emergency Electric Power System 8-8
8.2.2 Ranges of Emergency Electrical Power Required 8-8
8.2.3 Power for Safety-Related Instruments 8-8
8.2.4 Power for Effluent, Process, and Area Radiation Monitors 8-8
8.2.5 Power for Physical Security Control, Information, and Communication Systems 8-8
8.2.6 Power to Maintain Experimental Equipment in Safe Condition 8-8
8.2.7 Power for Active Confinement/Containment Engineered Safety Feature Equipment and Control Systems 8-8
8.2.8 Power for Coolant Pumps or Systems 8-9
8.2.9 Power for Emergency Cooling 8-9
8.2.10 Power for Engineered Safety Feature Equipment 8-9
8.2.11 Power for Emergency Lighting 8-9
8.2.12 Power for Instrumentation and Control Systems to Monitor Shutdown 8-9
8.2.13 Technical Specifications 8-9
8.3 References 8-10

FIGURES

Figure 8-1. Radioisotope Production Facility Electrical One Line Diagram 8-3

TABLES

Table 8-1. Summary of Radioisotope Production Facility and Ancillary Facilities Electrical Loads (2 pages) 8-1

TERMS

Acronyms and Abbreviations

AEC	active engineering control
ATS	automatic transfer switch
CAAS	criticality accident alarm system
HVAC	heating, ventilation, and air conditioning
IEEE	Institute of Electrical and Electronics Engineers
IROFS	item relied on for safety
MCC	motor control center
NEP	normal electrical power
NFPA	National Fire Protection Association
NO_x	nitrogen oxides
NWMI	Northwest Medical Isotopes, LLC
RPF	Radioisotope Production Facility
SEP	standby electrical power
UPS	uninterruptable power supply

Units

gal	gallon
hp	horsepower
hr	hour
Hz	hertz
km	kilometer
kV	kilovolt
kW	kilowatt
L	liter
mi	mile
min	minute
sec	second
V	volt

This page intentionally left blank.

8.0 ELECTRICAL POWER SYSTEMS

This chapter provides a description of the normal electrical power (NEP) and emergency electrical power systems within the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF). The RPF design uses high-quality, commercially available components and wiring in accordance with applicable code. Electrical power circuits will be isolated sufficiently to avoid electromagnetic interference with safety-related instrumentation and control functions. The facility is designed for passive, safe shutdown and to prevent uncontrolled release of radioactive material if NEP is interrupted or lost. Uninterruptable power supplies (UPS) automatically provide power to systems that support the safety functions protecting workers and the public.

The NEP system is designed to provide reasonable assurance that use or malfunction of electrical power systems will not damage the RPF or prevent safe RPF shutdown. In addition, the RPF has a non-safety standby electrical power (SEP) system to reduce or eliminate process downtime due to electrical outages. A combination of UPSs and the SEP system will provide emergency electrical power (defined in Section 8.2) to the RPF.

Table 8-1 lists the RPF electrical loads, including the NEP system peak loads, which systems have UPSs, and the loads for those systems supported by the SEP system.

Table 8-1. Summary of Radioisotope Production Facility and Ancillary Facilities Electrical Loads (2 pages)

Demand	Normal electrical peak power load		Uninterruptable power	Standby electrical peak power load	
	kW	hp		kW	hp
Target fabrication system	125	168	No	0	0
Target receipt and disassembly system	30	40	No	0	0
Target dissolution system	40	54	No	40	54
Molybdenum recovery and purification system	30	40	No	25	34
Uranium recovery and recycle system	10	13	No	10	13
Waste handling system	25	34	No	5	7
Radiation monitoring and CAAS systems	5	7	Yes[a]	5	7
Standby electrical power system	N/A		No	N/A	N/A
General facility electrical power	173	232	Yes[a]	101	135
Process vessel ventilation system	40	54	No	40	54
Facility ventilation system					
Ventilation Zone I	67	90	No	67	90
Ventilation Zone II/III	215	288	No	215	288
Ventilation Zone IV	295	396	No	295	396
Laboratory ventilation	38	51	No	10	13
Supply air	49	66	No	49	66
Fire protection system	0.8	1	Yes[a]	0[b]	0[b]
Plant and instrument air system	60	83	No	60	83
Gas supply system	0.8	1	No	0.8	1
Process chilled water system	280	375	No	140	188

Table 8-1. Summary of Radioisotope Production Facility and Ancillary Facilities Electrical Loads (2 pages)

Demand	Normal electrical peak power load		Uninterruptable power	Standby electrical peak power load	
	kW	hp		kW	hp
Facility chilled water system	1,300	1,743	No	0	0
Facility heated water system	47	63	No	0	0
Process stream system	0.8	1	No	0.8	1
Demineralized water system	0.8	1	No	0	0
Supply air system					
Chemical supply system	49	66	No	49	66
Facility process control and communications systems	5	7	Yes	5	7
Energy recovery	5	7	No	0	0
Safeguards and security	40	54	Yes	40	54
Administrative building	90	121	No	18	24
Waste management building	11	15	No	3	4

[a] Only parts of the system are provided with uninterruptable power supplies.

[b] The fire detection and fire alarm subsystems will be provided by an uninterruptable power supply with a 24-hr capacity. Chapter 9.0 provides additional detail.

CAAS = criticality accident alarm system N/A = not applicable.

8.1 NORMAL ELECTRICAL POWER SYSTEMS

The NEP system will connect to electric utility power from the off-site utility transmission and distribution system at a point of common coupling. This point of common coupling will be located near the property line on the NWMI site. The NEP distribution system will operate in a redundant electrical system topology from the utility transmission and distribution system to the 480 volt (V) service entrance switchgear that services the RPF electrical distribution system and the devices and equipment within the facility. The RPF electrical distribution system is designed to support the safety functions protecting workers, the public, special nuclear material activities, and radioisotope production operation processes, as described in Chapter 4.0, "Radioisotope Production Facility Description," and to minimize the number of points where a failure in the RPF is a single point of power conveyance.

Figure 8-1 provides electrical one-line diagrams for the electrical distribution topology. Power will be provided to the NWMI site from an underground utility feed ❶ to the pad-mounted switchgear located outside of the RPF building. Power will then be routed underground from the switchgear to the Administrative Building ❷ and the RPF ❸.

The underground feeders ❸ to the RPF will comprise two redundant full-capacity service laterals to the RPF. Each service lateral will support redundant full-capacity service transformers ❹ that will normally carry half the RPF load. Either of the RPF feeders can be opened and the tie breaker closed, as needed, allowing the other feeder to carry the entire RPF load.

Any RPF loads requiring SEP will be provided power from the diesel generator when required ❺.

[Proprietary Information]

Figure 8-1. Radioisotope Production Facility Electrical One Line Diagram

The two underground feeders will be located on each side of the switchgear and will normally carry approximately half of the electrical load. However, each underground feeder will be capable of carrying the entire load of the facility. The designed NEP topology will provide the RPF with redundancy. In addition, each underground feeder can be maintained and inspected independently, due to redundancy, while the RPF and associated safety functions are serviced with electrical power.

The 480 V service entrance equipment will have a main-tie-main arrangement on the service entrance electrical bus, with a service main on either end of a common bus. The common bus will be segregated by a tie-breaker. In normal mode operation, the two main breakers will be closed and the tie-breaker open. In the event one feeder is unavailable, the other feeder will carry the entire RPF load by opening the unavailable feeder main breaker and closing the tie breaker.

Electrical distribution on the load side of the 480 V service entrance switchgear and the heating, ventilation, and air conditioning (HVAC) redundant loads will be serviced from opposite sides of the switchgear through electrical equipment and feeders, including motor control centers (MCC), switchboards, and distribution panel boards. Equipment, systems, and devices designed with redundant or N+1 capability will be fed from opposite sides of the service entrance switchgear.

Systems requiring emergency electrical power in the event of the loss of NEP will be serviced by an on-site diesel generator through the SEP system. Section 8.2 provides additional information on the SEP system.

UPSs will be provided for selected systems for the RPF, as identified in Table 8-1. UPS systems include unit device, rack-mounted, and/or larger capacity cabinet units. These UPS systems will service loads requiring uninterruptable power on a short-term basis. The UPS systems will be backed up by the on-site diesel generator to extend the duration of power available to connected loads.

Internal to the RPF and Administration Building, the NEP distribution system will service end user equipment and devices. Feeders, busing, overcurrent protection, devices, and equipment will provide the conveyance and conductor protection throughout the building. Design of the electrical distribution system includes recommended practices from the Institute of Electrical and Electronics Engineers (IEEE) 493, *Recommended Practice for the Design of Reliable Industrial and Commercial Power Systems*, and IEEE 379, *Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*. The electrical distribution system topology will employ a redundant power conveyance system.

The distribution system will include overcurrent protective devices, surge arresters, fusing, relays, and similar safety-related protective devices. These safety devices will conform to the requirements of the National Fire Protection Association (NFPA) 70, *National Electric Code*, relevant IEEE standards and recommendations, and local codes and standards.

8.1.1 Design Basis of the Normal Electric Power System

The NEP system design basis will provide sufficient and reliable electrical power to the RPF systems and components requiring electrical power for normal operations, including the electrical requirements of the system, equipment, instrumentation, control, communication, and devices related to the safety functions and devices.

There are no items relied on for safety (IROFS) applicable to the NEP, per Chapter 13.0, "Accident Analysis," Section 13.2.5 (loss of power accident analysis scenario). The NEP will provide power to the active engineered control (AEC) systems through the instrumentation, monitoring, alarm, and related control systems. The design basis is provided in Chapter 3.0, "Design of Structures, Systems, and Components."

8.1.2 Design for Safe Shutdown

In the event of the loss of NEP, UPSs automatically provide power to the RPF systems and components that support the safety functions protecting workers and the public. The following systems and components are supported with UPSs:

- Process and facility monitoring and control systems
- Facility communication and security systems
- Emergency lighting
- Fire alarms
- Radiation protection and criticality accident alarm system (CAAS)

The UPSs will be designed to operate for a period of up to 120 minutes (min). The fire protection system will have a UPS that provides 24 hours (hr) of uninterrupted power. If NEP service is reestablished within a determined timeframe (to be provided in the Operating License Application), the RPF will resume normal operation. Upon loss of normal power:

- Inlet bubble-tight isolation dampers within the Zone I ventilation system will close, and the HVAC system will automatically be placed into the passive ventilation mode of operation
- The process vessel vent system will automatically be placed into the passive ventilation mode of operation, and all electrical heaters will cease operation as part of the passive operation mode
- Pressure-relief confinement system for the target dissolver offgas system will be activated on reaching the system relief setpoint, and dissolver offgas will be confined in the offgas piping, vessels, and pressure-relief tank
- Process vessel emergency purge system will be activated for hydrogen concentration control in tank vapor spaces
- Uranium concentrator condensate transfer line valves will be automatically configured to return condensate to the feed tank due to residual heating or cooling potential for transfer of process fluids to waste tanks
- Equipment providing a motive force for process activities will 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)

8.1.3 Ranges of Electrical Power Required

The RPF power service will be 480 V, 3-phase, 120 amp, 60 hertz (Hz). The total power required for the facility will be approximately 2,998 kilowatt (kW) (4,020 horsepower [hp]). Table 8-1 lists the loads for different locations and processes within the RPF.

8.1.4 Use of Substations Devoted Exclusively to the Radioisotope Production Facility

The RPF will receive power from Columbia Water and Light through the Grindstone Substation. This substation is approximately 2.4 kilometer (km) (1.5 miles [mi]) to the northwest of the RPF. The substation is 169 kilovolt (kV) that converts the current to 13,000 – 800 V for public distribution. The use of a shared substation will not affect the safe shutdown of the RPF.

8.1.5 Special Processing of Electrical Service

Details on special processing of the electrical service, such as isolation, transformers, noise limiters, lightning arresters, or constant voltage transformers, will be provided in the Operating License Application.

8.1.6 Design and Performance Specification

Design and performance specifications of principal and non-standard components will be provided in the Operating License Application.

8.1.7 Special Routing or Isolation

Special routing and isolation of wiring and circuits will be provided in the Operating License Application.

8.1.8 Deviations from National Codes

The RPF electrical system will be designed to meet all required national codes and standards, as described in Chapter 3.0.

8.1.9 Technical Specifications

As evaluated in Chapter 13.0, the RPF is designed to safely shut down without NEP for occupational safety and for protection of the public and environment. The NEP system will not require a technical specification per the guidelines in Chapter 14.0, "Technical Specifications."

8.2 EMERGENCY ELECTRICAL POWER SYSTEMS

Emergency electrical power is defined by NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content*, as any temporary substitute for normal electrical service. A combination of UPSs and the SEP system will provide emergency electrical power to the RPF, although only selected UPS systems will have a safety function. A 1,000 kW (1,341 hp) diesel generator will provide SEP.

Figure 8-1 also provides the electrical distribution topology for the SEP system. Power from this generator will service the RPF through an automatic transfer switch (ATS). The normal power side of the ATS will be connected to the RPF service entrance switchgear, with the load side of the ATS to be connected to the standby switchboard. The SEP system is designed to support the safety functions during RPF operations to protect workers, the public, and environment.

The SEP system design includes recommended practices from IEEE 446, *Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications*, NFPA 110, *Standard for Emergency and Standby Power Systems*, IEEE 379, and IEEE 493.

The SEP system will include overcurrent protective devices, surge arresters, fusing, relays, and similar safety-related protective devices. These safety devices will conform to the requirements of NFPA 70, relevant IEEE standards and recommendations, and local codes and standards.

SEP will be available to the exhaust system through a redundant electrical distribution topology. Approximately half of the exhaust electrical load requiring standby will be connected to an MCC, with the other half connected to a redundant MCC.

The standby switchboard will service equipment and devices in the hot cell, control room, exhaust system ventilation system, and other loads requiring standby power. Feeders, busing, overcurrent protection, devices, and equipment will provide the conveyance and conductor protection throughout the building.

During normal operations, loads connected to the standby switchboard will be serviced through the ATS with normal and facility electric power. In this way, any malfunctions of the SEP system during RPF operation with NEP will not interfere with normal RPF operations or prevent safe facility shutdown. When the ATS senses a loss of normal power, the switch will signal the on-site diesel generator to start up. When the diesel generator voltage and frequency are within acceptable limits, the ATS will switch from the normal power source to the diesel generator power source. Loads connected to the standby switchboard will continue to be serviced by the diesel generator until the normal power source returns. The ATS will sense the normal power source voltage and frequency. Once the voltage and frequency are within acceptable limits and after a prescribed delay, the ATS will switch from the diesel generator power source to the normal power source.

UPSs will be provided, as required. The function of the UPSs is to provide power to select loads while the diesel generator starts. The UPS systems will include unit devices, rack-mounted, and/or larger capacity cabinet units. The RPF loads requiring uninterruptable power on a short-term basis will be backed up by the on-site diesel generator to extend the duration of UPS power available to connected loads.

The 1,000 kW (1,341 hp) diesel generator will be serviced with a 3,785 liter (L) (1,000 gallons [gal]) diesel tank. This capacity will enable the generator to operate for 11–14 hr, depending on actual loads, without requiring additional fuel.

8.2.1 Design Basis of the Emergency Electric Power System

The emergency electrical power system design basis is to provide uninterrupted power to instrumentation, control, communication systems, and devices required to support the safety functions protecting workers and the public, and to provide sufficient electrical power to the RPF to ensure safe shutdown in the event of loss of NEP. The system design basis also provides SEP to operate select process-related equipment to limit the impacts of loss of NEP on RPF production operations. Additional information on the design basis is provided in Chapter 3.0.

8.2.2 Ranges of Emergency Electrical Power Required

The RPF power service is 480 V, 3-phase, 42 amp, 60 Hz. The total peak SEP for the RPF is 1,178.6 kW (1,585 hp). Table 8-1 lists the backup peak electrical power loads for different locations and processes within the RPF.

8.2.3 Power for Safety-Related Instruments

Safety-related instrumentation will be provided with UPSs. The UPSs will provide power to safety-related instruments while the diesel generator starts and will provide service loads requiring uninterruptable power on a short-term basis. The diesel generator will maintain power until the normal power system is operating within acceptable limits.

8.2.4 Power for Effluent, Process, and Area Radiation Monitors

Effluent, process, and area radiation monitors will be provided with the UPSs. The UPSs will provide service loads requiring uninterruptable power for up to 120 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits.

8.2.5 Power for Physical Security Control, Information, and Communication Systems

Physical security control, information, and communication systems will be provided with a UPS. The UPS provides service loads requiring uninterruptable power for up to 120 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits.

8.2.6 Power to Maintain Experimental Equipment in Safe Condition

There are no experimental equipment or facilities in the RPF.

8.2.7 Power for Active Confinement/Containment Engineered Safety Feature Equipment and Control Systems

Based on the analysis in Chapter 13.0, the Zone I exhaust ventilation subsystems operations, equipment, and components ensures the confinement of hazardous materials during normal and abnormal conditions, including natural phenomena, fires, and explosions. After a loss of NEP, the Zone I exhaust ventilation subsystem will automatically place itself into the passive mode, including inlet bubble-tight isolation dampers that close to provide passive confinement.

The system will remain in this configuration until the voltage and frequency of power from the diesel generator are within acceptable limits. At that point, the system can be manually started and operated in a reduced ventilation mode with one operating group of HVAC fans and components. The Zone I exhaust ventilation subsystems are designed to function in a manner, whether operational or not, consistent with occupational safety and protection of workers, the public, and environment. Therefore, this system is not considered an IROFS.

8.2.8 Power for Coolant Pumps or Systems

Based on the analysis provided in Chapter 5.0, "Coolant Systems," the coolant system is designed to function in a manner, whether operational or not, consistent with occupational safety and protection of the public and the environment. Therefore, power to coolant systems is not considered an IROFS.

8.2.9 Power for Emergency Cooling

Based on the analysis provided in Chapter 5.0, an emergency cooling water system is not required.

8.2.10 Power for Engineered Safety Feature Equipment

Engineered safety features requiring power will be provided with UPSs. The UPSs will provide service loads requiring uninterruptable power for up to 120 min. The diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will be provided in the Operating License Application.

8.2.11 Power for Emergency Lighting

Power required for emergency lighting will be provided by UPSs. The UPSs will provide service loads requiring uninterruptable power for up to 120 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will be provided in the Operating License Application.

8.2.12 Power for Instrumentation and Control Systems to Monitor Shutdown

Power for instrumentation and control systems used to monitor safe shutdown will be provided with UPSs. The UPSs will provide service loads requiring uninterruptable power for up to 120 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will be provided in the Operating License Application.

8.2.13 Technical Specifications

As evaluated in Chapter 13.0, the RPF is designed to safely shut down without SEP consistent with occupational safety and protection of the public and the environment. The UPS systems, as required, are anticipated to be part of the technical specification for the system being supported. The SEP system will not require a technical specification per the guidelines in Chapter 14.0.

8.3 REFERENCES

IEEE 379, *Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*, Institute of Electrical and Electronics Engineers, Piscataway, New Jersey, 2014.

IEEE 446, *Recommended Practice for Emergency and Standby Power Systems for Industrial and Commercial Applications*, Institute of Electrical and Electronics Engineers, Piscataway, New Jersey, 2014.

IEEE 493, *Recommended Practice for the Design of Reliable Industrial and Commercial Power Systems (Gold Book)*, Institute of Electrical and Electronics Engineers, Piscataway, New Jersey, 2007.

NFPA 70, *National Electrical Code (NEC)*, National Fire Protection Association, Quincy, Massachusetts, 2014.

NFPA 110, *Standard for Emergency and Standby Power Systems*, Institute of Electrical and Electronics Engineers, Piscataway, New Jersey, 2014.

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.

This page intentionally left blank.

Chapter 9.0 – Auxiliary Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1
June 2017

Prepared by:
Northwest Medical Isotopes, LLC
815 NW 9th Ave, Suite 256
Corvallis, OR 97330

This page intentionally left blank.

Chapter 9.0 – Auxiliary Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1

Date Published:
June 26, 2017

Document Number: **NWMI-2013-021**	*Revision Number*: **1**
Title: **Chapter 9.0 – Auxiliary Systems Construction Permit Application for Radioisotope Production Facility**	
Approved by: **Carolyn Haass**	*Signature*: Carolyn C Haass

This page intentionally left blank.

REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required
1	6/26/2017	Incorporate changes based on responses to NRC Requests for Additional Information	C. Haass

This page intentionally left blank.

