Public
Chapter 5

NORTHWEST MEDICAL ISOTOPES

Chapter 5.0 – Coolant Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0
June 2015

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Chapter 5.0 – Coolant Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0

Date Published:
June 29, 2015

***Document Number*:** **NWMI-2013-021**	***Revision Number*:** **0**
***Title*:** **Chapter 5.0 – Coolant Systems** **Construction Permit Application for Radioisotope Production Facility**	
***Approved by*:** **Carolyn Haass**	***Signature*:** Carolyn C Haass

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REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required

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TERMS

Acronyms and Abbreviations

^{234}U	uranium-234
^{235}U	uranium-235
^{236}U	uranium-236
^{237}U	uranium-237
^{238}U	uranium-238
^{239}Pu	plutonium-239
CFR	Code of Federal Regulations
EOI	end of irradiation
I	iodine
IROFS	item relied on for safety
Kr	krypton
Mo	molybdenum
MURR	University of Missouri Research Reactor
NWMI	Northwest Medical Isotopes, LLC
OSTR	Oregon State University TRIGA Reactor
OSU	Oregon State University
RPF	radioisotope production facility
U	uranium
[Proprietary Information]	[Proprietary Information]
Xe	xenon

Units

°C	degrees Celsius
°F	degrees Fahrenheit
BTU	British thermal unit
cm	centimeter
cm^2	square centimeter
cm^3	cubic centimeter
ft^2	square feet
g	gram
hr	hour
in.	inch
$in.^2$	square inch
kW	kilowatt
L	liter
lb	pound
rem	roentgen equivalent in man
W	watt
wk	week
wt%	weight percent

5.0 COOLANT SYSTEMS

5.1 SUMMARY DESCRIPTION

Cooling water systems are used to control the temperature of process solutions in the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) from process activities and the heat load resulting from radioactive decay of the fission product inventory. The RPF is located at a separate site, independent from the reactors used to irradiate the targets. Therefore, the RPF cooling system does not influence operation of a reactor primary core cooling system.

Chilled water is used as the primary cooling fluid to process vessels. A central process chilled-water loop is used to cool three secondary loops: one large geometry secondary loop in the hot cell, one criticality-safe geometry secondary loop in the hot cell, and one criticality-safe geometry secondary loop in the target fabrication area. The central process chilled-water loop relies on air-cooled chillers, while the secondary loops are cooled by the central chilled-water system through plate-and-frame heat exchangers. Selected process demands require cooling at less than the freezing point of water. These demands are met with water-cooled refrigerant chiller packages, cooled by the secondary chilled water loops.

5.1.1 Irradiated Target Basis

Thermal characteristics of irradiated targets entering the RPF depend on the source reactor and decay time prior to receipt. Heat load estimates are currently based on preliminary calculations for targets irradiated at the Oregon State University (OSU) TRIGA[1] Reactor (OSTR) [Proprietary Information]. The calculations are based on the OSTR operating at a power of [Proprietary Information] irradiating a target for [Proprietary Information]. The charged target is assumed to contain [Proprietary Information] composed of:

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

Estimates are limited to prediction of actinides and fission products during irradiation of a fresh uranium target containing a limited set of assumed impurities. Calculations for recycled uranium, a broader set of impurities, and potential activation products are not currently available.

The OSTR calculations resulted in an average power per target of [Proprietary Information]. The basis for this extrapolation is discussed in Chapter 4.0, "Radioisotope Production Facility Description," Section 4.2 (biological shielding). Assuming a similar cycle time produces an average target power of [Proprietary Information]. The MURR target (in prototypical reactor locations), radionuclide inventory, and thermal characteristics modeling is underway and will be completed to support the Operating License Application.

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

[Proprietary Information]

Source: .[Proprietary Information]

Figure 5-1. Individual Irradiated Target Heat Generation

[Proprietary Information]

Source: [Proprietary Information]

Figure 5-2. Weekly Irradiated Target Receipt Heat Generation

Figure 5-1 describes the variation of heat generation with decay time for an individual average target irradiated at MURR and OSTR over 1 week. Due to location of the RPF relative to the reactor sites, the minimum decay time for receipt of targets [Proprietary Information]. The combination of reactor source and minimum decay time produces an estimated individual target heat load of [Proprietary Information] or MURR and OSTR irradiated targets, respectively.