CONTENTS

9.0 RADIOISOTOPE PRODUCTION FACILITY AUXILIARY SYSTEMS ... 9-1
9.1 Heating Ventilation and Air Conditioning Systems ... 9-1
9.1.1 Design Basis ... 9-2
9.1.2 System Description ... 9-2
9.1.2.1 Confinement ... 9-6
9.1.2.2 Supply Air System ... 9-7
9.1.2.3 Exhaust Air System ... 9-10
9.1.2.4 Cleanroom Subsystem ... 9-13
9.1.2.5 Physical Layout and Location ... 9-14
9.1.2.6 Principles of Operation ... 9-14
9.1.3 Operational Analysis and Safety Function ... 9-15
9.1.4 Instrumentation and Control Requirements ... 9-16
9.1.5 Required Technical Specifications ... 9-16
9.2 Material Handling ... 9-17
9.3 Fire Protection Systems and Programs ... 9-18
9.3.1 Design Basis ... 9-18
9.3.2 System Description ... 9-18
9.3.2.1 Fire Suppression Subsystem ... 9-19
9.3.2.2 Fire Detection and Alarm Subsystem ... 9-20
9.3.2.3 Fire Extinguishers ... 9-21
9.3.3 Operational Analysis and Safety Function ... 9-21
9.3.3.1 Radioisotope Production Facility Fire Areas ... 9-22
9.3.3.2 Other Radioisotope Production Facility Systems ... 9-36
9.3.3.3 Architectural Features ... 9-36
9.3.4 Instrumentation and Control Requirements ... 9-37
9.3.5 Required Technical Specifications ... 9-38
9.4 Communication Systems ... 9-39
9.4.1 Design Basis ... 9-39
9.4.2 System Description ... 9-39
9.4.3 Operational Analysis and Safety Function ... 9-39
9.4.4 Instrumentation and Control Requirements ... 9-40
9.4.5 Required Technical Specifications ... 9-40
9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material ... 9-41
9.5.1 Design Basis ... 9-41
9.5.2 System Description ... 9-41
9.5.2.1 Special Nuclear Materials ... 9-41
9.5.2.2 Byproduct Materials ... 9-42
9.5.3 Operational Analysis and Safety Function ... 9-42
9.5.4 Instrumentation and Control Requirements ... 9-42
9.5.5 Required Technical Specifications ... 9-42
9.6 Cover Gas Control in Closed Primary Coolant Systems ... 9-43
9.6.1 Design Basis ... 9-43
9.6.2 System Description ... 9-43
9.6.3 Operational Analysis and Safety Function ... 9-43
9.6.4 Instrumentation and Control Requirements ... 9-43
9.6.5 Required Technical Specifications ... 9-43

9.7 Other Auxiliary Systems ... 9-44
9.7.1 Utility Systems ... 9-44
9.7.1.1 Design Basis ... 9-44
9.7.1.2 System Description ... 9-44
9.7.1.3 Operational Analysis and Safety Function ... 9-61
9.7.1.4 Instrumentation and Control Requirements ... 9-61
9.7.1.5 Required Technical Specifications ... 9-61
9.7.2 Control and Storage of Radioactive Waste ... 9-62
9.7.2.1 Design Basis ... 9-62
9.7.2.2 System Description ... 9-62
9.7.2.3 Operational Analysis and Safety Function ... 9-76
9.7.2.4 Instrumentation and Control Requirements ... 9-76
9.7.2.5 Required Technical Specifications ... 9-76
9.7.3 Analytical Laboratory ... 9-77
9.7.3.1 Design Basis ... 9-77
9.7.3.2 System Description ... 9-77
9.7.3.3 Operational Analysis and Safety Function ... 9-78
9.7.3.4 Instrumentation and Control Requirements ... 9-79
9.7.3.5 Required Technical Specifications ... 9-79
9.7.4 Chemical Supply ... 9-79
9.7.4.1 Design Basis ... 9-79
9.7.4.2 System Description ... 9-79
9.7.4.3 Operational Analysis and Safety Function ... 9-86
9.7.4.4 Instrumentation and Control Requirements ... 9-86
9.7.4.5 Required Technical Specifications ... 9-86
9.8 References ... 9-87

FIGURES

Figure 9-1. Ground Level Confinement 9-4
Figure 9-2. Upper Level Confinement 9-5
Figure 9-3. Lower Level Confinement 9-6
Figure 9-4. Facility Ventilation System Diagram 1 9-8
Figure 9-5. Facility Ventilation System Diagram 2 9-9
Figure 9-6. Process Flow Diagram for Process Vessel Ventilation Treatment 9-12
Figure 9-7. Life Safety Plan (First Floor) 9-23
Figure 9-8. Life Safety Plan (Second Floor) 9-24
Figure 9-9. Second Floor Mechanical Utility Area 9-45
Figure 9-10. Medium-Pressure Steam System 9-46
Figure 9-11. Low-Pressure Steam System 9-47
Figure 9-12. Chilled Water System Large Geometry Hot Cell Loop 9-50
Figure 9-13. Chilled Water System Critically Safe Hot Cell Loop 9-51
Figure 9-14. Chilled Water System Target Fabrication Loop 9-52
Figure 9-15. Process Chilled Water System 9-53
Figure 9-16. Demineralized Water System 9-56
Figure 9-17. Plant Air System 9-57
Figure 9-18. Nitrogen and Helium Supply System 9-58
Figure 9-19. Hydrogen and Oxygen Supply System 9-59
Figure 9-20. Waste Management Process Flow Diagram and Process Flow Streams 9-64
Figure 9-21. High-Dose Liquid Waste Solidification Subsystem and Low-Dose Collection Tank Location 9-65
Figure 9-22. Simplified High-Dose Waste Handling Process Flow Diagram 9-66
Figure 9-23. High-Dose Waste Treatment and Handling Equipment Arrangement 9-67
Figure 9-24. Low-Dose Liquid Waste Evaporation System Location 9-68
Figure 9-25. Low-Dose Liquid Waste Disposition Process 9-69
Figure 9-26. Low-Dose Liquid Waste Solidification Equipment Arrangement 9-70
Figure 9-27. Spent Resin Dewatering Operational Flow Diagram 9-71
Figure 9-28. Spent Resin Collection Tanks Location 9-71
Figure 9-29. Solid Waste Encapsulation Operational Flow Diagram 9-72
Figure 9-30. High-Dose Waste Decay Operational Flow Diagram 9-72
Figure 9-31. High Dose Waste Decay Cell Equipment Arrangement 9-73
Figure 9-32. High Dose Waste Handling Operational Flow Diagram 9-73
Figure 9-33. Waste Handling Flow Diagram 9-74
Figure 9-34. Waste Handling Equipment Arrangement 9-75
Figure 9-35. Analytical Laboratory Layout 9-78
Figure 9-36. Chemical Supply Room Equipment Layout 9-80
Figure 9-37. Nitric Acid Flow Diagram 9-81
Figure 9-38. Sodium Hydroxide Flow Diagram 9-83
Figure 9-39. Hydrogen Peroxide Flow Diagram 9-84

TABLES

Table 9-1. Facility Areas and Respective Confinement Zones 9-3
Table 9-2. Indications for Facility Ventilation System Parameters 9-16
Table 9-3. Purge Gas Flows 9-60
Table 9-4. Tanks Requiring Purge Gas 9-60
Table 9-5. High-Dose Waste Tank Capacities 9-66
Table 9-6. Low-Dose Waste Tank Capacities 9-69
Table 9-7. Subsystem 100, Nitric Acid Tank Sizes 9-82
Table 9-8. Subsystem 200, Sodium Hydroxide Tank Sizes 9-83
Table 9-9. Subsystem 300, Reductant and Nitrogen Oxide Absorber Solutions Tank Sizes 9-84
Table 9-10. Subsystem 400, Hydrogen Peroxide Tank Sizes 9-84
Table 9-11. Subsystem 600, Fresh Uranium Ion Exchange Resin Tank Sizes 9-85

TERMS

Acronyms and Abbreviations

^{89}Sr	strontium-89
^{90}Sr	strontium-90
^{99}Mo	molybdenum-99
^{230}Th	thorium-230
^{231}Pa	protactinium-231
^{233}Pa	protactinium-233
^{233}U	uranium-233
^{235}U	uranium-235
^{237}Np	neptunium-237
^{238}Pu	plutonium-238
^{238}U	uranium-238
^{239}Pu	plutonium-239
^{240}Pu	plutonium-240
^{241}Am	americium-241
ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
DBF	design basis fire
DOT	U.S. Department of Transportation
H_2	hydrogen gas
HEGA	high-efficiency gas adsorption
HEPA	high-efficiency particulate air
HIC	high-integrity container
HNO_3	nitric acid
HVAC	heating, ventilation, and air conditioning
IBC	International Building Code
ICP-MS	inductively coupled plasma mass spectrometry
IROFS	item relied on for safety
IRU	iodine removal unit
IX	ion exchange
Kr	krypton
LAN	local area network
LEU	low-enriched uranium
Mo	molybdenum
MURR	University of Missouri Research Reactor
NaOH	sodium hydroxide
NESHAP	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NO_x	nitrogen oxide
NRC	U.S. Nuclear Regulatory Commission
NWMI	Northwest Medical Isotopes, LLC
OSTR	Oregon State University TRIGA Reactor
OSU	Oregon State University
PFHA	preliminary fire hazards analysis
RCA	radiologically controlled area
RPF	Radioisotope Production Facility
SNM	special nuclear material
TiO_2	titanium dioxide
U	uranium

U.S.	United States
U.S.C.	United States Code
VoIP	Voice over Internet Protocol
Xe	xenon

Units

°C	degrees Celsius
°F	degrees Fahrenheit
μ	micron
cm	centimeter
cm^2	square centimeter
ft	feet
ft^2	square feet
ft^3	cubic feet
gal	gallon
gmol	gram-mol
hr	hour
in.	inch
$in.^2$	square inch
kg	kilogram
L	liter
lb	pound
m	meter
M	molar
m^2	square meter
m^3	cubic meter
min	minute
mm	millimeter
W	watt
wt%	weight percent

9.0 RADIOISOTOPE PRODUCTION FACILITY AUXILIARY SYSTEMS

This chapter provides the descriptions of the auxiliary systems for the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) that have not been addressed in previous chapters. These auxiliary systems are important to the safe operation of the RPF and to protect the health and safety of workers, the public, and environment. The chapter is organized in accordance with NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content*, as augmented by the *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* (NRC, 2012).

The RPF auxiliary systems include the following:

- Heating and ventilation, and air conditioning (HVAC) systems
- Fire protection systems
- Communication systems
- Possession and use of byproduct, source, and special nuclear material
- Cover gas control in the closed primary coolant system
- Other auxiliary systems, including utility systems, control and storage of radioactive waste, analytical laboratory, and chemical supply

For each auxiliary system, a description is provided of the system's capability to function as designed without compromising RPF operations and to shut down the RPF during normal operations or under RPF accident conditions. Each auxiliary system description includes:

- Design basis
- System description
- Operational analysis and safety function
- Instrumentation and control requirements
- Required technical specifications and their bases, including testing and surveillance

9.1 HEATING VENTILATION AND AIR CONDITIONING SYSTEMS

The RPF HVAC system, also referred to as the facility ventilation system, is designed to ensure that temperature, relative humidity, and air exchange rates are within the design-basis limits for personnel and equipment and to ensure that all normal sources of airborne radioactive material are controlled so that occupational doses do not exceed the requirements of Title 10, *Code of Federal Regulations*, Part 20, "Standards for Protection Against Radiation" (10 CFR 20). The system design is consistent with NWMI's as low as reasonably achievable (ALARA) program.

The RPF design features ensure that airflow and relative pressure will prevent inadvertent diffusion or other uncontrolled release of airborne radioactive material from the RPF. The facility is also designed and operated to ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment can occur.

The analyses of system operations show that planned releases of airborne radioactive material to the unrestricted environment will not expose the public to doses that exceed the limits of 10 CFR 20 and the NWMI ALARA program. NWMI's ALARA program is discussed in Chapter 11.0, "Radiation Protection Program and Waste Management," and a detailed airborne exposure analysis is provided in Chapter 11, Section 11.1.1.1.2.

9.1.1 Design Basis

The facility ventilation system is designed to provide confinement of hazardous chemical fumes and airborne radiological materials and conditioning of the RPF environment for facility personnel and equipment. The design basis of the facility ventilation system and the process vessel ventilation system is provided in Chapter 3.0, "Design of Structures, Systems, and Components," Section 3.5.7.2; and the safety functions are provided in Chapter 6.0, "Engineered Safety Features," Section 6.2.1.1.

9.1.2 System Description

The facility ventilation system will maintain a series of cascading pressure zones to draw air from the cleanest areas of the facility to the most contaminated areas. Zone IV will be a clean zone that is independent of the other ventilation zones. Zone IV will be slightly positively pressurized with respect to the atmosphere. Zone III will be the cleanest of the potentially contaminated areas, with each subsequent zone being more contaminated and having lower pressures.

A common supply air system will provide 100 percent outdoor air to all Zone III areas and some Zone II areas that require makeup air in addition to that cascaded from Zone III. Three separate exhaust systems will maintain zone pressure differentials and containment: (1) the Zone I exhaust system will service the hot cell, waste loading areas, target fabrication enclosures, and process vessel ventilation subsystems in Zone I; (2) the Zone II/III exhaust system will service exhaust flow needs from Zone II and Zone III in excess of flow cascaded to interior zones; and (3) a laboratory exhaust system will service fume hoods in the laboratory area.

Supply air will be conditioned using filters, heater coils, and cooling coils to meet the requirements of each space. Abatement technologies, primarily high-efficiency particulate air (HEPA) filtration and activated carbon, will be used to ensure that air exhausted to the atmosphere meets 40 CFR 61, "National Emission Standards for Hazardous Air Pollutants" (NESHAP), and applicable State law.

A stack monitoring and sampling system will be employed to demonstrate compliance with the stated regulatory requirements for exhaust.

The RPF ventilation system will include the air supply, process ventilation, and exhaust air systems and associated filters, fans, dampers, ducts, and control instrumentation. The supply air system will draw in and condition fresh air and distribute it throughout the facility. A portion of the supply air will enter the process ventilation system through fume hoods, open-front enclosures, gloveboxes, and hot cells, and will be removed with other exhaust air systems through the stacks to the environment after being treated.

The safety functions of the ventilation systems will serve to protect workers, the public, and environment by maintaining confinement barriers in a multiple confinement barrier system.

The RPF will typically be ventilated such that airflows travel from areas of lower potential for contamination to areas of higher potential. The ventilation system functions will include temperature and air quality control to meet production and worker needs.

The RPF building ventilation system will have four confinement zone designations, with airflow directed from lowest to highest potential for contamination: Zone I, Zone II, Zone III, and Zone IV. Figure 9-1 through Figure 9-3 show the facility confinement boundaries on the ground level (first level), upper level (second level), and lower level (basement), respectively. Table 9-1 defines the confinement zone applicable to major spaces. The zones are defined as follows:

- **Zone I**, shown in pink, is the initial confinement barrier and includes gloveboxes, vessels, tanks, piping, hot cells, and the Zone I exhaust subsystem.
- **Zone II**, shown in orange, is the secondary confinement subsystem and includes the walls, floors, ceilings, and doors of the laboratories with the gloveboxes, HEPA filter rooms, and the Zone II ventilation exhaust subsystem. Laboratory gloveboxes and fume hoods are also Zone II.
- **Zone III**, shown in green, is the tertiary confinement barrier and includes the walls floor, ceilings, and doors of the corridor that surround the operating galleries, and the mechanical mezzanine.
- **Zone IV**, shown in blue, is the non-confinement ventilation zone – the positively pressurized areas served by unitary, non-safety, and commercial-grade equipment. These areas will include the administration support area, truck bays, and maintenance utility areas.

Table 9-1. Facility Areas and Respective Confinement Zones

Area	Zone
Hot cells (production)	I
Tank hot cell	I
Solid waste treatment hot cell	I
High dose waste solidification hot cell	I
Uranium decay and accountability hot cell	I
HIC vault	I
Analytical laboratory gloveboxes	I
R&D hot cell laboratory hot cells	I
Target fabrication room and enclosures	II
Utility room	II
Analytical laboratory room and hoods	II
R&D hot cell laboratory room and hoods	II
Waste loading hot cell	II
Maintenance gallery	II
Manipulator maintenance room	II
Exhaust filter room	II
Airlocks[a]	II, III
Irradiated target basket receipt bay	III
Waste loading truck bay	III
Operating gallery and corridor	III
Electrical/mechanical supply room	III
Chemical supply room	III
Corridors	III
Decontamination room	III
Loading docks	IV
Waste management loading bay	IV
Irradiated target receipt truck bay	IV
Maintenance room	IV
Support staff areas	IV

[a] Confinement zone of airlocks will be dependent on the two adjacent zones being connected.

HIC = high integrity container.
R&D = research and development.

[Proprietary Information]

Figure 9-1. Ground Level Confinement

[Proprietary Information]

Figure 9-2. Upper Level Confinement

[Proprietary Information]

Figure 9-3. Lower Level Confinement

9.1.2.1 Confinement

Confinement is an engineered safety feature of the HVAC system. Confinement is the term used to describe the boundary that surrounds radioactive materials and the associated ventilation system. Confinement systems are designed to localize any release of radioactive material to controlled areas in normal operational states and to mitigate the consequences of design basis accidents. Radiation protection control features (e.g., adequate shielding and confinement ventilation systems) minimize hazards associated with radioactive materials. The principal design and safety objective of the confinement system is to protect on-site personnel and the off-site public. The second design objective is to minimize the reliance on administrative or complex active engineering controls to provide a confinement system as simple and fail-safe as reasonably possible.

The process vessel ventilation system will serve as the primary confinement pressure boundary and is safety-related. The Zone I exhaust subsystem is an engineered safety feature that (along with shielding) will create a secondary confinement boundary; enclosing the vessels and process offgas within the hot cells. Confinement of the hot cells will be achieved through both the confinement ventilation system and the shielding provided by the steel and concrete structures comprising the walls, roofs, penetrations, and covers of the cells.

Secondary confinement will be accomplished by the zone boundaries, associated ventilation systems, and HEPA filter plenums to filter exhaust air prior to discharge at the facility ventilation stacks. Secondary confinement will also be accomplished through the use of bubble-tight isolation dampers. These dampers will isolate the ducts at the zone boundary under certain scenarios to ensure that all potential releases have been HEPA-filtered prior to exiting the facility (i.e., release to atmosphere). The safety aspects of the confinement system are discussed in Chapter 6.0, "Engineered Safety Features," Section 6.1, including the design response to off-normal conditions (e.g., loss of power).

9.1.2.2 Supply Air System

The RPF supply air system will provide conditioned air for facility workers and equipment and supply makeup air for RPF exhaust air systems. The supply air system will provide filtered and conditioned air to all Zone III spaces and some Zone II spaces at a ventilation rate of 100 percent outside air. The three air supply handling units will be sized at 50 percent capacity each, for redundancy. Two of the three units will be operating, while the third is on standby. If a single unit fails, the standby unit will start automatically. Each unit will consist of an outdoor air louver, filters, cooling coil, heating coil, heat recovery coil, isolation dampers, and a fan.

Variable-speed fans will be modulated to control the pressure in the common air plenum. The heating and cooling coils in each air-handling unit will be controlled based on a common supply air temperature sensor. Reheat coils will be provided in the supply ducts to each space, as required, to further condition the supply air, based on space temperature thermostats.

Outside air will be drawn into the RPF air supply system through air-handling units (Figure 9-4). The units will normally supply a constant volume of conditioned air to the Zone II and Zone III areas of the RPF.

Zone III air will be cascaded into Zone II areas through engineered leakage pathways by a negative pressure differential, maintaining the desired pressure drop between the zones (Figure 9-4). Terminal unit components in the supply duct system will include airflow control valves and reheat coils. The terminal reheat coils will provide final tempering of the supply air to maintain the Zone II space temperature setpoint. Zone II supply airflow control valves will operate in conjunction with exhaust valves to control the pressure differential in each zone by maintaining a fixed difference between the total supply and exhaust air flows for each Zone II space. Exhaust from Zone II will be expelled through the 23 meter (m) (75-foot [ft]) high Zone II exhaust stack. Additional detailed information on the Zone II stack design will be developed for the Operating License Application.

The isolation dampers and backdraft dampers in the supply duct system at the zone boundary (Figure 9-5) will close when required to provide confinement at the zone boundary. The supply air system HVAC controls will operate through the building management system.

[Proprietary Information]

Figure 9-4. Facility Ventilation System Diagram 1

[Proprietary Information]

Figure 9-5. Facility Ventilation System Diagram 2

9.1.2.3 Exhaust Air System

The RPF will have four exhaust air subsystems: Zone I exhaust, Zone II/III exhaust, laboratory exhaust, and process vessel ventilation exhaust. Each exhaust system will be provided with two 100 percent capacity exhaust fans and filter trains for complete redundancy on all exhaust subsystems. This redundancy is important to ensure confinement ventilation pressure differentials are maintained at all times. [Proprietary Information]. Exhaust ducts upstream of the filter trains will be round to minimize areas where contamination can accumulate, and are sized to minimize particulate settling in the duct. Each exhaust system will have a separate stack, with the exception of the process vessel ventilation subsystem, which will merge with the Zone I exhaust stream. A stack monitoring and sampling system will be provided on each stack to demonstrate compliance with applicable State law.

9.1.2.3.1 Zone I Exhaust System

The Zone I exhaust system will serve the hot cell, high-integrity container (HIC) loading area, and solid waste loading area. This exhaust system will maintain Zone I spaces at negative pressure with respect to atmosphere. The disassembly hot cell station will be maintained at a slightly lower pressure due to the increased likelihood of contamination in that area. All makeup air to Zone I spaces will be cascaded from Zone II spaces. Space temperature control will not be provided for Zone I spaces unless thermal loads are expected to cause temperatures to exceed equipment operating ranges without additional cooling. HEPA filters will be included on both the inlet and outlet ducts to Zone I. The outlet HEPA filters will minimize the spread of contamination from the hot cell into the ductwork leading to the exhaust filter train. The inlet HEPA filters will prevent contamination spread in case of an upset condition that results in positive pressurization of Zone I spaces with respect to Zone II spaces. The process vessel ventilation subsystem will enter the Zone I exhaust subsystem just upstream of the filter train.

The Zone I exhaust system will expel air from the hot cells and glovebox enclosures located within the RPF. The system will also capture exhaust from the process vessel ventilation system. The Zone I hot cell and glovebox enclosure will draw ventilation air from the surrounding Zone II spaces through HEPA filters. The exhaust air from each cell will pass through local HEPA filters.

Negative space pressure in Zone I will be controlled through local exhaust airflow control valves for each cell. The exhaust from the cells will collect in a Zone I system duct header and then be drawn through final, testable, HEPA filters and carbon adsorbers prior to discharge into the exhaust stack. The speed of the Zone I exhaust fans will be controlled to maintain a negative pressure setpoint in the Zone I exhaust duct header. The exhaust fans will be fully redundant. If the operating fan fails, the standby fan will start automatically. Exhaust from Zone I will be expelled through the 23 m (75-ft) high Zone I exhaust stack. Detailed information on the Zone I stack design will be developed for the Operating License Application.

9.1.2.3.2 Zone II/III Exhaust System

The Zone II/III exhaust system will serve the Zone II spaces and those Zone III spaces that do not provide cascaded air flow into Zone II. This exhaust system will maintain Zone II spaces at negative pressure and Zone III spaces at a less negative pressure with respect to atmosphere. Makeup air to Zone II spaces will either be cascaded from Zone III spaces or supplied from the supply air subsystem to meet additional space conditioning needs. All makeup air to Zone III spaces will be provided from the supply air subsystem.

The RPF Zone II exhaust system will expel air from the operating areas, workrooms, and fume hoods to maintain confinement. This confinement is important to safety to protect facility workers from radiological and hazardous chemical releases. The exhaust air from these spaces will collect in a Zone II exhaust header and will then be drawn through final, testable, HEPA filters and carbon adsorbers prior to discharge into the exhaust stack (Figure 9-4). The exhaust fan speed will be controlled to maintain the desired negative pressure in the RPF Zone II exhaust header. The exhaust fans will be fully redundant. If the operating fan fails, the standby fan will start automatically.

Air flow control valves in the Zone II room exhaust duct system will operate in conjunction with the zone supply valves to produce an offset between the exhaust and supply flow rates. The flow offset will enable a negative space pressure. Flow control valves in the fume hood exhaust ducts will maintain a constant volume through each fume hood. The control valves will automatically modulate to compensate for a drop in air pressure due to loading of local filters.

9.1.2.3.3 Laboratory Exhaust System

The laboratory exhaust system will provide fume hood and glovebox exhaust capability. This essentially is a Zone I system, but is separate from the main Zone I exhaust system to accommodate the large flow fluctuations from changing fume hood positions. These highly variable flow conditions will be controlled better through a separate exhaust system. This exhaust system will minimize the potential pressure perturbations and control difficulties that could result from including the fume hoods on the main Zone I exhaust system. Makeup air for increased fume hood exhaust flow will be supplied from the common supply air system.

9.1.2.3.4 Process Vessel Ventilation Treatment System

Due to the relatively short timeframe from neutron fission operations at a reactor to target dissolution and processing in the RPF, there will be an amount of short-lived tellurium isotopes in some process streams. The decay of these tellurium isotopes will create iodine isotopes. While most of these process streams will not likely evolve any iodine species into the offgas, this event cannot be precluded. To ensure the safety of the facility, the offgas from these special process streams will be collected and routed to an iodine removal system. Figure 9-6 provides a flow diagram for the process vessel vent subsystems that flow to the process vessel vent iodine removal unit (IRU).

The locations that are routed to the iodine removal subsystem include the following:

- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

[Proprietary Information]

Figure 9-6. Process Flow Diagram for Process Vessel Ventilation Treatment

Iodine removal unit for target dissolution offgas system – This system in the tank hot cell will include offgas from the target disassembly and the target dissolution offgas systems.

- **Target disassembly** – [Proprietary Information].
- **Target dissolution** – [Proprietary Information].

After the offgas filter will be the dissolver offgas system's vacuum pumps and tanks, then the stream will flow through the secondary fission gas adsorbers and into the process vessel vent header.

Iodine removal unit for uranium, molybdenum, and waste accumulation tanks –Some of the liquids in the hot cell will contain tellurium isotopes that generate iodine isotopes during decay. A portion of the iodine will remain in the dissolver solution. Although it is not likely that much of the iodine will evolve into the offgas, these streams will be passed through an IRU before the process vessel vent header.

The expected offgas streams that feed this IRU will be from tank hot cell vessels, including the Mo feed tanks, impure U collection and lag storage tanks, U recovery waste tanks, and the liquid waste handling tanks. [Proprietary Information]. This offgas stream will flow into the process vessel vent header.

General vessel vent system – This header system will service the remaining vessels in the tank hot cell, including the pure U lag storage tanks (14), recycled U collection tank, and tanks attributed to the U concentrators. This offgas stream will flow into the process vessel vent header without additional treatment.

High volume evaporative vent from waste handling – This system will service the three waste solidification unit operations (low-dose liquid waste, high-dose liquid waste, and solid waste) and the low-dose evaporation tanks.

The low-dose evaporation tanks will have high flowrate and elevated temperatures to allow water to evaporate. The header will collect these humid air sweeps and dilute with additional air bleed to ensure that the evaporated water does not condense in the ducting or pipes. This offgas stream will flow into the process vessel vent header.

Target fabrication vent – The target fabrication area ventilation is required for confining: (1) offgas from the dissolver and other process vessel, and (2) offgas from the calcination or reduction furnace systems, where hydrogen will be diluted with air to less than the lower flammability limit. This offgas stream will flow into the process vessel vent header.

Process vessel vent iodine removal unit – The process vessel vent IRU (VV-SB-520) system will consist of a sorbent bed of charcoal or activated carbon to remove iodine from the vessel vent gases. The process vessel vent IRU is part of an item relied on for safety (IROFS) RS-03, "Hot Cell Secondary Confinement Boundary." Chapter 6.0, Section 6.2.1, and Chapter 13.0, "Accident Analysis," Section 13.2.2.8, provide additional detail on the safety function.

Process vessel vent filter – This treatment operation will consist of HEPA filtration and the exhaust fan and will flow to the Zone I exhaust system.

9.1.2.4 Cleanroom Subsystem

The Mo purification hot cell cleanroom subsystem is designed to provide filtered and conditioned air at an exchange rate to meet the standards of an ISO 14644-1, "Cleanrooms and Associated Controlled Environments—Part 1: Classification of Air Cleanliness," Class 8 cleanroom. The cleanroom will be maintained at a slightly positive pressure relative to its surroundings to ensure that unfiltered air does not infiltrate the cleanroom. Air inside the cleanroom will be continually recirculated through a dedicated filtration system to remove internally generated contaminants. Air will be 100 percent recirculated, with the only air exchange with the surroundings of the cleanroom occurring through exfiltration and makeup air entering on the suction side of the fan. The cleanroom air handling unit and filters will be located inside the hot cell and, therefore, must be remotely maintainable. Periodic cleanroom certification testing will also need to be performed remotely with permanently installed instrumentation.

9.1.2.5 Physical Layout and Location

All supply air handling units, supply fans, exhaust fans, and associated heat recovery coils will be located in the mechanical/electrical area (supply air handler room) located on the second floor. This area will house the Zone II and Zone III subsystem air-handling units and fans. The exhaust HEPA filter plenums and exhaust fans will be located in the mechanical area on the second floor.

9.1.2.6 Principles of Operation

The RPF ventilation system will maintain the facility at the desired temperatures and negative pressurization during normal operations. Supply air temperature from the air-handling units will be held constant through the use of heating and cooling coils. Reheat coils will be provided to further temper supply air to occupied areas based on local thermostat demand. The systems also have design features to maintain constant overall building pressures, the Zone I header pressure, and Zone II exhaust header pressures during normal operations. Local room pressurization will be obtained by the airflow offset between supply and exhaust.

Supply airflows will be held constant through the use of supply fan variable-frequency drives and flow measuring stations. Exhaust airflow will be controlled based on building pressure and exhaust header pressure demands and to ensure that the HEPA filter plenum rated airflows are not exceeded. Variable-frequency drives on the exhaust fans will be provided to maintain required exhaust flows when flow resistance resulting from exhaust filter dirt loading increases.

Makeup air to maintain a constant air pressure differential between the Zone II and Zone III areas will be provided by the Zone III supply air. Zone III will provide overall building pressure control during normal operations by modulation of the exhaust/return airflow path, while the supply air remains fixed.

Pressure and flow conditions for the process enclosures and laboratory ventilation will be manually controlled using volume dampers and valves. Airflow control valves will be installed in each room's main supply and exhaust ducts to maintain laboratory design space pressure. These valves will be located outside of the laboratory modules.

The Zone I exhaust system for each module will be adjusted manually using a valve located in the room duct header near the air inlet end to maintain minimum vacuum pressure. A static pressure tap will be located near the air inlet end of the header and will be attached to a magnehelic gauge to monitor the header pressure relative to the laboratory module space pressure (on the radiologically controlled area [RCA]-designed portion of the system). The system is designed to maintain the Zone I process enclosures at their design pressure during normal operations and have the capacity to draw the required inflow of air in the event of a design breach of an enclosure.

The Zone II exhaust system is designed to maintain the Zone II enclosures at their required pressure. A balancing valve located in the exhaust duct of each enclosure will initially be partially closed. As the local filter of the enclosure loads up and a drop in pressure increases across the filter, the valve will be adjusted to reestablish flow in the design range. Differential pressure gauges will be provided at each enclosure to monitor the filter pressure drop and measure the pressure drop across only the enclosure. The enclosure's pressure drop reading will be calibrated to its acceptable face velocity range to monitor enclosure performance.

The Zone II supply air system is designed to provide the supply air volume rate required for each space. The system will supply makeup air as required for the Zone I and II process enclosures, general exhaust, and to maintain the design temperatures in the laboratories. [Proprietary Information] to prevent the entrainment of potentially contaminated air back out of the process enclosures.

9.1.3 Operational Analysis and Safety Function

Chapter 11.0 and Chapter 13.0 provide an analysis of normal and off-normal operation of the RPF HVAC system. Chapter 11.0, Section 11.1.1.1 presents that normal release analysis. Chapter 13.0, Section 13.2 evaluates various accident sequences that involve failure of the ventilation components, radiological spills, and the release of high-dose solutions, vapors, or gases from within the hot cell liquid confinement, secondary confinement, or shielding boundary.

Defense-in-depth – Failure of the air balance system is not in itself an accident, but represents the failure of a system designed to mitigate other accidents that lead to an airborne release of radionuclides in the form of particulates or gases. Systems that will mitigate these releases include the primary confinement and primary offgas treatment system, which will capture particulates, absorb iodine, and absorb Xe and Kr and other gaseous radionuclides, to slow the release following decay to more stable isotopes. In the target fabrication processes, uranium will be handled in physical forms that do not contribute to a high-dose rate factor in airborne releases. Uranium solutions will also be processed in closed systems with filtered process ventilation systems to remove the small amounts of activity normally released.

Item relied on for safety – Based on the Chapter 13.0 analysis, the hot cell secondary confinement (Zone I exhaust ventilation subsystem) has been designated as an IROFS (RS-03, "Hot Cell Secondary Confinement Boundary"). The operations, equipment, and components of this system will ensure the confinement of hazardous materials during normal and abnormal conditions, including natural phenomena, fires, and explosions. Components of the dissolver offgas subsystem and the process vessel ventilation system have also been designated as IROFS. The safety functions of the confinement system are discussed in more detail in Chapter 6.0, Section 6.1.

Chapter 13.0 evaluates a fire that could cause the carbon retention beds to ignite, leading to the release of radionuclides into the RPF exhaust stack. Based on analysis of this accident, the exhaust stack height was identified as an IROFS (FS-05, "Exhaust Stack Height"). This analysis is discussed in more detail in Chapter 13.0. This passive engineered control is designed and fabricated with a fixed height for safe release of gaseous effluents.

9.1.4 Instrumentation and Control Requirements

Section 9.1.2.6 provides a general description of the operation of the RPF ventilation system. Ventilation system control and monitoring is discussed in Chapter 7.0, "Instrumentation and Control Systems." Table 9-2 summarizes the system parameters (in general) and whether they are monitored or alarmed. The system sequence of operation will be developed and provided in the Operating License Application.

Table 9-2. Indications for Facility Ventilation System Parameters

Parameter	Alarm	Monitor
Equipment operating status	✓	✓
Damper position status		✓
Exhaust header pressure	✓	✓
Fan speed	✓	✓
Filter differential pressures	✓	✓
Equipment bearing vibration	✓	✓
Equipment bearing temperatures	✓	✓
HEPA filter unit air inlet temperature	✓	✓
HEPA filter unit airflow rate	✓	✓
First-stage HEPA inlet temperature	✓	✓
Fan motor amperage	✓	✓
Fan thermal overload	✓	
Zone I header pressure	✓	✓
Zone II header pressure	✓	✓
Confinement zone pressure differentials	✓	✓

HEPA = high-efficiency particulate air.

9.1.5 Required Technical Specifications

The technical specifications associated with the ventilation system, if applicable, will be discussed in Chapter 14.0, "Technical Specifications, as part of the Operating License Application.

9.2 MATERIAL HANDLING

The RPF does not handle or store reactor fuel. Material handling activities are discussed in Chapter 4.0, "Radioisotope Production Facility Description," Sections 4.3 and 4.4, and are analyzed in Chapter 13.0.

9.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

The fire protection system is designed to provide varying levels of notification of a fire event, suppress small fires, and prevent small fires from becoming large fires. Notification of personnel will be achieved through detection of a fire by automatic detection devices, manual pull stations, automatic sprinklers, and the use of alarm devices that broadcast within the building and transmit signals to the central alarm station and RPF control room. Suppression of fires will be accomplished through the use of automatic sprinklers where appropriate. The suppression system will include all piping, valves, and fittings from the water supply (i.e., water storage tanks or municipal hydrants) to the automatic sprinklers and standpipes in the building.

9.3.1 Design Basis

The fire protection system design provides detection and suppression of fires in the RPF. The fire protection system design basis includes:

- Providing varying levels of notification of a fire event and transmitting the notification to the site central alarm station and RPF control room
- Suppressing small fires
- Preventing small fires from becoming large fires

Additional information on the design basis is provided in Chapter 3.0, Section 3.5.2.7.

9.3.2 System Description

The fire protection system will provide detection and suppression of fires within the RPF, generation of alarm signals indicating the presence and location of fires, and execution of commands appropriate for the particular location of the fire.

A complete addressable fire alarm system, with both automatic and manual initiation, will be provided throughout the RPF. Detection devices will report to a local alarm panel. All alarms (fire, supervisory and trouble) will transmitted to the site central alarm station and RPF control room. Fire protection system components will have fail-safe features and audible/visual alarms for operability and trouble indication.

The fire detection and alarm subsystem will include smoke detectors, heat detectors, water flow and tamper switches, manual pull stations, horns and strobes, and a notification system. The building fire suppression subsystem will include automatic sprinkler, HEPA filter plenum deluge water sprays, and portable fire extinguishers. Water will be supplied from the exterior fire hydrant supply via connections to the domestic water system. Firewater booster pumps will increase the system pressure in the fire suppression subsystem piping.

Space has been reserved so that if required, the fire protection system can have a dedicated water storage facility onsite. The need for dedicated storage will be dependent on the reliability and flow rate of the city water supply. The storage tank capacity is anticipated to be [Proprietary Information], and will be determined for the Operating License Application. If an on-site water storage system is found to be necessary, an electric motor-driven fire pump will serve as the primary pressure source, and a redundant diesel engine-driven fire pump will provide backup.

Fire protection water will be distributed throughout the building via a gridded water system. Vertical risers will supply various systems, with redundant risers also provided. From the vertical risers, the automatic sprinkler part of the system will feed a series of sprinkler heads that have temperature-sensitive links. When a set temperature is reached at the sprinkler head, the links will melt or break (depending on type) and release water in an umbrella-shaped spray pattern.

The fire protection system is designed to provide a constant flow of water to an area experiencing a fire for a minimum of 120 min. The size of that area will be determined using guidelines from the International Fire Code (IFC, 2012). For sprinkler systems, the International Fire Code uses a design based on the National Fire Protection Association (NFPA) 13, *Standard for the Installation of Sprinkler Systems*. Fire hose stations will also provide flow for use in fighting fires.

Because water from the sprinklers may become contaminated with materials it contacts, areas where hazardous materials are present are designed to hold firewater runoff for sampling prior to release to the environment. Additional detailed information on the firewater runoff storage will be developed for the Operating License Application.

The fire protection system is divided into two major subsystems. The subsystems and components are categorized as follows:

- Fire suppression subsystem consisting of automatic sprinklers, a HEPA filter plenum deluge, glovebox fire suppression, and fire hydrants
- Fire detection and alarm subsystem consisting of:
  - Controls (e.g., fire alarm control panel, subpanels, or devices used for control of devices)
  - General area detection (e.g., room smoke and heat detectors, manual pull stations)
  - Duct smoke detection for non-nuclear ventilation systems, glovebox heat detection
  - HEPA filter plenum heat detection
  - Fire suppression subsystem monitoring devices (e.g., waterflow switches, tamper switches, fire pump, and water storage monitoring devices)
  - Occupant notification
  - Alarm transmission to the central alarm station and RPF control room

9.3.2.1 Fire Suppression Subsystem

The fire suppression subsystem will include automatic sprinklers, HEPA filter plenum deluge, and fire hydrants. The need for fire suppression in gloveboxes will be evaluated and additional information will be provided in the Operating License Application. In addition to the automatic features of the fire suppression subsystem, manual response capabilities will be provided by fire extinguishers with an appropriate classification (discussed further in Section 9.3.2.3).

A 20.3 centimeter (cm) (8-inch [in.]) network of main piping (commonly called a grid) will be provided. Vertical piping, referred to as risers and sized at 15.2 cm (6 in.), will be provided to support the fire suppression subsystem components (sprinklers, HEPA filter plenum deluge, and hydrants). The RPF will also be provided with redundant sprinkler risers. The connection between the risers and sprinkler piping will be provided with control valves, check valves, waterflow switches, and a test/drain assembly for detection of waterflow and system maintenance. Piping from the risers will support automatic sprinklers located throughout the facility. The automatic sprinkler system is designed in accordance with NFPA 13.

The HEPA filter plenum deluge will be also supplied by the 20.3 cm (8-in.) piping network and will be part of a larger filter plenum fire safety design that includes fire screens, demisters, plenum drains, and plenum dampers. The automatic feature will include a deluge valve that is activated via heat detectors in the ducts serving the plenum. When high temperatures are sensed in the air stream, the heat detector will send a signal to the fire alarm control panel, which in turn will send a signal to the deluge valve to operate. Water will flow through the deluge valve into the leading portion of the plenum to cool the air before it reaches the HEPA filters. The heat detectors and deluge valve for a particular plenum will be paired such that only plenums that are experiencing high temperatures will react. A manual bypass feature will be also provided to allow waterflow if the deluge valve fails to open.

A separate, manually activated feature is designed to spray directly on the HEPA filters and is intended to only be used if the HEPA filter ignites. The manual feature will include a control valve connected via piping to a spray nozzle directed at the HEPA filters.

The fire hydrants, located on the exterior of the building, will be supported by the 30.5 cm (12-in.) municipal water supply line provided for the RPF. Two 8-in. connections will support the 20.3 cm (8-in.) loop that surrounds the building. Four fire hydrants, one at each corner of the building, will be provided. The fire hydrants are not designed for natural phenomenon hazards and cannot be relied on for seismic accidents. The fire hydrant subsystem is designed in accordance with NFPA 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, and the International Fire Code (IFC, 2012). The subsystem is designed to support fire flows of 5,680 L/min (1,500 gal/min) overall and at least 1,893 L/min (500 gal/min) at each fire hydrant.

9.3.2.2 Fire Detection and Alarm Subsystem

The fire detection and alarm subsystem will provide a range of fire detection capabilities and notification methods. The primary means of detection will be by monitoring the fire suppression system devices, including flow switches that indicate release of water from automatic sprinklers or deluge valves, and tamper switches that supervise valve position. Smoke and heat detection will be provided in specific locations to provide detection of fires in spaces where water damage concerns warrant improved manual intervention (e.g., computer server rooms), areas deserving additional life safety (e.g., egress locations), or other safety-driven functions. As required by NFPA 101, *Life Safety Code*, and NFPA 72, *National Fire Alarm Code*, smoke detection will be provided above the main fire alarm control panel and any subpanels necessary to perform control functions for the system.

For ventilation units, smoke and heat detection will be provided in support of several safety aspects. Smoke detectors will be provided in:

- Non-nuclear ventilation systems, in accordance with NFPA 90A, *Standard for the Installation of Air-Conditioning and Ventilating Systems*, and the International Fire Code (IFC, 2012)
- Air intakes, to address smoke infiltration from wild land fires and fires in other facilities that might spread smoke to the surrounding area
- Nuclear ventilation systems, to support shutdown and minimize the spread of contaminated smoke to other areas of the RPF

Heat detectors will be provided in the Zone I and II ventilation system exhausts for both notification of high temperatures and release of the automatic portion of the HEPA filter plenum deluge capability. Control modules and relays will be integrated into the fire detection and alarm subsystem. Control modules will provide signals for releasing the deluge valves for the HEPA filter plenum deluge capability, and control methods will be integrated for shutdown of non-safety HVAC systems.

Alarms received by the fire alarm control panel will be transmitted via a copper cable or fiber optic cable network to monitoring stations in the RPF. The fire alarm control panel will also provide notification through the facility-wide infrastructure to the central alarm station. The central alarm station will provide data to the Columbia Fire Department for response.

The fire detection and alarm subsystem will receive its primary power supply from a dedicated circuit off of the normal building power. Internal batteries will provide a secondary power source, with connection to the standby generator. The batteries will be sized to provide 24 hours (hr) of backup power, plus 10 min of alarm power, as required by NFPA standards.

9.3.2.3 Fire Extinguishers

In addition to the automatic features of the fire suppression subsystem, manual response capabilities will be provided via fire extinguishers with an appropriate classification. Fire extinguishers will be located throughout the building, as required by NFPA 10, *Standard for Portable Fire Extinguishers*. Specific extinguisher types, such as those for metal fires or particular chemicals, will be specified depending on the hazard.

9.3.3 Operational Analysis and Safety Function

Chapter 13.0 identifies fire hazards and evaluates adverse events and accident sequences. The criticality safety evaluations discussed in Chapter 6.0 include the impact of fire suppression water in its analysis. Chapter 13.0 provides an evaluation of the accident sequences that involve either combustible solids or liquids, or explosive gases, in close proximity to the high uranium process streams or the high-dose process streams. As part of this analysis, an emergency purge gas system was identified to prevent flammable concentration in process vessel headspaces. IROFS FS-03, "Process Vessel Emergency Purge System," is discussed in Chapter 13.0, Section 13.2.7, and in Chapter 6.0.

The following summarizes NWMI-2013-039, *Preliminary Fire Hazards Analysis* (PFHA), which was prepared to demonstrate that the RPF will maintain the ability to perform safe-shutdown functions and minimize radioactive material releases to the environment in the event of a fire. The PFHA objectives were to:

- Consider potential in situ and transient fire hazards
- Determine the effects of a fire in any location in the RPF and the ability to safely shut down the facility and/or minimize and control the release of radioactivity to the environment
- Specify measures for fire prevention, detection, suppression, and containment for each fire area housing structures, systems, and components that are important to safety, in accordance with U.S. Nuclear Regulatory Commission (NRC) guidelines and regulations

The PFHA assessed the fire hazards at the RPF, support facilities, and surrounding project site. The analysis also assessed the fire safety criteria identified in NRC Regulatory Guide 1.189, *Fire Protection for Nuclear Power Plants*. The PFHA provided a consequence evaluation of a design basis fire (DBF) scenario within each fire area, assuming the loss of automatic and manual fire suppression. The PFHA also identified facility design features and fire hazard mitigating features for personnel safety and property protection commensurate with the NRC criteria.

9.3.3.1 Radioisotope Production Facility Fire Areas

The fire hazards, life safety considerations, fire protection features, and DBF for designated fire areas within the RPF are discussed below.

The RPF will be subdivided into separate fire areas for the purposes of limiting the spread of fire, protecting personnel, and limiting the consequential damage to the facility. Figure 9-7 and Figure 9-8 provide the delineation of fire areas on the first floor and second floor of the RPF, respectively. The determination of fire area boundaries was based on consideration of the following:

- Types, quantities, density, and location of combustible materials
- Location and configuration of equipment
- Consequences of inoperable equipment
- Location of fire detection and suppression systems
- Personnel safety and exit requirements

Fire areas will typically be bounded by 2-hr fire-rated barriers to separate:

- Processing areas and radioactive material storage areas from each other and adjacent areas
- Rooms with major concentrations of electrical and mechanical equipment from adjacent areas
- Computer and control rooms from adjacent areas
- Maintenance shops from adjacent areas
- Combustible storage areas from adjacent areas
- Fan rooms and plenum chambers from adjacent areas
- Office areas from moderate and high fire hazard areas

In one case, two fire areas will be separated by 3-hr fire-rated barrier walls. The fire-rated barrier design and construction are in accordance with the International Building Code (IBC) (ICC, 2012) and NFPA 221, *Standard for High Challenge Fire Walls, Fire Walls, and Fire Barrier Walls.*

Where fire-rated assemblies are partially or fully penetrated by pipes, ducts, conduits, raceways, or other devices, fire-rated barrier material will be placed in and around the penetrations to maintain the fire-resistance rating of the assembly. All openings in the fire barriers will be protected, consistent with the designated fire-resistance rating of the barrier. Fire doors will be rated commensurate with the fire-rated barrier in which they are installed, and comply with the requirements of NFPA 80, *Standard for Fire Doors and Other Opening Protectives.*

[Proprietary Information]

Figure 9-7. Life Safety Plan (First Floor)

[Proprietary Information]

Figure 9-8. Life Safety Plan (Second Floor)

9.3.3.1.1 Hot Cell, Waste Handling, and Shipping Areas

As the most consequential fire area within the RPF, the hot cell area will be a single-story, non-combustible, high bay structure. The footprint of this area will be [Proprietary Information]. The hot cell area will include parts of the irradiated target receipt bay and waste management areas, Mo recovery and purification process, U recovery and recycle process, high bay above the hot cell area, and operating and maintenance galleries. An overhead crane system will be used to transfer radioactive materials between the different operations.

Life Safety Considerations

The hot cell area is anticipated to handle hazardous materials that exceed the maximum allowable quantity limits established in the IBC (ICC, 2012). Therefore, the hot cell area will be designed as High Hazard H-3 and/or H-4 occupancy in accordance with the IBC and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

The common path of egress travel for an H-3 occupancy equipped throughout with an automatic sprinkler system will be 7.6 m (25 ft), in accordance with the IBC Section 1014.3. The exit access travel distance for a fully sprinklered H-3 occupancy will be limited to 45.7 m (150 ft), in accordance with the IBC. Dead-ends in corridors should not exceed 6.1 m (20 ft), in accordance with IBC Section 1018.4.

Access to the crane platforms will be limited to maintenance and service personnel only. IBC Section 505.3 defines equipment platforms as not being habitable and are considered to not be occupiable space. Because the crane platforms will normally be unoccupied with limited access, these crane platforms will not be required to meet IBC means of egress requirements.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

The hot cell, waste, handling and shipping areas will be separated from other fire areas of the building by 2-hr fire-rated barrier walls, with the exception of the wall between the production area and the administrative area, which will have a 3-hr fire-rated barrier wall. Penetrations in the fire-rated barrier walls will be protected with penetration seals, providing a fire rating equivalent to the barriers.

The hot cell area could be exposed to a fire in an adjacent fire area when the large access doors are opened during radiological material transfer activities. The primary areas of concern include the interface (open doors) between the unloading and waste truck bays with the production area. To prevent a fire from spreading between these areas, administrative controls will be implemented that dictate personnel procedures and limit combustibles around interface access doors. Fire spread between areas will be therefore mitigated by personnel actions, limited combustibles, and fire-rated boundaries.

Fire Protection Features

The hot cell area requires the following fire protection features to provide a defense-in-depth approach to fire protection. This approach will result in a fire being quickly detected and suppressed, which will mitigate fire-induced damage.

- **Automatic** – Automatic sprinkler systems will be installed throughout the production area, with the exception of the hot cell enclosure. Self-contained fire suppression systems may be located on equipment such as cranes and forklifts. An automatic fire detection and alarm system will be installed throughout the production area. Analysis of the need for sprinklers in the hot cell area and additional detailed information on these systems will be developed for the Operating License Application.
- **Manual** – Manual fire suppression will consist of portable fire extinguishers and Class I standpipe system hose valves that will be provided within the production area. Manual fire alarm pull stations will be provided at exits from the production area.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect the means of egress from the facility and separation between fire areas. Fuel traps will be provided where the diesel-powered vehicles interface with the production area. Underhung collection pans will be provided under the crane gearboxes.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The following fire hazards and ignition sources were considered for evaluation of a DBF scenario within the production area.

- **Scenario 1** – A fire starts within the irradiated target shipping cask that is caused by agitation and spontaneous ignition of the pyrophoric uranium dust or particulate.
- **Scenario 2** – A fire or explosion starts within a tank or exhaust system that is caused by the uncontrolled accumulation of hydrogen gas. Hydrogen generation represents a fire hazard, where the accident sequence is initiated by failure of the sweep gas subsystem.

- **Scenario 3** – A fire starts within the exhaust stack system that is caused by the ignition of the carbon retention bed and/or HEPA filters.
- **Scenario 4** – A fire starts adjacent to a semi-tractor trailer that is caused by the rupture of the fuel tank and ignition of the unconfined (static) diesel spill.
- **Scenario 5** – A fire starts on a diesel-driven forklift that is caused by the rupture of the fuel tank and ignition of the unconfined (static) diesel spill.
- **Scenario 6** – A fire starts in a crane collection pan that is caused by the rupture of the gearbox and ignition of the confined (static) silicone oil pool.

The DBF scenario for the production area consists of a diesel fuel spill and ignition from an unknown source caused by the operation of a semi-tractor trailer or forklift. The semi-tractor trailer is assumed to have two 284 L (75-gal) diesel fuel tanks (568 L [150 gal total]), along with rubber tires, a battery, and small amounts of other combustible material. A small amount of permanent combustibles, including electrical cables, polyethylene tarps, isopropyl alcohol, vinyl, and trash bins, may also be present. These combustibles will be limited by administrative controls.

The DBF scenario postulates that the entire contents of the fuel tanks will spill, forming an approximately 15.5 m (50-ft) diameter pool with a 3 millimeter (mm) (0.12-in.) depth, and will then ignite. The DBF postulates that any combustibles located within the fuel spill diameter will also ignite and be completely consumed (NWMI-2013-039).

Consequences of an Automatic Fire Suppression Failure

Failure of the automatic fire suppression system will cause a delay in responding to a fire, resulting in the combustibles being completely consumed during the DBF. The adoption of administrative controls will limit combustibles and minimize the spread of fire. However, smoke and hot gases could damage equipment located within the production area.

The Columbia Fire Department will be notified of a fire by either actuation of a manual fire alarm pull box station or the automatic smoke or temperature detection systems. The DBF would be contained within the irradiated target receipt bay and operating gallery by the 2-hr rated fire walls. If the automatic fire suppression system fails to operate, the fire department is expected to arrive well before the 2-hr fire walls have failed and extinguish the fire using portable extinguishers or the hose stream supported by the Class I standpipe system. The required response time of the fire department will be determined for the Operating License Application.

Conclusion

While the DBF for this area is unlikely to result in a radiological release with the radioactive material being contained in a U.S. Department of Transportation (DOT) Type B cask, the potential exists for a release in some of the other scenarios described. Additional information, and a determination if the fire protection systems in this fire area will be considered IROFS, will be provided in the Operating License Application.

9.3.3.1.2 Target Fabrication Area

The target fabrication area will be located adjacent to the production area on the east side of the RPF and will be a noncombustible structure with an industrial F-1 occupancy. Two-hour fire-rated barrier walls will separate the target fabrication area from other fire areas of the building. Penetrations in the fire-rated barrier walls will be protected with penetration seals, providing a fire rating equivalent to the barriers.

The footprint of the target fabrication area will be [Proprietary Information] over most of the area. This area will be dedicated to the production of low-enriched uranium (LEU) targets.

Life Safety Considerations

The target fabrication area is required to meet IBC life safety criteria (ICC, 2012) and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

An accessible means of egress will be provided in accordance with the IBC. Exit access will be provided to the target fabrication area, with direct exit discharge from the RPF. The maximum distances to the exit access in the target fabrication area will be within the following parameters for a High Hazard H-3 occupancy. The common path of egress travel for an H-3 occupancy equipped throughout with an automatic sprinkler system will be 30.5 m (100 ft), in accordance with IBC Section 1014.3. The exit access travel distance for a fully sprinklered F-1 occupancy will be limited to 76.2 m (250 ft), in accordance with IBC Table 1016.2. Dead-ends in corridors will not exceed 15.2 m (50 ft), in accordance with IBC Section 1018.4, Exception 2. No deviations from the IBC life safety criteria have been identified.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

The target fabrication area could be exposed to a fire in an adjacent area when the large access doors are opened during target transfer or waste shipping activities. The primary area of concern is an open doorway to the production area.

To prevent a fire from spreading between these areas, administrative controls will be implemented that dictate personnel procedures and limit combustibles around interface access doors. Additional information on these controls will be provided in the Operating License Application. Fire spread between areas will therefore be mitigated by personnel actions, limited combustibles, and 2-hr fire-rated boundaries.

Fire Protection Features

The target fabrication area requires the following fire protection features to provide a defense-in-depth approach to fire protection. This approach will result in a fire being quickly detected and suppressed, reducing fire-induced damage.

- **Automatic** – An automatic fire suppression system will be installed throughout the target fabrication area. An automatic fire detection and alarm system will be also installed throughout the target fabrication area. The system specifics will be determined during detailed design and included in the Operating License Application.
- **Manual** – Manual fire suppression will be provided within the target fabrication area and consist of portable fire extinguishers and Class I standpipe system hose valves. Manual fire alarm pull stations will be provided at exits from the target fabrication area.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect the means of egress from the facility and separation between fire areas.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The following fire hazards and ignition sources were considered for evaluation of a DBF scenario within the target fabrication area.

- **Scenario 1** – A fire or explosion starts within the reduction subsystem, caused by ignition of a nitrogen or hydrogen gas mixture by the high temperature created by the oven (determined to be highly unlikely based on credible physical conditions [Chapter 13.0]).
- **Scenario 2** – A pyrophoric fire of uranium metal (determined to be highly unlikely based on credible physical conditions [Chapter 13.0]).
- **Scenario 3 –** A fire starts with combustible materials or equipment in the target fabrication area.

The DBF event was determined to be a fire of combustible materials such as paper products (Scenario 3). The DBF for the target fabrication area involves ignition of in situ combustibles located within the area caused by an electrical short circuit or a maintenance welding operation. The combustible loading of the area was considered low. The fire also consumes other transient combustibles located within the area.

Consequences of an Automatic Fire Suppression Failure

Failure of the automatic fire suppression system will cause a delay in responding to a fire, resulting in the combustibles being completely consumed during the DBF. The adoption of administrative controls will limit combustibles and minimize the spread of fire. However, smoke and hot gases could damage equipment located within the target fabrication area.

In the event of a fire, the Columbia Fire Department will be notified by either actuation of a manual fire alarm pull box station or the automatic smoke or temperature detection systems. The DBF would be contained within the target fabrication area by the 2-hr rated fire walls. If the automatic fire suppression system fails to operate, the fire department is expected to arrive well before the 2-hr fire walls have failed and extinguish the fire using portable extinguishers or the hose stream supported by the Class I standpipe system. The required response time of the fire department will be determined for the Operating License Application.

Conclusion

The above analysis and description show that the fire protection and life safety systems within the target fabrication area are designed such that they will function in a manner, whether operational or not, consistent with occupational safety and protection of the public and environment. Two of the three scenarios described are considered highly unlikely. The DBF for this fire area would result in minimal or no release to the public because of the low radiological source term and the fact that the standard combustibles described are unlikely to be mixed with the LEU materials. Therefore, this system will likely not be considered an IROFS.

9.3.3.1.3 Administration and Support Area

The administration and support area will be located adjacent to the production area on the south side of the RPF and will be a single-story, noncombustible structure with business (Group B) and assembly (Group A-2) occupancies. The administration and support area will be Type IIB construction and separated from the remainder of the RPF by 3-hr fire-rated barrier walls.

The administration and support area will include the main entry and security access points, break room, control room, conference room, men's and women's lavatories, and several small offices. The control room will be separated from the remainder of the administration and support area by 2-hr fire-rated barrier walls.

The operations performed within the administration and support areas will be consistent with office space uses. The occupant load of the administration and support area will include non-production work staff.

Life Safety Considerations

The administration and support area is required to meet IBC life safety criteria (ICC, 2012) and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

An accessible means of egress will be provided to the area in accordance with the IBC. Exit access will be provided to the administration and support area by one main exit at the front of the building and a secondary exit located to the south side of the RPF. A break room will also be provided with an additional exit. The maximum distances to the exit access within the administration and support area will be within the following parameters. The travel distance for the common path of egress travel for a mixed use business (B) and assembly (A-2) occupancy equipped throughout with an automatic sprinkler system will be 23 m (75 ft), in accordance with IBC Table 1014.3. The exit access travel distance for a fully sprinklered mixed-use business (B) and assembly (A-2) occupancy will be limited to 76 m (250 ft), in accordance with IBC Table 1016.2. Dead-ends in corridors will not exceed 6.1 m (20 ft), in accordance with IBC Section 1018.4.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

The administration and support area will be separated from other fire areas of the RPF by 3-hr fire-rated barriers. Penetrations in the fire-rated barrier walls will be protected with penetration seals, providing a fire rating equivalent to the barriers. Load-bearing structural elements are not required to be protected by fire-resistive construction.

To prevent a fire from spreading between areas, administrative controls will be implemented that dictate personnel procedures and limit combustibles around access doors. Fire spread between areas will be therefore mitigated by personnel actions, limited combustibles, and 3-hr fire-rated boundaries.

Fire Protection Features

The administration and support area requires the following fire protection features to provide a defense-in-depth approach to fire protection. This approach will result in a fire being quickly detected and suppressed, reducing fire-induced damage.

- **Automatic** – An automatic wet-pipe sprinkler system will be installed throughout the administration and support area. An automatic fire detection and alarm system will also be installed throughout this area. Additional detailed information will be developed for the Operating License Application.
- **Manual** – Manual fire suppression will be provided within the administration and support area and consist of portable fire extinguishers and Class I standpipe system hose valves. Manual fire alarm pull stations will be provided at the exits from the administration and support area.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect the administration and support area from other occupied areas of the facility.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The DBF was determined to consist of ordinary combustibles (e.g. paper products and office furniture) ignited within a closed office caused by an electrical short circuit. The combustible loading of the office was considered low. The fire also consumes other transient combustibles located within the office and spreads to nearby cubicles.

The DBF would result in the complete combustion of the combustible materials in the area of origin. No credit was taken for fire suppression activities. The administration and support area was considered a single fire area, and the result of the DBF was the complete loss of function of the area.

Consequences of an Automatic Fire Suppression Failure

Failure of the automatic fire suppression system will cause a delay in responding to a fire, resulting in the combustibles being completely consumed during the DBF. The adoption of administrative controls will limit combustibles and minimize the spread of fire. However, smoke and hot gases could damage equipment located within the administrative and support area.

In the event of a fire, the Columbia Fire Department will be notified of a fire by either actuation of a manual fire alarm pull box station or the automatic smoke or temperature detection systems. The DBF would be contained within the administrative and support area by the 3-hr rated fire walls. If the automatic fire suppression system fails to operate, the fire department is expected to arrive well before the 3-hr fire walls have failed and extinguish the fire using portable extinguishers or the hose stream supported by the Class I standpipe system. The required response time of the fire department will be determined for the Operating License Application.

Conclusion

The above analysis and description show that the fire protection and life safety systems within the administration and support area are designed such that they will function in a manner, whether operational or not, consistent with occupational safety and protection of the public and environment. Because this fire area is not expected to contain anything other than check sources for instrumentation, no releases to the public are expected to occur. Therefore, this system will likely not be considered an IROFS. Additional detailed information will be developed for the Operating License Application.

9.3.3.1.4 Irradiated Target Receipt and Waste Management Truck Bay Areas

The irradiated target receipt and waste management truck bay areas will be located adjacent to the production area on the north side of the RPF and will be a noncombustible enclosure that is considered a storage S-2 occupancy area. The truck bay will be capable of accepting three semi-tractor trailers at the same time. Each truck bay will be separated from the production area (cask unloading) by a 2-hr fire-rated rollup door. The doors to the production area will be closed when the doors to the outside are open.

This area will be used for the receipt of irradiated LEU targets and shipments involved with the disposal of radiological waste material. Radiological material will be transported in approved containers. The casks will reside on the heavy-duty tractor-trailer for delivery and removal from the RPF. The heavy duty tractor-trailer will be present when the retractable doors are open to the production area.

Life Safety Considerations

The irradiated target receipt and waste management truck bay areas will be required to meet IBC life safety criteria (ICC, 2012) and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

An accessible means of egress will be provided in accordance with the IBC. Exit access will be provided to the truck bays. The maximum distances to the exit access within the truck bay will be established and conform to IBC code based on industrial occupancies. The common path of egress travel for an S-1 occupancy equipped throughout with an automatic sprinkler system will be 30.5 m (100 ft), in accordance with IBC Table 1014.3. The exit access travel distance for a fully sprinklered S-1 occupancy will be limited to 76.2 m (250 ft), in accordance with IBC Table 1016.2. Dead-ends in corridors will not exceed 15.2 m (50 ft), in accordance with IBC Section 1018.4, Exception 2.

No deviations from the IBC life safety criteria have been identified.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

The irradiated target receipt and waste management truck bay areas will be separated from other fire areas in the building by 2-hr fire-rated barriers. Penetrations in the fire-rated barrier walls will be protected with penetration seals, providing a fire rating equivalent to the barriers. Load-bearing structural elements are not required to be protected by fire-resistive construction, as indicated in the IBC (ICC, 2012).

The truck bay could be exposed to a fire in an adjacent fire area when the large access doors are opened to attach or disconnect a trailer to or from a tractor. To prevent a fire from spreading between these areas, administrative controls will be implemented that dictate personnel procedures and limit combustibles around the interface access doors. Personnel actions, limited combustibles, and 2-hr fire-rated boundaries will therefore mitigate fire spread between areas.

Fire Protection Features

The irradiated target receipt and waste management truck bay areas will require the following fire protection features to provide a defense-in-depth approach to fire protection. This approach will result in a fire being quickly detected and suppressed, reducing fire-induced damage.

- **Automatic** – An automatic sprinkler system will be installed throughout the truck bay area. However, due to the large quantity of diesel fuel and number of tires on the heavy-duty tractor-trailer, alternative suppression systems may be considered. An automatic fire detection and alarm system will be installed throughout the truck bay area. Additional detailed information will be developed for the Operating License Application.
- **Manual** – Manual fire suppression will be provided within the truck bay area and consist of portable fire extinguishers and Class I standpipe system hose valves. Manual fire alarm pull stations will be provided within the truck drive-through.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect the means of egress from the facility and separation between fire areas. Built-in fuel traps and sloped floors will be provided to control potential fuel spills within the area. The fuel traps and sloped floors will also be used for containment of potentially contaminated firefighting water. The fuel and/or water will drain to outdoor underground collection tanks for testing and removal.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The following fire hazards and ignition sources were considered for evaluation of a DBF scenario within the truck bay area.

- **Scenario 1** – A fire starts due to maintenance activities (e.g., spark ignition or open flame).
- **Scenario 2** – A fire is caused by hot work (e.g., welding, flame, or plasma cutting).
- **Scenario 3** – A fire starts adjacent to a semi-tractor trailer that is caused by the rupture of the fuel tank and ignition of the unconfined (static) diesel spill.
- **Scenario 4** – A fire starts on a diesel-driven forklift that is caused by the rupture of the fuel tank and ignition of the unconfined (static) diesel spill.

The DBF for the truck bay consists of a diesel fuel spill and ignition from an unknown source caused by the operation of a diesel-powered semi-tractor trailer (Scenarios 3 and 4). The truck is assumed to have two 284 L (75-gal) diesel fuel tanks, along with 32 hard rubber tires, a battery, and small amounts of other combustible material. The truck may also carry some combustibles on noncombustible pallets when supporting radiological material-handling operations. Administrative controls will be used to limit temporary combustible items within the production area.

The DBF scenario postulates that the entire contents of the fuel tanks will spill and drain to the built-in fuel trap. The area of the fire will be limited to the area of the built-in fuel trap trench, which was estimated to be approximately 2.6 m^2 (28 ft^2).

The results of the DBF were postulated as the complete combustion of the combustible materials in the irradiated target receipt truck bay area. No credit was taken for fire suppression activities. The DBF fire could result in the complete loss of function for the systems and/or equipment in the area.

Consequences of an Automatic Fire Suppression Failure

Failure of the automatic fire suppression system will cause a delay in responding to a fire, resulting in the combustibles being completely consumed during the DBF. The adoption of administrative controls will limit combustibles and minimize the spread of fire. However, smoke and hot gases could damage equipment located within the truck bay area.

In the event of a fire, the Columbia Fire Department will be notified of a fire by either actuation of a manual fire alarm pull box station or the automatic smoke or temperature detection systems. The DBF would be contained within the truck bay area by the 2-hr rated fire walls. If the automatic fire suppression system fails to operate, the fire department is expected to arrive well before the 2-hr fire walls have failed and extinguish the fire using portable extinguishers or the hose stream supported by the Class I standpipe system. The required response time of the fire department will be determined for the Operating License Application.

Conclusion

The above analysis and description show that the fire protection and life safety systems within the truck bay are designed such that they will function in a manner, whether operational or not, consistent with occupational safety and protection of the public and environment. Because the radioactive material will be contained in DOT Type B casks, a fire in this area should not result in a radiological release to the public. Therefore, this system will likely not be considered an IROFS. Additional detailed information will be developed for the Operating License Application.

9.3.3.1.5 Laboratory Area

The laboratory area will be located adjacent to the production area on the west side of the RPF and will be a single-story, noncombustible structure with a High Hazard H-3 and H-4 occupancy. The footprint of the laboratory area will be approximately [Proprietary Information] over most of the area. This area will process and analyze quality and process control samples during production of the molybdenum-99 (^{99}Mo) product, fabrication of targets for irradiation, and processing of waste for disposal.

Typical RPF analysis will include:

- An inductively coupled plasma mass spectrometry (ICP-MS) to analyze mass quantities of isotopic [Proprietary Information]
- A kinetic phosphorescence analyzer for [Proprietary Information]
- Alpha spectroscopy for [Proprietary Information]
- Beta activity by liquid scintillation spectrometry for strontium-89/strontium-90 (^{89}Sr/^{90}Sr)
- Gamma energy analysis

A variety of gloveboxes and fume hoods will be located within the analytical laboratory area.

Life Safety Considerations

The laboratory area is required to meet IBC life safety criteria (ICC, 2012) and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

An accessible means of egress will be provided in accordance with the IBC. Exit access will be provided to the laboratory area, with direct exit discharge from the RPF. The maximum distances to the exit access within the laboratory area will be within the following parameters. The common path of egress travel for a mixed High Hazard H-3 occupancy equipped throughout with an automatic sprinkler system will be 7.6 m (25 ft), in accordance with IBC Table 1014.3. The exit access travel distance for a fully sprinklered mixed H-3 occupancy will be limited to 45.7 m (150 ft), in accordance with IBC Table 1016.2. Dead-ends in corridors will not exceed 6.1 m (20 ft), in accordance with IBC Section 1018.4. No deviations from the IBC life safety criteria have been identified.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

The laboratory area will be separated from other fire areas of the building by 2-hr fire-rated barriers. Penetrations in the fire-rated barrier walls will be protected with penetration seals, providing a fire rating equivalent to the barriers. The laboratory area could be exposed to a fire in an adjacent fire area when the large access doors are opened during material transfer activities. The primary area of concern in this case is an open doorway to the production area. To prevent a fire from spreading between these areas, administrative controls will be implemented that dictate personnel procedures and limit combustibles around the interface access doors. Fire spread between areas will therefore be mitigated by personnel actions, limited combustibles, and 2-hr fire-rated boundaries.

Fire Protection Features

The laboratory area requires the following fire protection features to provide a defense-in-depth approach to fire protection. This approach will result in a fire being quickly detected and suppressed, reducing fire-induced damage.

- **Automatic** – An automatic fire suppression system will be designed and installed throughout the laboratory area. An automatic fire detection and alarm system will also be installed throughout the laboratory area. The system specifics will be determined during detailed design and provided in the Operating License Application.
- **Manual** – Manual fire suppression will be provided within the laboratory area and consist of portable fire extinguishers and Class I standpipe system hose valves. Manual fire alarm pull stations will be provided at the exits from the laboratory area.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect the means of egress from the facility and separation between fire areas.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The DBF scenario for the laboratory area will be developed for the Operating License Application.

Consequences of an Automatic Fire Suppression Failure

The consequences of the failure of the automatic fire suppression system in the laboratory area will be determined for the Operating License Application.

Conclusion

More analysis is needed to determine if the fire protection system in this area would be considered an IROFS. Additional detailed information will be developed for the Operating License Application.

9.3.3.1.6 Utility Areas

Utility areas (e.g., electrical rooms, mechanical rooms, fire riser rooms, etc.) will be noncombustible spaces separated from other fire areas by fire-rated barrier walls. The footprint of each utility room will vary, but will be classified as utility (Group U) occupancies in accordance with the IBC (ICC, 2012). These utility areas will include rooms that house electrical equipment (e.g., power and lighting panels, transformers, and associated operations equipment distribution systems) and other common industrial equipment (e.g., air handling units, boilers, fans, pumps, and associated piping distribution systems). Personnel will not normally occupy the utility areas.

Life Safety Considerations

The utility areas are required to meet IBC life safety and means of egress criteria (ICC, 2012) and will be provided with emergency lighting, illuminated exit signs, automatic sprinklers, and an automatic and manually actuated fire alarm system with audible and visual indicating devices as necessary.

An accessible means of egress will be provided in accordance with the IBC. The maximum distances to the exit access within the utility areas will be within the following parameters for utility occupancies. The common path of egress travel for a utility occupancy equipped throughout with an automatic sprinkler system will be 22.9 m (75 ft), in accordance with IBC Table 1014.3. The exit access travel distance for a fully sprinklered utility occupancy will be limited to 121.9 m (400 ft), in accordance with IBC Table 1016.1. Dead-ends in corridors will not exceed 15.2 m (50 ft), in accordance with IBC Section 1018.4, Exception 2.

Exposure Fire Potential/Potential for Fire Spread between Fire Areas

For the purpose of this analysis, the utility areas are each considered separate areas, each with 2-hr fire-rated barrier walls used to limit the spread of fire.

HEPA filters and exhaust carbon beds will be encased by stainless steel housings that can be isolated from the inlet and outlet ductwork by isolation dampers. Fire detectors will also be provided in each HEPA filter housing and inlet ductwork. Therefore, isolation dampers will prevent the fire from propagating from the filter housing to other fire areas.

Fire Protection Features

The utility areas require the following fire protection features to provide a defense-in-depth approach to fire protection. This approach results in a fire being quickly detected and suppressed, reducing fire-induced damage.

- **Automatic** – An automatic wet-pipe sprinkler or other approved fire suppression system will be installed throughout each utility area. An automatic fire detection and alarm system will also be installed throughout each utility area. Additional detailed information will be developed for the Operating License Application.
- **Manual** – Manual fire suppression will be provided within each utility area that consists of portable fire extinguishers.
- **Passive** – Passive fire protection will be provided in the form of fire-rated construction to protect separation between fire areas. Isolation dampers will be provided in the inlet and outlet of each HEPA filter housing to prevent fire from spreading to other fire areas.

Fire Hazards, Ignition Sources, and Design Basis Fire Scenarios

The following were considered DBF scenarios for the utility areas.

- **Scenario 1** – A fire starts due to maintenance activities, ignited from a spark or open flame.
- **Scenario 2** – A fire starts from overheated electrical systems and equipment.
- **Scenario 3** – A fire starts in or near a transformer.
- **Scenario 4** – A natural gas leak occurs.

The DBF for the utility area consists of a natural gas leak resulting in an explosive mixture of natural gas and a detonation or deflagration. Additional information for this accident sequence will be provided in the Operating License Application.

Consequences of an Automatic Fire Suppression Failure

The consequences of a failure of the automatic fire suppression system in the utility area will be determined for the Operating License Application.

Conclusion

More analysis is needed to determine if the fire protection system in this area should be considered an IROFS. Additional detailed information will be developed for the Operating License Application.

9.3.3.2 Other Radioisotope Production Facility Systems

9.3.3.2.1 Facility Ventilation and Smoke Management

The RPF ventilation system requirements must satisfy the process, building, safety, and regulatory requirements unique to the ^{99}Mo production process. To assist in the confinement of airborne radioactive contamination, the RPF ventilation system is designed to create pressure gradients and cause air to flow from areas of lesser contamination potential to areas of greater contamination potential. Confinement zone exhaust ductwork will have fire dampers consistent with NFPA 45, *Standard on Fire Protection for Laboratories Using Chemicals*, and will be constructed to maintain fire ratings where ducting penetrates fire-rated barriers, as appropriate. The confinement ventilation systems will also include HEPA and high-efficiency gas adsorption (HEGA) filtration systems located in a dedicated mechanical area. The Zone I ventilation system will comprise the secondary confinement boundary and be classified as an IROFS (RS-03). Chapter 13.0 provides additional information on the accident analysis that identified this IROFS.

A combination of passive and active smoke management strategies will be used to minimize the spread of smoke, maintain tenable conditions for the evacuation of building occupants, and limit the damage caused by smoke. These strategies will be designed in accordance with NFPA 92, *Standard for Smoke Control Systems*. The smoke control methods for each fire area will be developed for the Operating License Application.

9.3.3.2.2 HEPA Filtration Systems

The HEPA filters and housings are a component of the hot cell secondary confinement boundary that will be classified as an IROFS (RS-03). The HEPA filters are expected to contain low levels of radiological material and will be located in designated fire areas. The filter housings are expected to be large, with a maximum size being approximately [Proprietary Information] in face area. The large filter face area will require automatic and manual sprinklers in the plenum housings and contaminated water collection or retention. In addition, the HEPA filter housings will be located within 2-hr fire-rated barrier walls that are protected by automatic sprinkler systems.

9.3.3.2.3 Crane Superstructure

The structural steel supporting the facility overhead crane has been classified as an IROFS (FS-02, "Overhead Cranes"). Therefore, the crane superstructure must remain standing during and after a fire event to prevent damage to irradiated material. Additional detailed information will be developed for the Operating License Application.

9.3.3.2.4 Security and Safeguard Components

Security systems are discussed in Chapter 12.0, "Conduct of Operations."

9.3.3.3 Architectural Features

The codes and standards applicable to the RPF are defined in Chapter 3.0. The objectives of the NRC fire protection program will primarily be achieved through compliance with prescriptive criteria, as defined by the PFHA (NWMI-2013-039).

Types of Construction

All structures within the RPF complex confines will be constructed of Type IIB, noncombustible material, as defined by IBC Chapter 6 (ICC, 2012). Additional detailed information will be developed for the Operating License Application.

9.3.4 Instrumentation and Control Requirements

The fire protection system will report the status of the fire protection equipment to the central alarm station and the RPF control room, with sufficient information to identify the general location and progress of a fire within the protected area boundaries. Initiating devices for the fire detection and alarm subsystem, which will include monitoring devices for the fire suppression subsystem, will indicate the presence of a fire within the facility. Once an initiating device activates, signals will be sent to the fire alarm control panel. The fire alarm control panel will transmit signals to the central alarm station and perform any ancillary functions, such as shutting down the ventilation equipment or actuating the deluge valves.

As required by NFPA 101 and NFPA 72, smoke detection will be provided above the main fire alarm control panel and any subpanels necessary to perform control functions for the system. For ventilation units, smoke and heat detection will be provided in support of several safety aspects. Smoke detectors will be provided in non-nuclear ventilation systems in accordance with NFPA 90A and the IFC. Smoke detectors will also be provided in air intakes to address smoke infiltration from wildland fires and fires in other facilities that might spread smoke to the surrounding area.

Smoke detection will be provided in ventilation systems servicing potentially contaminated zones to support shutdown and minimize the spread of contaminated smoke to other areas of the RPF. Heat detectors will be provided in these ventilation system exhausts for both notification of high temperatures and release of the automatic portion of the HEPA filter plenum deluge subsystem.

Control modules and relays will be integrated into the fire detection and alarm subsystem to initiate reactions required for safety. Control modules will provide signals for releasing of deluge valves on the HEPA filter plenum deluge subsystem. Control methods will also be integrated for shutdown of the HVAC systems. Shutdown of electrical equipment or computers will also occur as deemed necessary by the design effort.

Alarms received by the fire alarm control panel will be transmitted via a copper cable or fiber optic cable network to monitoring stations in the RPF. The fire alarm control panel will also provide notification through the sitewide infrastructure to the central alarm station. The alarm stations will provide data to the Columbia Fire Department for response.

System Monitoring

The fire protection system will be monitored by the fire alarm control panel, which will transmit signals to the central alarm station via a digital alarm communicator transmitter and to the RPF control room. Command and control functions will be exclusively available at the fire alarm control panel. Localized monitoring of the various fire pumps will occur at the respective pump controllers.

Control Capability and Locations

The fire detection and alarm subsystem will be controlled exclusively from the fire alarm control panel. Numerous devices in the fire suppression subsystem can be operated manually. The fire pumps can be started manually via their respective controllers. Valves and hydrants will be turned manually, and no air or electrically operated valves will be provided. Deluge valves for the HEPA filter plenum water spray can be activated manually, in addition to the bypass valves that are integrated into the design.

Automatic and Manual Actions

The fire detection and alarm subsystem is intended to operate automatically. Manual intervention will be required for some operations, such as shutdown of outside air intake fans or dampers, due to the need to avoid false activation or to maintain operational status in emergency conditions.

The fire suppression subsystem will be split between automatic and manual operations. The sprinkler systems (including the pumps) and the demister section of the HEPA filter plenum deluge subsystem are designed to operate automatically. The filter section of the plenum deluge subsystem and fire hydrants are designed for manual operation. Certain portions, however, can be operated manually as necessary.

The demister section of the HEPA filter plenum deluge subsystem will have a manual bypass and a manual actuator as part of the deluge valve. Portable fire extinguishers will be manually operated.

Maintenance and testing activities on both systems will require manual interaction. The maintenance and testing requirements included in NFPA 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*, and NFPA 72 require manual operation of valves, starting of pumps, testing of circuits with meters, and other functions that necessitate manual actions.

Interlocks, Bypasses, and Permissives

The fire protection system, as designed, will not be subject to external interlocks, bypasses, or permissives (i.e., those outside the system itself). There will be inherent interlocks, bypasses, and permissives within the various fire protection system equipment, which will be designed to the criteria and requirements discussed in Chapter 3.0. For example, the fire detection and alarm subsystem can be controlled via passwords and allow for bypassing certain functions; however, the passwords will be limited to testing technicians and are not available to general building personnel. Thus, there will be no ability for the system to be locally manipulated without proper authorization. Additional information will be provided in Chapter 7.0 for the Operating License Application.

9.3.5 Required Technical Specifications

The technical specifications associated with the fire protection systems, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.4 COMMUNICATION SYSTEMS

The RPF communication systems will relay information during normal and emergency conditions for general operations and emergencies within the RPF. These systems are designed to enable the RPF operator on duty to be in communication with the supervisor on duty, health physics staff, and other personnel required by the technical specifications, and to enable the operator, or other staff, to announce the existence of an emergency in all areas of the RPF complex. Two-way communication will be provided between all operational areas and the control room.

9.4.1 Design Basis

The communications system design basis is to provide communications during normal and emergency conditions between vital areas of the RPF and the Administration Building. This communications capability will include the ability of operators or other designated staff members to announce an emergency in all areas of the RPF and provide two-way communications between all operational areas and the control room. Design of the telecommunication system also complies with Electronic Industries Alliance and Telecommunications Industry Association requirements.

9.4.2 System Description

The communication system is designed to provide two-way communication between the RPF control room and other site locations necessary for safe RPF operations. This system will provide (1) communications capability between RPF operators, their supervisor, health physics personnel, and other personnel as required by the technical specification, and (2) the ability to make facility-wide emergency announcements and summon emergency assistance.

The telephone and data/local area network (LAN) telecommunications system will include a service entrance communications room. The service provider's outside plant optical fiber will terminate on a wall-mounted service provider entrance patch panel. An optional outside plant copper telephone cable from the service provider will terminate at the wall-mounted overvoltage entrance protection terminal modules for use in legacy non-Voice over Internet Protocol (VoIP)-based equipment. The main entrance room will be connected with a telecommunications room with fiber and copper backbone cable. The telecommunications room will support the offices, laboratory area, target fabrication area, shipping and receiving areas, and other required telephone and data/LAN outlets. Grounding of the telecommunication system will comply with Telecommunications Industry Association and NFPA requirements. The process control system will be physically separated from and not connected to the communication system. Additional information will be provided in the Operating License Application.

9.4.3 Operational Analysis and Safety Function

Chapter 13.0 identifies and evaluates adverse events and accident sequences. The accident analysis has not identified the need to credit the communication system. The communication system is designed such that it will function in a manner, whether operational or not, consistent with occupational safety and protection of the public and environment.

9.4.4 Instrumentation and Control Requirements

Chapter 7.0 discusses the instrumentation and control requirements associated with the communication systems.

9.4.5 Required Technical Specifications

The technical specifications associated with the communication systems, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.5 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

The RPF is designed to ensure that:

- No uncontrolled release of radioactive materials (solid, liquid, or airborne) from the facilities can occur
- Personnel exposures to radiation, including ingestion or inhalation, do not exceed limiting values in 10 CFR 20, as defined in Chapter 11.0, and are consistent with the NWMI ALARA program.

The operating procedures developed for the Operating License Application will ensure that only radioactive byproducts handled by the RPF are permitted, unless specifically authorized by the 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," license or an additional license.

9.5.1 Design Basis

The design basis for the possession and use of special nuclear material (SNM) and byproduct material is to ensure that no uncontrolled release of radioactive materials (solid, liquid, or airborne) from the facilities can occur and that personnel exposures to radiation, including ingestion or inhalation, do not exceed limiting values in 10 CFR 20 and are consistent with the NWMI ALARA program. Additional information on the design basis is provided in Chapter 3.0.

9.5.2 System Description

SNM is defined by Title I of the *Atomic Energy Act of 1954* (42 U.S.C. 2011 et seq.) as plutonium, uranium-233 (^{233}U), or uranium enriched in the isotopes ^{233}U or ^{235}U. The RPF will receive, store, and process fresh unirradiated uranium metal and irradiated uranium with an enrichment of 19.75 weight percent (wt%) ±0.20 wt% ^{235}U (LEU).

Byproduct material, as defined by the Atomic Energy Act, is radioactive material (except SNM) yielded in or made radioactive by exposure to radiation incident to the process of producing or using SNM. As defined by NRC regulations, byproduct material includes any radioactive material (except enriched uranium or plutonium) produced by a nuclear reactor. The RPF will handle byproduct material during the separation of ^{99}Mo and the recycling of the irradiated LEU.

Source material is defined as the element thorium or the element uranium, provided that the uranium has not been enriched in the isotope ^{235}U. Source materials will not be present in the RPF.

9.5.2.1 Special Nuclear Materials

SNM will be handled in two areas of the RPF: the target fabrication and irradiated material areas (i.e. hot cells). The target fabrication area SNM inventory is listed in Chapter 4.0, Table 4-1, and the irradiated material area SNM inventory is provided in Chapter 4.0, Table 4-2.

Chapter 4.0 also provides a description of the design of spaces and equipment to ensure that there is no uncontrolled release of radioactive materials (solid, liquid, or airborne) from the RPF and that personnel exposures to radiation, including ingestion or inhalation, do not exceed limiting values in 10 CFR 20 consistent with the RPF ALARA program, as described in Chapter 11.0. Associated procedures are defined in Chapter 12.0. The NWMI emergency preparedness and physical security plans are provided in Chapter 12.0, Appendix A and B, respectively. Fire protection provisions are described in Section 9.3.2.1.

9.5.2.2 Byproduct Materials

Byproduct materials handled in the RPF include ^{99}Mo and radioactive waste materials. A description of the Mo recovery process design is provided in Chapter 4.0, Section 4.3.5. A description of the waste processing design is provided in Chapter 11.0, Section 11.2. A detailed inventory of byproduct materials within each of the main systems within the RPF is provided in the following chapters:

- Target fabrication – Chapter 4.0, Section 4.4.2
- Target receipt and disassembly – Chapter 4.0, Sections 4.3.2 and 4.3.3
- Target dissolution – Chapter 4.0, Section 4.3.4
- Molybdenum recovery and purification – Chapter 4.0, Section 4.3.5
- Uranium recovery and recycle – Chapter 4 0, Section 4.4.1
- Waste handling – Chapter 11.0, Section 11.2

Chapter 4.0 and Section 9.7.2 provide descriptions of the design of spaces and equipment to ensure that there is no uncontrolled release of radioactive materials (solid, liquid, or airborne) from the RPF and that personnel exposures to radiation, including ingestion or inhalation, do not exceed limiting values in 10 CFR 20 consistent with the NWMI ALARA program (Chapter 11.0). Associated procedures will be defined in Chapter 12.0, as part of the Operating License Application.

9.5.3 Operational Analysis and Safety Function

The criticality safety of SNM is discussed in Chapters 4.0 and 6.0, and the material control and accounting of SNM is discussed in Chapter 12.0, Section 12.13. The byproduct materials associated with the RPF process are addressed in Chapter 4.0, and byproduct materials within the waste processing and storage areas are described in Section 9.7.2 and Chapter 11.0, Section 11.2.

9.5.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the processes associated with the possession and use of byproduct materials and SNM are discussed in Chapter 7.0 and Chapter 12.0, Section 12.13.

9.5.5 Required Technical Specifications

The technical specifications associated with the possession and use of byproduct materials and SNM, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.6 COVER GAS CONTROL IN CLOSED PRIMARY COOLANT SYSTEMS

As discussed in Section 9.7.1.2.2, the RPF provides cooling water to numerous process tanks. The radiolytic decomposition of water within this system could result in the production of hydrogen (H_2) and oxygen mixtures. This section provides a discussion of the cover gas control system within the process coolant system.

9.6.1 Design Basis

Information on the design basis of cover gas control in the closed primary coolant system (process chilled water system) is provided in Chapter 3.0, Section 3.5.2.7.

9.6.2 System Description

The process chilled water system is described in Section 9.7.1.2.2. The accumulation of combustible gases within this system will be controlled by the "sweep" gas system that is described in Section 9.7.1.2.6. Gases entrained in the chilled water system will be released in the cooling water collection tanks. Hydrogen, which is the primary component of evolved combustible gases, diffuses very rapidly and will be diluted by the airflow provided by the sweep gas flow.

The plant air supply system (described in Section 9.7.1.2.4) will provide low-flow [Proprietary Information] purge gases to Tanks TK-420 and TK-320. The process vessel vent system will collect the purge gas from each of the tanks and merge the collected vent subsystems into the main facility ventilation system for treatment and filtration. These systems will work together to prevent explosive gas mixtures from developing.

9.6.3 Operational Analysis and Safety Function

Chapter 13.0 evaluates the accident sequences that involve either combustible solids or liquids, or explosive gases, in close proximity to the high uranium process streams or the high-dose process streams. This analysis determined that if the purge air system was not operational, a hydrogen-air concentration in selected tanks could rise above 25 percent of the lower explosive limit, and an ignition source could cause a deflagration or detonation, resulting in the release of radionuclides into the air. The tanks associated with the cooling system are not anticipated to require IROFS controls.

9.6.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the cover gas control in the closed primary coolant system are discussed in Chapter 7.0.

9.6.5 Required Technical Specifications

The technical specifications associated with the cover gas control in the closed primary coolant system, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.7 OTHER AUXILIARY SYSTEMS

Other RPF auxiliary systems that are important to the safety of workers, the public, and environment will include the following:

- Process utilities
- Control and storage of radioactive waste (waste management)
- Analytical laboratory
- Chemical supply

The followings subsections describe these auxiliary systems, including their design basis, system description, operational analysis and safety function, instrumentation and control requirements, and technical specifications.

9.7.1 Utility Systems

The utility systems will provide heating, cooling, process water, compressed gases, instrument, motive force, and other functions to support uranium processing, waste handling, and ventilation. The utility systems will include the following subsystems:

- Process steam
- Process chilled water
- Demineralized water
- Plant and instrument air
- Gas supply, which supplies nitrogen, helium, hydrogen, and oxygen
- Purge/sweep gas

The utility systems are designed to ensure that any potential malfunctions do not cause accidents in the RPF or an uncontrolled release of radioactivity. The systems are designed to ensure that in the event radioactive material is released by the operation of one of these systems, potential radiation exposures would not exceed the limits of 10 CFR 20 and are consistent with the NWMI ALARA program. No function or malfunction of the auxiliary systems will interfere with or prevent safe shutdown of the RPF.

9.7.1.1 Design Basis

The utility systems design basis is provided in Chapter 3.0, Section 3.5.2.7.

9.7.1.2 System Description

Figure 9-9 shows the second floor mechanical utility area where the process steam, chilled process water, demineralized water, and plant or instrument supply air units will be housed. Helium, hydrogen, and oxygen will be provided by bottled gases located near the point of use either in the laboratory area or the target fabrication area. Nitrogen will be provided by a tube trailer for nitrogen located outside of the laboratory area.

[Proprietary Information]

Figure 9-9. Second Floor Mechanical Utility Area

9.7.1.2.1 Process Steam

The process steam system will be divided into a medium-pressure central heating loop (Figure 9-10) and a low-pressure secondary loop within the hot cell (Figure 9-11). Medium pressure steam will be generated by a natural gas-fired boiler (ST-H-100). Low-pressure steam in the secondary loop will be generated by medium-pressure steam in a shell-and-tube heat exchanger (ST-E-200). Medium-pressure steam will be at least 4.2 kilograms (kg)/square centimeter (cm^2) (60 pounds [lb]/square inch [$in.^2$]) gauge, to provide an adequate temperature differential to generate 1.7 kg/cm^2 (25 $lb/in.^2$) gauge steam for the low-pressure steam loop.

[Proprietary Information]

Figure 9-10. Medium-Pressure Steam System

[Proprietary Information]

Figure 9-11. Low-Pressure Steam System

Low-pressure steam will be generated in a vertical shell-and-tube heat exchanger. Automatic blowdown and makeup water streams will limit the content of sludge or dissolved solids in the boiler and steam generation heat exchanger.

9.7.1.2.2 Chilled Water

Process Chilled Water

The process chilled water system is a central process chilled water loop that will cool the three secondary loops:

- One large geometry secondary loop in the hot cell (Figure 9-12)
- One criticality-safe geometry secondary loop in the hot cell (Figure 9-13)
- One criticality-safe geometry secondary loop in the target fabrication area (Figure 9-14).

The central process chilled water loop will rely on three air-cooled chillers, each sized to accommodate 50 percent of the process cooling demands (Figure 9-15). The secondary loops will be cooled by the central chilled water system through plate-and-frame heat exchangers.

Several process demands will require cooling at less than the freezing point of water. These demands will be met with water-cooled refrigerant chiller units, cooled by the secondary chilled water loops.

The chilled water system will operate with cascading pressure differentials. The central system will operate at the highest pressure, and the secondary loops will operate at a pressure between the central system and the process fluid. The large-geometry secondary loop in the hot cell will meet the cooling demands where fissile material leaking through a heat exchanger is not a credible event. The other cooling loops will be inherently criticality-safe by geometry, so active controls will not be required to keep fissile material out of the chilled water return. At each process cooling demand where fissile material may be present, conductivity sensors will monitor the chilled water return to detect heat exchanger leaks.

Facility Chilled and Heating Water

The HVAC system will maintain the occupied space at 24°C (75°F) (summer) and 22°C (72°F) (winter), with active ventilation to support workers and equipment. The facility chilled water and heating water systems will provide heating and cooling media to the HVAC system.

The facility chilled water system (FCW) will supply the HVAC system with cooling water that is circulated through the chilled water coils in the air-handling units. The air will be drawn across the coils and cooled to be delivered to the RPF production area to maintain temperature. The FCW will provide cooling water at a temperature of 9°C (48°F) to the HVAC air-handling unit cooling coils. There will be three equal-sized facility chillers located adjacent to the RPF: two in operation and one spare.

The heating water system (HW) will supply the HVAC system with heating water that is circulated through the heating water coils in the air handling units. The air will be drawn across the coils and cooled to be delivered to the RPF production area to maintain temperature. The HW will provide heating water at a temperature of 82°C (180°F) to the HVAC air-handling unit heating coils and reheat coils. The heating water will be generated as a byproduct stream of the steam boilers.

This page intentionally left blank.

[Proprietary Information]

Figure 9-12. Chilled Water System Large Geometry Hot Cell Loop

[Proprietary Information]

Figure 9-13. Chilled Water System Critically Safe Hot Cell Loop

[Proprietary Information]

Figure 9-14. Chilled Water System Target Fabrication Loop

[Proprietary Information]

Figure 9-15. Process Chilled Water System

9.7.1.2.3 Demineralized Water

Demineralized water will be produced by removing mineral ions from municipal water through an ion exchange (IX) process (Figure 9-16). Water passes through anion and cation exchange media located in separate IX tanks (DX-IX-100 and 110), and the demineralized water will accumulate in a storage tank (DW-TK-120). A feed pump will provide the water at 4.2 kg/cm^2 (60 lb/in.^{2}) gauge (DW-P-125) for RPF process activities. The IX media will be regenerable using a strong acid and a strong base (DW-P-105 and 115). Acid and base will be fed from local chemical drums by toe pumps.

9.7.1.2.4 Plant and Instrument Air

Plant air will be provided for several activities (e.g., tool operation, pump power, purge gas in tanks, valve actuation, and bubbler tank level measurement) (Figure 9-17). Small, advective flows of plant air will be used throughout the RPF to prevent accumulation of combustible gases to hazardous concentrations. Combustible gases will be evolved from process liquids due to exposure of these liquids to ionizing radiation.

The plant air system will provide air to the instrument air subsystem. The instrument air subsystem will use plant air that is filtered and dried (IA-V-110A, 110B, and IA-F-110). Plant air will be generated by a compressor (PA-K-100) and cooled to near-ambient temperatures by an aftercooler (PA-E-100). The lead/lag configuration can supply reduced flow after a single compressor failure. The plant air receiver will provide buffer capacity to make up the difference between peak demand and compressor capacity (PA-V-100).

Instrument air will be dried in regenerable desiccant beds to a dew point of no greater than -40°C (-40°F) and filtered to a maximum 40 micron (μ) particle size. The instrument air receiver will provide buffer capacity (IA-V-120) to make up the difference between peak demand and compressor capacity.

9.7.1.2.5 Gas Supply

Gas supply of helium (Figure 9-18), hydrogen, and oxygen (Figure 9-19) will be supplied by standard gas bottles. Nitrogen will be provided from a tube truck (Figure 9-18). The nominal capacity of the gas bottles will be 8,495 L (300 ft^3). The nitrogen will be fed from the tube truck (GS-Z-100) to the chemical supply room where manifold piping will be used to distribute the gas. The primary use of nitrogen will be in the reducing furnaces during target fabrication.

Helium, hydrogen, and oxygen gas bottles will be located near the points of use. Gas supply pressures will be regulated to 1.7 kg/cm^2 (25 lb/in.^{2}) gauge at the bottle (Figure 9-19. Where lower pressures are required, point-of-use gas regulators will be installed. Automatic gas cylinder changeover valves will provide a continuous gas supply when one bottle (or rack of bottles) is empty, and alert the operator when bottles need to be replaced. Hydrogen and oxygen gas bottles will be stored in ventilated gas cabinets with 13 air changes/min to mitigate the risk of leaks. The ventilation demand will be 8.8 L/min (250 ft^3/min) air for each gas cabinet.

This page intentionally left blank.

[Proprietary Information]

Figure 9-16. Demineralized Water System

[Proprietary Information]

Figure 9-17. Plant Air System

[Proprietary Information]

Figure 9-18. Nitrogen and Helium Supply System

[Proprietary Information]

Figure 9-19. Hydrogen and Oxygen Supply System

9.7.1.2.6 Purge Gas

The plant air and nitrogen supply systems, described in Section 9.7.1.2.4 and Section 9.7.1.2.5, provide purge gases to the required tanks. Depending on the tank, the purge gas will be provided through the bubbler tank level measurement device or other means. The purge gas flow rates are specified as either high flow for conditions of a large tank or high radioactivity, or low flow where the tank is small and radioactivity is low. Table 9-3 provides the purge gas flows for both the high and low flow rates.

Table 9-3. Purge Gas Flows

Gas flow	Flow rate		Units (basis)
	L/min	gal/min	
High purge	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Low purge	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

[a] NWMI-2013-CALC-005, *Tank Air Bleed Estimate*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2014.

[b] NWMI-2013-CALC-009, *Uranium Purification System Equipment Sizing*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2014.

H_2 = hydrogen gas. U = uranium.

The process vessel vent system will collect the purge gas from each of the vessels and treat it before discharge to the Zone I exhaust. The process vessel vent system merges the collected vent subsystems into the main facility ventilation system for treatment and filtration. These systems will work together to prevent explosive gas mixtures from developing in the headspace of the process vessels. The tanks anticipated to require purge gas are listed in Table 9-4. Additional information on the purge gas system will be developed for the Operating License Application.

Table 9-4. Tanks Requiring Purge Gas

Tank number	Tank name	Tank number	Tank name
DS-D-100	Dissolver 1	UR-TK-120A	Impure uranium collection tank 2A
DS-D-200	Dissolver 2	UR-TK-120B	Impure uranium collection tank 2B
DS-TK-800	Waste collection and sampling tank 1	UR-TK-140A	Impure uranium collection tank 3A
DS-TK-820	Waste collection and sampling tank 2	UR-TK-140B	Impure uranium collection tank 3B
MR-TK-100	Feed tank 1A	UR-TK-160A	Impure uranium collection tank 4A
MR-TK-140	Feed tank 1B	UR-TK-160B	Impure uranium collection tank 4B
MR-TK-180	U solution collection tank	UR-TK-200	IX feed tank 1
MR-TK-200	Feed tank 2	UR-TK-900	IX waste collection tank 1
MR-TK-340	Waste collection tank	UR-TK-920	IX waste collection tank 2
UR-TK-100A	Impure uranium collection tank 1A	WH-TK-100	High-dose waste collection tank
UR-TK-100B	Impure uranium collection tank 1B	WH-TK-240	High-dose concentrate collection tank

IX = ion exchange. U = uranium.

9.7.1.3 Operational Analysis and Safety Function

Chapter 13.0 evaluates the accident sequences that involve fissile solution or solid materials being introduced into systems not normally designed to process these solutions or solid materials. The accident analysis associated with utilities addresses fissile solution leaks across a mechanical boundary between process vessels or backflows into a utility system.

Defense-in-depth – The tank and vessel walls will be made of corrosion-resistant materials and have wall thicknesses that are rated for long service with harsh acidic or basic chemicals.

Items relied on for safety – Based on the analysis conducted in Chapter 13.0, Section 13.2, the following IROFS are implemented.

- IROFS CS-10, "Closed Safe Geometry Heating/Cooling Loop with Monitoring and Alarm"
- IROFS CS-20, "Evaporator/Concentrator Condensate Monitoring"
- IROFS CS-27, "Closed Heating/Cooling Loop with Monitoring and Alarm"
- IROFS FS-03, "Process Vessel Emergency Purge System"
- IROFS CS-18, "Backflow Prevention Device"

The analyses that identified these IROFS and the associated system descriptions are addressed in Chapter 13.0 and Chapter 6.0, respectively.

9.7.1.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the processes associated with the utility system are discussed in Chapter 7.0.

9.7.1.5 Required Technical Specifications

The technical specifications associated with the utility system, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.7.2 Control and Storage of Radioactive Waste

The radioactive waste control and storage systems are designed to ensure that (1) any potential malfunctions do not cause accidents in the RPF or uncontrolled release of radioactivity, and (2) in the event radioactive material is released by the operation of one of these systems, potential radiation exposures would not exceed the limits of 10 CFR 20 and remain consistent with the NWMI ALARA program. No function or malfunction of the auxiliary systems will interfere with or prevent safe shutdown of the RPF.

9.7.2.1 Design Basis

The waste handling system design basis is provided in Chapter 3.5.2.7.

9.7.2.2 System Description

To fulfill the design basis, the control and storage of radioactive waste will include the following functions:

- High-dose liquid waste handling (collection, concentration, and solidification)
- Low-dose liquid waste handling (collection, evaporation, recycle and solidification)
- Spent resin dewatering
- Solid waste encapsulation
- High-dose waste decay
- High-dose waste handling
- Waste handling
- Waste Staging and Shipping Building (Class A storage)

These functions are described in detail in the following subsections.

Figure 9-21 summarizes the weekly design basis volumes and the average annual weekly volumes of all waste handling process streams. The design basis volume is based on eight University of Missouri Research Reactor (MURR) targets and 30 Oregon State University (OSU) TRIGA[1] Reactor (OSTR) targets per week to provide appropriately sized tanks. The annual weekly average is based on processing eight MURR targets per week for 44 weeks per year and 30 OSTR targets per week for eight weeks per year and is used in the sizing of the high-dose decay storage.

[1] TRIGA (Training, Research, Isotopes, General Atomics) is a registered trademark of General Atomics, San Diego, California.

This page intentionally left blank.

[Proprietary Information]

Figure 9-20. Waste Management Process Flow Diagram and Process Flow Streams

9.7.2.2.1 High-Dose Liquid Waste Handling

Figure 9-21 shows the location in the hot cell area where the high-dose liquid waste will be processed. High-dose liquid waste will be collected in the high-dose waste collection tank (shown in Figure 9-22), which will provide the needed handling capacity to match the volume of liquid waste generated by the upstream processes. Chapter 4.0 provides descriptions of the high-dose liquid streams that will be directed to the collection tank.

[Proprietary Information]

Figure 9-21. High-Dose Liquid Waste Solidification Subsystem and Low-Dose Collection Tank Location

The process stream volumes are summarized in Figure 9-20, and Table 9-5 provides the high-dose waste tank capacities. The process streams include:

- Caustic scrubber waste
- Oxidizing column waste
- NO_x absorber waste
- Regeneration waste from TiO_2 #1 IX
- Raffinate/rinsate from #2 IX
- Raffinate/rinsate from #3 IX
- U IX waste

[Proprietary Information]

Figure 9-22. Simplified High-Dose Waste Handling Process Flow Diagram

Table 9-5. High-Dose Waste Tank Capacities

Tank ID	Description/purpose	Tank capacity	
		gal	L
WH-TK-100	High-dose waste accumulation tank	5,050	19,000
WH-TK-240	High-dose concentrate accumulation tank	1,270	4,800

Additions to the collection tank are in discrete, analyzed batches. Sodium hydroxide solution will be added as needed to neutralize any excess acidity. The neutralized liquid will be forwarded to the high-dose waste concentrator, where water is evaporated from the high-dose liquid, condensed, and directed to the condensate collection tank. The evaporator bottoms will be directed to a high-dose concentrate collection tank.

Figure 9-23 shows the arrangement of the high-dose waste handling equipment. A HIC will be transferred into the high-dose waste treatment hot cell through the HIC transfer drawer, and docked with the high-dose solidification mixer. Solidification agent will be transferred to the designated bin from the distribution hopper, which will be loaded by operators in the low-dose waste solidification area. High-dose liquid waste concentrate from the waste concentrate collection tank and solidification agent will be metered into the HIC by the high-dose solidification mixer that may consist of an in-line mixer or a sacrificial paddle within the HIC. After filling and mixing are complete, the high-dose solidification mixer will be disengaged, and the HIC lidded and prepared for transfer to the high-dose waste decay subsystem for storage.

[Proprietary Information]

Figure 9-23. High-Dose Waste Treatment and Handling Equipment Arrangement

9.7.2.2.2 Low-Dose Liquid Waste Handling

Figure 9-24 shows the location of the low-dose liquid waste collection tank. Low-dose condensate from the high-dose concentrator will be held in the condensate collection tank (Figure 9-25). Chapter 4.0 provides descriptions of the low-dose liquid streams that will be directed to the collection tank. The process stream volumes are summarized in Figure 9-20, and Table 9-6 provides the low-dose waste tank capacities. Low-dose liquid received from other upstream processes, combined with the low-dose condensate not recycled, will be transferred to the low-dose waste collection tank where the contents of the tank will be analyzed and adjusted with sodium hydroxide (NaOH) to neutralize any residual acids. Once neutralized, the low-dose waste will then be forwarded to the first of two evaporation tanks located on the second floor (Figure 9-24). In these heated tanks, the liquid will be held at elevated temperatures (60°C [140°F]), and high rates of ventilation air will be passed through the tank. The heated tank contents, plus the high rate of ventilation, will evaporate excess water, reducing the volume of solid waste generated. Samples will be collected and analyzed to ensure compliance with waste acceptance criteria.

[Proprietary Information]

Figure 9-24. Low-Dose Liquid Waste Evaporation System Location

[Proprietary Information]

Figure 9-25. Low-Dose Liquid Waste Disposition Process

Table 9-6. Low-Dose Waste Tank Capacities

Tank ID	Description/purpose	Tank capacity	
		gal	L
WT-TK-400	Condensate tank for high-dose evaporator	4,300	16,250
WH-TK-420	Low-dose waste accumulation tank	5,900	22,300
WH-TK-500	Low-dose waste evaporation tank (LD-1)	5,900	22,300
WH-TK-530	Low-dose evaporation tank (LD-2)	2,600	9,800

The partially concentrated low-dose liquid waste will be transferred to the low-dose waste solidification area (Figure 9-26), where the waste will be metered into a drum that has been placed in the low-dose solidification hood (WH-EN-600). Solidification product vendor information indicates that a ratio of 56.7 to 79.4 kg (125 to 175 lb) of solidification agent is sufficient to solidify 59 to 178 L (42 to 47 gal) of liquid waste within a 55-gal drum. The drum will be lidded at the drum lidding station. With time, the mixture will solidify within the waste drum. The filled waste drum will be loaded onto a shipping pallet and transferred by pallet jack to the shipping and receiving airlock door.

[Proprietary Information]

Figure 9-26. Low-Dose Liquid Waste Solidification Equipment Arrangement

9.7.2.2.3 Spent Resin Dewatering

Spent resin dewatering will be conducted in the high-dose waste treatment hot cell. Figure 9-27 provides the flow diagram for the spent resin dewatering subsystem. This subsystem will transfer uranium recovery and recycle system spent IX resin slurry from the spent resin collection tanks located in the tank hot cell (Figure 9-28) to the dewatering filling head in the high-dose waste treatment hot cell (Figure 9-23). The dewatering filler head will remove liquid from the resin. Dry resin will be collected in a waste drum, and the liquid returned to the low-dose waste collection tank.

The solid waste drum transfer drawer (WH-TP-810) (Figure 9-23) will be opened, and the high-dose waste handling crane will be used to lift the drum and place it in the solid waste drum scan for characterization. After characterization is complete, the drum will be transferred by the high-dose waste handling crane from the solid waste drum scan feed conveyor and placed into a five-drum rack. As determined by characterization, the drum will either be held for decay storage or transferred to the high-dose waste handling system for transfer to a shipping cask.

[Proprietary Information]

Figure 9-27. Spent Resin Dewatering Operational Flow Diagram

[Proprietary Information]

Figure 9-28. Spent Resin Collection Tanks Location

9.7.2.2.4 Solid Waste Encapsulation

Figure 9-29 provides the flow diagram for the solid waste encapsulation subsystem. Operators will enter the maintenance gallery and retrieve the solid waste drum cart from the waste collection port and transfer the drum cart into the high-dose waste treatment hot cell (Figure 9-23). The solid waste drum access port will be opened, and the solid waste encapsulation grout mixer (WH-Z-800) filling nozzle will be docked for waste encapsulation. After the grout filling is complete, the solid waste encapsulation grout mixer filling nozzle will be removed, and the solid waste drum access port closed. The solid waste drum transfer drawer will be opened, and the high-dose waste handling crane will be used to lift the drum and place it in the solid waste drum scan for characterization. After characterization is complete, the drum will be transferred by the high-dose waste handling crane from the solid waste drum scan feed conveyor and placed into a five-drum rack. As determined by the drum's characterization, the drum will either be held for decay storage or transferred to the high-dose waste handling subsystem for transfer to a shipping cask.

[Proprietary Information]

Figure 9-29. Solid Waste Encapsulation Operational Flow Diagram

9.7.2.2.5 High-Dose Waste Decay

Figure 9-30 provides the flow diagram for the high-dose waste decay subsystem. This subsystem will provide lag storage capability for solidified liquid waste and the five-drum racks with high-dose source terms. After HICs or five-drum racks have been filled and lidded in the high-dose waste treatment hot cell, they will be transferred to the high-dose waste decay subsystem.

[Proprietary Information]

Figure 9-30. High-Dose Waste Decay Operational Flow Diagram

The high-dose waste decay cell lift (WH-L-900) (Figure 9-31) will lower the HIC or five-drum rack into the high-dose waste decay cell, where the high-dose waste decay cell conveyor (WH-CN-900) will transfer the HIC or five-drum rack to its decay storage position. The HIC or five-drum rack will remain in storage for a set amount time to allow for short-lived radioisotopes in the waste to decay to lower levels. When the HIC or five-drum rack has decayed to an acceptable activity level, the high-dose waste decay cell conveyor (WH-CN-900) will transfer the HIC or five-drum rack to the high-dose waste decay cell lift, where the HIC or rack will be raised into the high-dose waste treatment hot cell and then transferred to the high-dose waste handling area.

[Proprietary Information]

Figure 9-31. High Dose Waste Decay Cell Equipment Arrangement

9.7.2.2.6 High-Dose Waste Handling

Figure 9-32 provides the flow diagram for the high-dose waste handling subsystem. This subsystem will provide the capability to remotely transfer high-dose waste containers into a shipping cask. When a HIC or two five-drum racks are ready for shipment, the high-dose waste handling crane will be used to open the high-dose waste shipping transfer port (WH-TP-1000) and then transfer the HIC or two five-drum racks, from within the high-dose waste handling area, through the high-dose waste shipping transfer port, and into a shipping cask.

[Proprietary Information]

Figure 9-32. High Dose Waste Handling Operational Flow Diagram

9.7.2.2.7 Waste Handling

The simplified operational flow diagram for the waste handling subsystem is shown in Figure 9-33.

[Proprietary Information]

Figure 9-33. Waste Handling Flow Diagram

The waste handling subsystem will have multiple material handling capabilities. The liquid high-dose radiological waste and solid radiological waste handling will begin with the arrival of a truck and lowboy trailer transporting an empty DOT-approved cask (Figure 9-34). The truck, trailer, and shipping cask will enter the RPF to the waste management loading bay via an exterior facility high-bay door. The shipping cask will then be documented for material tracking and accountability per the safeguards and security system requirements. Operators will use the utility system's truck bay spray wand for any necessary wash-down of the truck, trailer, or shipping cask while located in the waste management loading bay. The operators will remove the shipping cask's upper impact limiter using the waste shipping overhead crane (WH-L-1100) (Figure 9-34). The upper impact limiter will be placed in the designated impact limiter landing zone and secured. Operators will unbolt the lid and prepare the DOT-approved shipping cask for loading per the cask loading and unloading procedure. At this point, the truck, trailer, and shipping cask will enter the waste loading area via a high-bay door. The trailer containing the DOT-approved shipping cask will be positioned below the high-dose waste shipping transfer port (WH-TP-1000) of the contaminated waste system. The truck will be disconnected from the trailer and exit the RPF via the high-bay doors in which the vehicle entered. All high-bay doors will be verified as closed and the shipping cask will then be in position and ready for loading per the contaminated waste system procedures.

[Proprietary Information]

Figure 9-34. Waste Handling Equipment Arrangement

After the DOT-approved cask has been loaded, the shipping cask will be separated from the high-dose waste shipping transfer port (WH-TP-1000). The truck will enter the RPF into the waste management loading bay via an exterior facility high-bay door, and operators will use the utility system's truck bay overhead spray wand for any necessary wash-down of the truck while located in the waste management loading bay. The truck will then enter the waste loading area via a high-bay door. The truck will be connected to the trailer and exit to the waste loading area in the waste management loading bay. At this point, the facility process control and communications system will allow operators to replace the shipping cask's upper impact limiter using the waste shipping overhead crane (WH-L-1100). The shipping cask will be documented for material tracking and accountability per the safeguards and security system requirements (Chapter 12.0). The truck, trailer, and shipping cask will exit the RPF through the high-bay doors in which the vehicle entered.

The liquid low-dose radiological waste handling process will begin with the arrival of a truck transporting the empty waste drum pallets to the fresh and unirradiated shipping and receiving area. The receiving area door will be opened, and the truck will be docked to the receiving bay, allowing for transfer of the waste drum pallets into the RPF. Pallet-loaded empty waste drums will be unloaded from the truck using the waste handling pallet jack (WH-PH-1100). All unloaded empty waste drum pallets will then be documented for material tracking and accountability per the safeguards and security system requirements. The pallet jack carrying an empty waste drum pallet will be transferred to the shipping and receiving airlock door, where the empty waste drums will enter the contaminated waste system for loading.

After the waste drums have been loaded with liquid low-dose radiological waste and re-palletized, a pallet containing full waste drums will be transferred via the waste handling pallet jack (WH-PH-1100) from the shipping and receiving airlock door to the waste loading area. The waste handling forklift (WH-PH-1110) will then enter the waste management loading bay via an exterior facility high-bay door.

A waste shipping truck will also enter the waste management loading bay via an exterior facility high-bay door. Operators will open the high-bay door to the waste loading area and use the forklift to load the waste drum pallet into the truck. The shipping truck will then be documented for material tracking and accountability per the safeguards and security system requirements. The truck containing the waste pallets will exit the RPF through the high-bay doors in which the vehicle entered.

9.7.2.2.8 Waste Staging and Shipping Building (Class A Storage)

The Waste Staging and Shipping Building will be approximately [Proprietary Information] and will provide additional waste storage and shipping preparation for Class A radioactive waste prior to disposal.

9.7.2.3 Operational Analysis and Safety Function

Chapter 13.0, Section 13.2 evaluates the accident sequences that involve fissile solution or solid materials being introduced into systems not normally designed to process these solutions or solid materials. The waste handling system is not geometrically safe; therefore, a number of IROFS have been identified.

- IROFS RS-01, "Hot Cell Liquid Confinement Boundary"
- IROFS RS-03, "Hot Cell Secondary Confinement Boundary"
- IROFS RS-04, "Hot Cell Shielding Boundary"
- IROFS RS-08, "Sample and Analysis of Low Dose Waste Tank Dose Rate Prior to Transfer Outside the Hot Cell Shielding Boundary"
- IROFS RS-10, "Active Radiation Monitoring and Isolation of Low Dose Waste Transfer"
- 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 U Concentration Prior to Discharge or Disposal"
- IROFS CS-18, "Backflow Prevention Device"
- 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, "Non-Destructive Assay (NDA) of Items with Inaccessible Surfaces"
- IROFS CS-24, "Independent NDA of Items with Inaccessible Surfaces"
- IROFS CS-25, "Target Housing Weighing Prior to Disposal"
- IROFS CS-26, "Active Discharge Monitoring and Isolation"
- IROFS FS-01, "Enhanced Lift Procedure"
- IROFS FS-02, "Overhead Cranes"

Additional information on the analyses that identified these IROFS is provided in Chapter 13.0.

9.7.2.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the processes associated with the control and storage of radioactive waste are discussed in Chapter 7.0.

9.7.2.5 Required Technical Specifications

The technical specifications associated with the control and storage of radioactive waste, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.7.3 Analytical Laboratory

The analytical laboratory will support production of the ^{99}Mo product and recycle of uranium. Samples from the process will be collected, transported to the laboratory, and prepared in the laboratory gloveboxes and hoods, depending on the analysis to be performed.

9.7.3.1 Design Basis

The RPF analytical laboratory design basis is to provide on-site analysis to support the production of ^{99}Mo product and fabrication of targets for irradiation. This analysis will be used to determine (1) mass, concentration and purity of SNM, (2) concentration of ^{99}Mo product and product impurities, (3) process stream chemical and radionuclide concentrations, and (4) chemical and radionuclide analysis for waste handling and disposition. Analysis will be required to:

- Verify acceptable ^{99}Mo product to ship
- Confirm uranium content
- Determine adjustments for feed tanks and other associated adjustments
- Verify that recycled uranium product complies with product specification
- Ensure compliance with waste acceptance criteria

9.7.3.2 System Description

The RPF analytical laboratory space will include the following:

- Hoods to complete sample preparation, waste handling, and standards preparation
- Hoods for specialty instruments, including an ICP-MS and kinetic phosphorescence analyzer
- Glovebox for ICP-MS
- Gloveboxes for sample delivery and preparation prior to sample transfer to hoods
- Countertops for the gamma spectroscopy system, low-energy photon spectroscopy, alpha spectroscopy system, liquid scintillation system, and beta-counting system
- Storage for chemical and laboratory supplies
- Benchtop systems, such as balances, pH meters, and ion-chromatography

The analytical laboratory layout is presented in Figure 9-35 and provides space for eight hoods, four gloveboxes, and two countertops.

[Proprietary Information]

Figure 9-35. Analytical Laboratory Layout

Analytical instrumentation will include the ICP-MS, kinetic phosphorescence analyzer, gamma energy analysis, alpha spectroscopy, liquid scintillation spectrometry, and gamma energy analysis.

9.7.3.3 Operational Analysis and Safety Function

Chapter 13.0 evaluates the accident sequences that involve miscellaneous chemical safety process upsets in areas without significant fissile or high-dose licensed material present (chemical storage areas and the laboratory). The accidents analyzed that are associated with the analytical laboratory include Accident Sequence S.R.31, "Chemical Burns from Contaminated Solutions During Sample Analysis." No laboratory IROFS have been identified.

Defense-in-depth – Operators and laboratory technicians will follow set protocols on sampling and analysis to identify the sampling locations, sampling techniques, containers to be used, transport routes to take, analysis procedures, reagents to use, equipment requirements, and disposal protocol for the sample residue material. Each of these procedures will be evaluated for standard safety protocols, including requirements in the safety datasheets for the chemicals used and safety requirements for the equipment used.

9.7.3.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the processes associated with the analytical laboratory will be discussed in Chapter 7.0 as part of the Operating License Application.

9.7.3.5 Required Technical Specifications

The technical specifications associated with the analytical laboratory, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.7.4 Chemical Supply

The chemical supply system will include tanks supplying aqueous chemicals to the process systems, flammable material storage cabinets used to segregate incompatible materials, and storage of chemical solids used in the process systems.

9.7.4.1 Design Basis

The chemical supply system design basis is to provide chemical solutions mixed to the required concentrations that are used within the target fabrication, target dissolution, Mo recovery and purification, and waste management systems. The system will provide nitric acid, NaOH, reductant and NO_x absorber solutions, hydrogen peroxide, and fresh uranium IX resin. Additional information is provided in Chapter 3.0, Section 3.5.2.7.

9.7.4.2 System Description

Figure 9-36 shows the layout of the chemical supply room within the RPF. Tanks are sized to provide support to the process requirements.

9.7.4.2.1 Subsystem 100, Nitric Acid

Subsystem 100 will consist of five tanks that provide the following functions:

- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

[Proprietary Information]

Figure 9-36. Chemical Supply Room Equipment Layout

Figure 9-37 provides the flow diagram for Subsystem 100, and Table 9-7 provides a summary description of the tanks in this subsystem.

[Proprietary Information]

Figure 9-37. Nitric Acid Flow Diagram

Table 9-7. Subsystem 100, Nitric Acid Tank Sizes

Tank number	Chemical	Working volume (L)	Total volume (L)	Diameter (in.)	Height (in.)
CS-TK-100	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	120	135
CS-TK-130	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	12	20
CS-TK-150A	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	84	83
CS-TK-150B	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	84	83
CS-TK-180A	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	84	110
CS-TK-180B	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	84	110
CS-TK-300	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	18	21
CS-TK-320	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	18	21
CS-TK-600A/B/C/D	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]		

HNO_3 = nitric acid. IX = ion exchange.

9.7.4.2.2 Subsystem 200, Sodium Hydroxide

Subsystem 200 will consist of three tanks that provide the following functions:

- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

Figure 9-38 provides the flow diagram for the NaOH subsystem, and Table 9-8 provides a summary description of the tanks in this subsystem.

[Proprietary Information]

Figure 9-38. Sodium Hydroxide Flow Diagram

Table 9-8. Subsystem 200, Sodium Hydroxide Tank Sizes

Tank number	Chemical	Working volume (L)	Total volume (L)	Diameter (in.)	Height (in.)
CS-TK-200	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	84	96
CS-TK-230	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	18	18
CS-TK-260	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	24	27
CS-TK-350	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	6	8

NaOH = sodium hydroxide. NO_x = nitrogen oxide.

9.7.4.2.3 Subsystem 300, Reductant and NO_x Absorber Solutions

Subsystem 300 will consist of three tanks that provide the following functions:

- [Proprietary Information]
- [Proprietary Information]

Table 9-9 provides a summary description of the tanks in Subsystem 300.

Table 9-9. Subsystem 300, Reductant and Nitrogen Oxide Absorber Solutions Tank Sizes

Tank number	Chemical	Working volume (L)	Total volume (L)	Diameter (in.)	Height (in.)
CS-TK-300	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	18	21
CS-TK-320	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	18	21
CS-TK-340	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	6	8

NO_x = nitrogen oxide.

9.7.4.2.4 Subsystem 400, Hydrogen Peroxide

Subsystem 400 will provide the following functions:

- [Proprietary Information]
- [Proprietary Information]

Figure 9-39 provides the flow diagram for the hydrogen peroxide subsystem. The subsystem will consist of one tank (CS-TK-400), which is summarized in Table 9-10.

[Proprietary Information]

Figure 9-39. Hydrogen Peroxide Flow Diagram

Table 9-10. Subsystem 400, Hydrogen Peroxide Tank Sizes

Tank number	Chemical	Working volume (L)	Total volume (L)	Diameter (in.)	Height (in.)
CS-TK-400	Hydrogen peroxide	[Proprietary Information]	[Proprietary Information]	9	12

9.7.4.2.5 Subsystem 600, Fresh Uranium Ion Exchange Resin

Subsystem 600 will consist of four tanks (one tank to support each uranium IX column) that provide the following functions:

- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

Table 9-11 provides a summary description of the tanks in Subsystem 600.

Table 9-11. Subsystem 600, Fresh Uranium Ion Exchange Resin Tank Sizes

Tank number	Chemical	Working volume (L)	Total volume (L)	Diameter (in.)	Height (in.)
CS-TK-600A	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	24	24
CS-TK-600B	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	24	24
CS-TK-600C	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	24	24
CS-TK-600D	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	24	24

IX = ion exchange.

These tanks will support preparation of fresh resin for addition to an IX column after spent resin has been removed. A description of the fresh resin makeup activity is summarized as follows:

- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

Once resin has been prepared by fines removal and washing, the makeup tank will be adjusted to contain a total volume of [Proprietary Information]. The makeup tank low-speed agitator will be started to suspend the resin inventory, and the valve opened to route the suspension to an IX column.

9.7.4.3 Operational Analysis and Safety Function

Chapter 13.0 evaluates accident sequences that involve miscellaneous chemical safety process upsets in areas without significant fissile or high-dose licensed material present (e.g., chemical storage areas and the laboratory). The backflow of fissile or radioactive solutions into auxiliary systems (e.g., chemical supply) was also analyzed and two preventive IROFS identified.

Defense-in-depth – NWMI will comply with U.S. Environmental Protection Agency and Occupational Safety and Health Administration regulations for the design, construction, and operation of chemical preparation and storage areas in the RPF. Chemical handling procedures will be provided to operators to ensure safe handling of chemicals according to applicable regulatory requirements and consistent with the material safety datasheets.

Items relied on for safety – Based on the analysis conducted in Chapter 13.0, Section 13.2, the following IROFS will be implemented:

- CS-18, "Backflow Prevention Device"
- CS-19, "Safe Geometry Day Tanks"

9.7.4.4 Instrumentation and Control Requirements

Instrumentation and control requirements for the processes associated with the chemical supply system will be discussed in Chapter 7.0 as part of the Operating License Application.

9.7.4.5 Required Technical Specifications

The technical specifications associated with the chemical supply system, if applicable, will be discussed in Chapter 14.0 as part of the Operating License Application.

9.8 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.

40 CFR 61, "National Emission Standards for Hazardous Air Pollutants," *Code of Federal Regulations*, Office of the Federal Register, as amended.

42 U.S.C. 2011 et seq., "Atomic Energy Act of 1954," *United States Code*, as amended.

ICC, 2012, "International Building Code (IBC) and Commentary 2012," International Code Council, Falls Church, Virginia, 2012.

IFC, 2012, *International Fire Code*, International Code Council, Falls Church, Virginia, 2012.

ISO 14644-1, "Cleanrooms and Associated Controlled Environments—Part 1: Classification of Air Cleanliness," International Organization for Standardization, Geneva, Switzerland, 1999.

NFPA 10, *Standard for Portable Fire Extinguishers*, National Fire Protection Association, Quincy, Massachusetts, 2013.

NFPA 13, *Standard for the Installation of Sprinkler Systems*, National Fire Protection Association, Quincy, Massachusetts, 2013.

NFPA 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, National Fire Protection Association, Quincy, Massachusetts, 2013.

NFPA 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*, National Fire Protection Association, Quincy, Massachusetts, 2014.

NFPA 45, *Standard on Fire Protection for Laboratories Using Chemicals*, National Fire Protection Association, Quincy, Massachusetts, 2015.

NFPA 72, *National Fire Alarm and Signaling Code*, National Fire Protection Association, Quincy, Massachusetts, 2013.

NFPA 80, *Standard for Fire Doors and Other Opening Protectives,* National Fire Protection Association, Quincy, Massachusetts, 2013.

NFPA 90A, *Standard for the Installation of Air-Conditioning and Ventilating Systems*, National Fire Protection Association, Quincy, Massachusetts, 2015.

NFPA 92, *Standard for Smoke Control Systems*, National Fire Protection Association, Quincy, Massachusetts, 2015.

NFPA 101, *Life Safety Code*, National Fire Protection Association, Quincy, Massachusetts, 2015.

NFPA 221, *Standard for High Challenge Fire Walls, Fire Walls, and Fire Barrier Walls*, National Fire Protection Association, Quincy, Massachusetts, 2015.

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, *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.

NWMI-2013-039, *Preliminary Fire Hazards Analysis*, Rev. C, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2015.

NWMI-2013-CALC-005, *Tank Air Bleed Estimate*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2014.

NWMI-2013-CALC-009, *Uranium Purification System Equipment Sizing*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2014.

Regulatory Guide 1.189, *Fire Protection for Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, 2009.

NWMI
NORTHWEST MEDICAL ISOTOPES

Chapter 13.0 – Accident Analysis

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1
June 2017

Prepared by:
Northwest Medical Isotopes, LLC
815 NW 9th Ave, Suite 256
Corvallis, Oregon 97330

This page intentionally left blank.

Chapter 13.0 – Accident Analysis

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 1

Date Published:
June 26, 2017

Document Number: NWMI-2013-021	*Revision Number*: 1
Title: Chapter 13.0 – Accident Analysis Construction Permit Application for Radioisotope Production Facility	
Approved by: Carolyn Haass	*Signature*: Carolyn C Haass

This page intentionally left blank.

REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required
1	6/26/2017	Incorporate changes based on responses to NRC Requests for Additional Information	C. Haass

This page intentionally left blank.

CONTENTS

13.0 RADIOISOTOPE PRODUCTION FACILITY ACCIDENT ANALYSIS 13-1
13.1 Accident Analysis Methodology and Preliminary Hazards Analysis 13-3
13.1.1 Methodologies Applied to the Radioisotope Production Facility Integrated Safety Analysis Process 13-3
13.1.1.1 Accident Likelihood Categories, Consequence Severity Categories, and Risk Matrix 13-5
13.1.1.2 Accident Consequence Analysis 13-7
13.1.1.3 What-If and Structured What-If 13-7
13.1.1.4 Hazards and Operability Study Method 13-8
13.1.1.5 Event Tree Analysis 13-8
13.1.1.6 Fault Tree Analysis 13-8
13.1.1.7 Failure Modes and Effects Analysis 13-8
13.1.2 Accident-Initiating Events 13-8
13.1.3 Preliminary Hazards Analysis Results 13-12
13.1.3.1 Hazard Criteria 13-12
13.1.3.2 Radioisotope Production Facility Accident Sequence Evaluation 13-13
13.2 Analysis of Accidents with Radiological and Criticality Safety Consequences 13-39
13.2.1 Reserved 13-40
13.2.2 Liquid Spills and Sprays with Radiological and Criticality Safety Consequences 13-40
13.2.2.1 Initial Conditions 13-40
13.2.2.2 Identification of Event Initiating Conditions 13-45
13.2.2.3 Description of Accident Sequences 13-45
13.2.2.4 Function of Components or Barriers 13-45
13.2.2.5 Unmitigated Likelihood 13-46
13.2.2.6 Radiation Source Term 13-46
13.2.2.7 Evaluation of Potential Radiological Consequences 13-48
13.2.2.8 Identification of Items Relied on for Safety and Associated Functions 13-51
13.2.2.9 Mitigated Estimates 13-55
13.2.3 Target Dissolver Offgas Accidents with Radiological Consequences 13-55
13.2.3.1 Initial Conditions 13-56
13.2.3.2 Identification of Event Initiating Conditions 13-57
13.2.3.3 Description of Accident Sequences 13-57
13.2.3.4 Function of Components or Barriers 13-57
13.2.3.5 Unmitigated Likelihood 13-57
13.2.3.6 Radiation Source Term 13-58
13.2.3.7 Evaluation of Potential Radiological Consequences 13-58
13.2.3.8 Identification of Items Relied on for Safety and Associated Functions 13-59
13.2.3.9 Mitigated Estimates 13-60

13.2.4 Leaks into Auxiliary Services or Systems with Radiological and Criticality Safety Consequences 13-60
13.2.4.1 Initial Conditions 13-60
13.2.4.2 Identification of Event Initiating Conditions 13-64
13.2.4.3 Description of Accident Sequences 13-65
13.2.4.4 Function of Components or Barriers 13-65
13.2.4.5 Unmitigated Likelihood 13-65
13.2.4.6 Radiation Source Term 13-66
13.2.4.7 Evaluation of Potential Radiological Consequences 13-66
13.2.4.8 Identification of Items Relied on for Safety and Associated Functions 13-66
13.2.4.9 Mitigated Estimates 13-70
13.2.5 Loss of Power 13-70
13.2.5.1 Initial Conditions 13-70
13.2.5.2 Identification of Event Initiating Conditions 13-70
13.2.5.3 Description of Accident Sequences 13-70
13.2.5.4 Function of Components or Barriers 13-71
13.2.5.5 Unmitigated Likelihood 13-71
13.2.5.6 Radiation Source Term 13-71
13.2.5.7 Evaluation of Potential Radiological Consequences 13-71
13.2.5.8 Identification of Items Relied on for Safety and Associated Functions 13-71
13.2.6 Natural Phenomena Events 13-72
13.2.6.1 Tornado Impact on Facility and Structures, Systems, and Components 13-72
13.2.6.2 High Straight-Line Winds Impact the Facility and Structures, Systems, and Components 13-73
13.2.6.3 Heavy Rain Impact on Facility and Structures, Systems, and Components 13-73
13.2.6.4 Flooding Impact to the Facility and Structures, Systems, and Components 13-74
13.2.6.5 Seismic Impact to the Facility and Structures, Systems, and Components 13-74
13.2.6.6 Heavy Snow Fall or Ice Buildup on Facility and Structures, Systems, and Components 13-75
13.2.7 Other Accidents Analyzed 13-76
13.2.7.1 Items Relied on for Safety for Radiological Accident Sequences (S.R.) 13-86
13.2.7.2 Items Relied on for Safety for Criticality Accident Sequences (S.C.) 13-88
13.2.7.3 Items Relied on for Safety for Fire or Explosion Accident Sequences (S.F.) 13-94
13.2.7.4 Items Relied on for Safety for Natural Phenomena Accident Sequences (S.N.) 13-94
13.2.7.5 Items Relied on for Safety for Man-Made Accident Sequences (S.M.) 13-95
13.2.7.6 Items Relied on for Safety for Chemical Accident Sequences (S.CS.) 13-95

13.3 Analysis of Accidents with Hazardous Chemicals 13-96
13.3.1 Chemical Burns from Contaminated Solutions During Sample Analysis 13-96
13.3.1.1 Chemical Accident Description 13-96
13.3.1.2 Chemical Accident Consequences 13-96
13.3.1.3 Chemical Process Controls 13-96
13.3.1.4 Chemical Process Surveillance Requirements 13-96
13.3.2 Nitric Acid Fume Release 13-97
13.3.2.1 Chemical Accident Description 13-97
13.3.2.2 Chemical Accident Consequences 13-97
13.3.2.3 Chemical Process Controls 13-97
13.3.2.4 Chemical Process Surveillance Requirements 13-97
13.4 References 13-98

FIGURES

Figure 13-1. Integrated Safety Analysis Process Flow Diagram 13-4

Figure 13-2. Unmitigated Off-Site Dose of Dissolver Product Spray Leak Accident 13-50

TABLES

Table 13-1. Likelihood Categories 13-5

Table 13-2. Qualitative Likelihood Category Guidelines 13-5

Table 13-3. Radioisotope Production Facility Consequence Severity Categories Derived from 10 CFR 70.61 13-6

Table 13-4. Radioisotope Production Facility Risk Matrix 13-6

Table 13-5. Radioisotope Production Facility Preliminary Hazard Analysis Accident Sequence Category Designator Definitions 13-9

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 13-9

Table 13-7. Radioisotope Production Facility Preliminary Hazards Analysis Primary Process Nodes and Subprocesses (2 pages) 13-10

Table 13-8. Crosswalk of Radioisotope Production Facility Preliminary Hazards Analysis Process Nodes and Top-Level Accident Sequence Categories 13-12

Table 13-9. Adverse Event Summary for Target Fabrication and Identification of Accident Sequences Needing Further Evaluation (4 pages) 13-14

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (4 pages) 13-18

Table 13-11. Adverse Event Summary for Molybdenum Recovery and Identification of Accident Sequences Needing Further Evaluation (3 pages) 13-22

Table 13-12. Adverse Event Summary for Uranium Recovery and Identification of Accident Sequences Needing Further Evaluation (4 pages) 13-25

Table 13-13. Adverse Event Summary for Waste Handling and Identification of Accident Sequences Needing Further Evaluation (2 pages) 13-29

Table 13-14. Adverse Event Summary for Target Receipt and Identification of Accident Sequences Needing Further Evaluation (2 pages) 13-31

Table 13-15. Adverse Event Summary for Ventilation System and Identification of Accident Sequences Needing Further Evaluation 13-33

Table 13-16. Adverse Event Summary for Node 8.0 and Identification of Accident Sequences Needing Further Evaluation (5 pages) 13-34

Table 13-17. Bounding Radionuclide Liquid Stream Concentrations (4 pages) 13-41

Table 13-18. Source Term Parameters 13-47

Table 13-19. Release Consequence Evaluation RASCAL Code Inputs 13-49

Table 13-20. Spray Release Consequence Summary 13-50
Table 13-21. Maximum Bounding Inventory of Radioiodine [Proprietary Information] 13-56
Table 13-22. Target Dissolver Offgas Accident Total Effective Dose Equivalent 13-59
Table 13-23. Bounding Radionuclide Liquid Stream Concentrations (4 pages) 13-61
Table 13-24. Analyzed Accidents Sequences (9 pages) 13-76
Table 13-25. Summary of Items Relied on for Safety Identified by Accident Analyses (3 pages) 13-84
Table 13-26. Accident Sequence Category Definitions 13-86

TERMS

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
MHA	maximum hypothetical accident
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, the MHA results, 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.

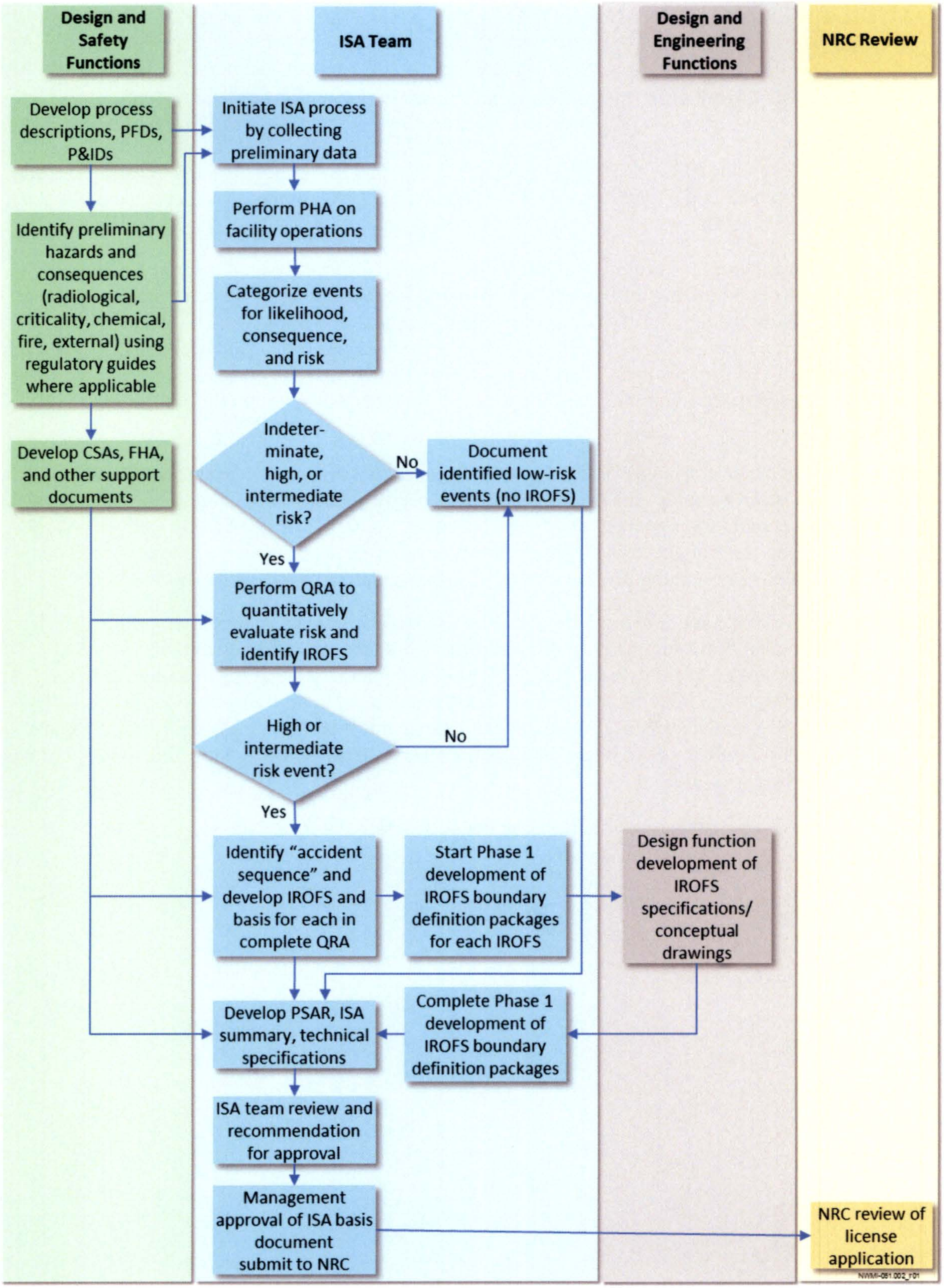

Figure 13-1. Integrated Safety Analysis Process Flow Diagram

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)	Unlikely (Likelihood category 2)	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 (4 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 (4 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

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
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
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

Table 13-10. Adverse Event Summary for Target Dissolution and Identification of Accident Sequences Needing Further Evaluation (4 pages)

PHA item numbers	Bounding accident description	Consequence	Accident sequence
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 offgasing 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 offgasing
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, Offgasing 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, 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]
^{157}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)**
^{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.

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. The unmitigated spray release of dissolver product solution is an immediate consequence event. 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.

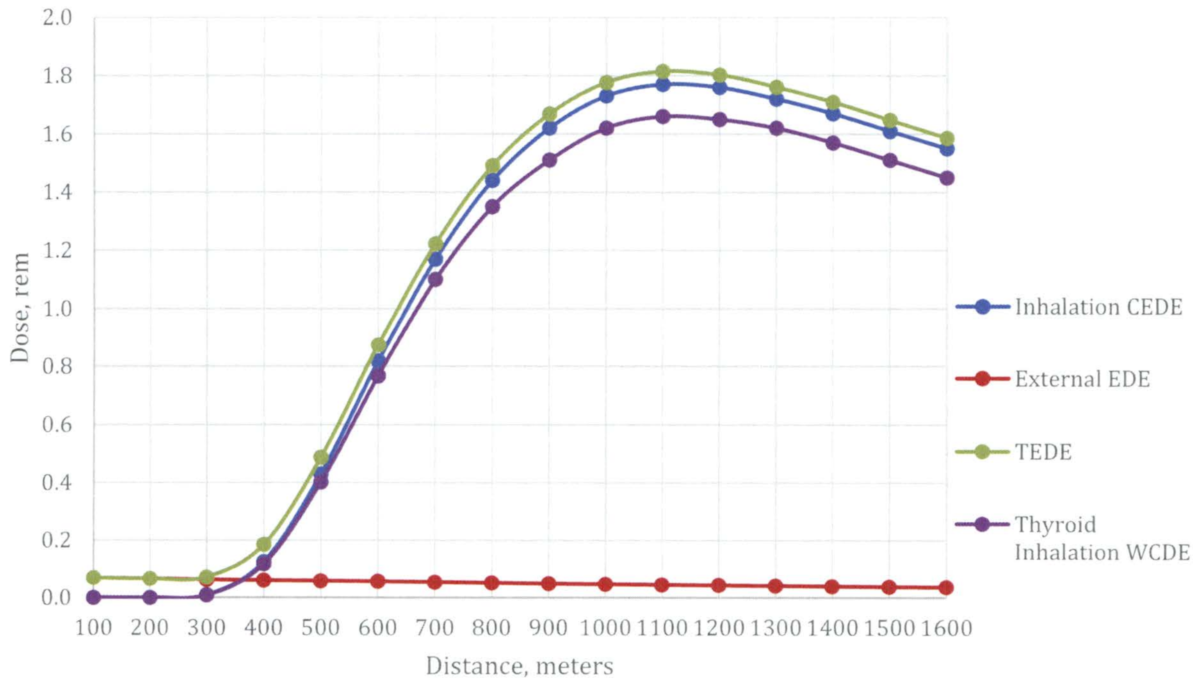

Figure 13-2. Unmitigated Off-Site Dose of Dissolver Product Spray Leak Accident

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.

Table 13-20. Spray Release Consequence Summary

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.

EDE = effective dose equivalent.

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 MHA, as discussed in Chapter 19.0, 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 MHA 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 MHA 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.

There is no credible event where the inventory on the IRUs would be released. Therefore, this evaluation focuses on accident sequences where the inventory at risk is 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.

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 MHA 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 off gas 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 of the MHA (Section 19.4.. The unmitigated dose consequences should be about 3.4 times less than the MHA 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 characteristic as the MHA, 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 resulting from the MHA. 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 MHA dose consequences and dividing by a ratio of the MHA and the accident source term. The maximum public dose is 6.65 rem at 1,100 m.

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

Distance (m)	TEDE (rem) 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.

[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 231.6 m (760 ft), or 16.8 m (55 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, or design basis earthquake, for the RPF is specified as the risk-targeted maximum considered earthquake (MCE_R), as determined in accordance with ASCE 7 and Federal Emergency Management Agency (FEMA) P-753, *NEHRP Recommended Seismic Provisions for New Buildings and Other Structures*. The MCE_R for this site is governed by the probabilistic maximum-considered earthquake ground-shaking, which has an annual frequency of exceedance of 4×10^{-4} (2,500-yr return period). This event is considered unlikely.

Using the provisions of ASCE 7 for standard industrial facilities with equivalent chemical hazards, Risk Category IV results in a design basis earthquake equal to the safe-shutdown earthquake specified. When designed in accordance with ASCE 7 and companion standards, the maximum probability of a complete or partial structural failure is 3 percent conditioned on the occurrence of the maximum-considered earthquake ground-shaking, or a probability of failure of 1.2×10^{-5}. Therefore, 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, $f_{earthquake} = 4\times10^{-4}$.

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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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 offgasing	• Low consequence event
S.R.23	Offgasing 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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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 immediate 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

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

Accident sequence designator from PHA	Descriptor	Preliminary IROFS Identified
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
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.	PEC	=	passive engineered control.
IROFS	=	items relied on for safety.	PHA	=	preliminary hazards analysis.
IRU	=	iodine removal unit.	SSC	=	structures, systems, and components.
IX	=	ion exchange.	TCE	=	trichloroethylene
LEU	=	low-enriched uranium.	U	=	uranium.
Mo	=	molybdenum.	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		

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-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		
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		✓

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

IROFS designator	Descriptor	Engineered safety feature	Administrative control
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	✓	
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.

13.4 REFERENCES

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

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 50, "Domestic Licensing of Production and Utilization Facilities," *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 70.61, "Performance Requirements," *Code of Federal Regulations*, Office of the Federal Register, as amended.

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

ACI 318, *Building Code Requirements for Structural Concrete*, American Concrete Institute, Farmington Hills, Michigan, 2014.

AISC 360, *Specification for Structural Steel Buildings*, American Institute of Steel Construction, Chicago, Illinois, 2010.

ANS 2.8, *Determining Design Basis Flooding at Power Reactor Sites*, American Nuclear Society, La Grange Park, Illinois, 1992,W2002.

ANSI N14.6, *Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More for Nuclear Materials*, American Nuclear Society, La Grange Park, Illinois, 1993.

ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors*, American Nuclear Society, La Grange Park, Illinois, 1998 (Reaffirmed 2007).

ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, American Society of Civil Engineers, Reston, Virginia, 2010.

ASME B30.2, *Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)*, American Society of Mechanical Engineers, New York, New York, 2005.

CDC, 2010, *NIOSH Pocket Guide to Chemical Hazards*, 2010-168c, Centers for Disease Control and Prevention, http://www.cdc.gov/niosh/npg/, downloaded February 27, 2015.

DC/COL ISG-07, *Interim Staff Guidance on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures*, U.S. Nuclear Regulatory Commission, Washington, D.C., 2008.

DOE-HDBK-3010, *DOE Handbook – Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Change Notice No. 1, U.S. Department of Energy, Washington, D.C., December 1994 (R2013).

DOE-STD-1090-2011, *Hoisting and Rigging*, U.S. Department of Energy, Washington, D.C., September 30, 2011.

EPA 520/1-88-020, Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, U.S. Environmental Protection Agency, Washington, D.C., September 1988.

FEMA, 2011, "Flood Insurance Rate Map," Panel 295 of 470, Boone County, Missouri and Incorporated Areas, Map # 29019C0295D, Federal Emergency Management Agency, Washington, D.C., March 17, 2011.

FEMA P-753, *NEHRP Recommended Seismic Provisions for New Buildings and Other Structures*, Federal Emergency Management Agency, Washington, D.C., 2009.

Hydrometeorological Report No. 51, *Probable Maximum Precipitation Estimates, United States East of the 105th Meridian*, U.S. Department of Commerce, National Oceanic and Atmospheric Administration, Washington, D.C., 1978.

Hydrometeorological Report No. 53 (NUREG/CR-1486), *Seasonal Variation of 10-Square Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian*, U.S. Department of Commerce, National Oceanic and Atmospheric Administration, U.S. Nuclear Regulatory Commission, Office of Hydrology National Weather Service, Washington, D.C., April 1980.

IBC, 2012, *International Building Code*, as amended, International Code Council, Inc., Washington, D.C., February 2012.

ICRP-26, *Recommendations of the International Commission on Radiological Protection*, International Commission on Radiological Protection, Ottawa, Canada, 1977.

ICRP-30, *Limits for Intakes of Radionuclides by Workers*, International Commission on Radiological Protection, Ottawa, Canada, 1979.

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.

LA-13638, *A Review of Criticality Accidents*, Los Alamos National Laboratory, Los Alamos, New Mexico, 2000.

NAP 1994, *Estimating Bounds on Extreme Precipitation Events*, National Academy Press, National Research Council, Washington, D.C., 1994.

NOAA Technical Report NWS 25, *Comparison of Generalized Estimates of Probable Maximum Precipitation with Greatest Observed Rainfalls*, National Oceanic and Atmospheric Administration, Washington, D.C., 1980.

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.

NUREG-1940, *RASCAL 4: Description of Models and Methods*, U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Washington, D.C., December 2012.

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.

NWMI-2013-CALC-006, *Overall Summary Material Balance – MURR Target Batch*, Rev. D, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2015.

NWMI-2013-CALC-011, *Source Term Calculations*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2015.

NWMI-2014-051, *Integrated Safety Analysis Plan for the Radioisotope Production Facility*, Rev. A, Northwest Medical Isotopes, Corvallis, Oregon, 2014.

NWMI-2014-CALC-014, *Selection of Dominant Target Isotopes for NWMI Material Balances*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2014.

NWMI-2015-RPT-009, *Fission Product Release Evaluation*, Rev. B, Northwest Medical Isotopes, LLC, Corvallis, Oregon, 2015.

NWMI-2015-SAFETY-001, *NWMI Radioisotope Production Facility Preliminary Hazards Analysis*, Rev. A, Northwest Medical Isotopes, Corvallis, Oregon, 2015.

NWMI-2015-SAFETY-004, *Quantitative Risk Analysis of Process Upsets Associated with Passive Engineering Controls Leading to Criticality Accident Sequences,* Rev. A, Northwest Medical Isotopes, Corvallis, Oregon, 2015.

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C., February 1983.

WSRC-TR-93-262, *Savannah River Site Generic Data Base Development*, Rev. 1, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina, May 1988.