The number of irradiated targets received by the RPF in a single week also varies with the source reactor. The MURR operation is based on irradiating eight targets per week, while the OSTR operation is based on irradiating [Proprietary Information]. Figure 5-2 indicates that the total heat load from targets received by the RPF is approximately the same from either reactor as a function of decay time. The weekly heat load from radionuclide decay is estimated at [Proprietary Information]. Therefore, heat load from receipt of MURR targets has been used as an upper bound for irradiated target receipts at the RPF.

5.1.2 Vessels Considered for Thermal Characterization

Thermal characteristics of RPF process vessels are evaluated in NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*. The vessels listed in Table 5-1 were selected to describe the RPF thermal characteristics. The thermal characteristics of every vessel containing radionuclides in the RPF have not been developed by the preliminary evaluation. However, the selected vessels were considered sufficient to span the range of potential heat generation rates anticipated to be contained in process vessels.

Table 5-1. Vessels Selected to Describe Radioisotope Production Facility Thermal Characteristics

Process location	Description
Vessels Equipped with Water-Cooling Jackets	
Dissolver 1/2 (DS-D-100/200) – Start of dissolution cycle	Dissolver vessel after insertion of dissolver basket. This configuration is included for completeness, but is not yet analyzed. Requires consideration of dissolver basket both before and after process solution added to the dissolver containing a dissolver basket.
Dissolver 1/2 (DS-D-100/200) – End of dissolution cycle	Dissolver solution after dissolution complete, prior to combination with transfer flush water. Assumes Kr/Xe and I isotopes transfer to dissolver offgas equipment during dissolution.
Mo system feed tank 1A/1B (MR-TK-100/140)	Dissolver solution after transfer to Mo system feed vessel, but prior to combination with transfer flush water.
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Input from Mo recovery	Process solution after recovery of Mo isotopes from the uranium-bearing process solution.
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Output to uranium recovery	Uranium-bearing process solution input to uranium recovery [Proprietary Information] of decay storage.
Ion exchange feed tank 1 (UR-TK-200)	Process solution feed to the first-cycle uranium ion exchange columns after composition adjustment for ion exchange feed.
High-dose waste concentrate collection tank (WH-TK-240)	Accumulated high-dose liquid waste after concentration by the waste handling system concentrator.
Vessels without Water-Cooling Jackets	
Uranium decay tank (e.g., UR-TK-700A) – Input from separation	Uranium-bearing process solution after separation of uranium from other isotopes.
Uranium decay tank (e.g., UR-TK-700A) – Output to target fabrication	Uranium bearing process solution after separation of uranium from other isotopes and [Proprietary Information].
Solid Transfer Containers (No Cooling Jackets)	
High-dose waste disposal container	High-dose waste concentrate after addition of solidification agent.
Irradiated target in cask at receipt	[Proprietary Information] in annular target cladding on receipt in the transfer cask. Flux based on both internal and external surfaces. Temperature not yet evaluated.
Dissolver basket in air	[Proprietary Information] in dissolver basket for transfer between target disassembly and the target dissolver. Annular configuration between basket wall and lifting post. Flux based on external surface only.

I = iodine.
Kr = krypton.
Mo = molybdenum.
[Proprietary Information]
Xe = xenon.

Three groups of vessels are shown in Table 5-1. The first group contains vessels that include water-cooling jackets to control process solution temperatures. The solution temperature control facilitates solution transfer from one vessel to another, minimizes solution evaporation during storage, or maintains conditions for operation of subsequent unit operations. The second group contains vessels that are not projected to require cooling. The third group contains vessels used for transfer or storage of solid material in air and are not influenced by the cooling water system. Uncooled vessels are included in the evaluation to provide a more complete description of the RPF vessel thermal characteristics.

Heat flux is estimated based on a simple steady-state heat balance for an individual vessel containing a heat-generating material. Only radial heat flow is considered, neglecting heat flow in the axial direction. The simplified heat balance neglects heat losses associated with evaporation of the liquid phase that might be present in the vessel. This type of heat balance is equivalent to modeling each vessel as an unvented vessel, even though most vessels in the RPF will be either open containers or vented by the vessel vent system. The high-dose waste disposal container and irradiated target in the cask at receipt represent the only two process conditions listed in Table 5-1 that are actually closed containers in the RPF.

The irradiated target in the cask at receipt and the dissolver basket in the air process locations are included in Table 5-1, even though the temperatures are not influenced by the coolant system, to indicate vessels will exist with relatively high surface temperatures within the RPF during operation. Estimates of the irradiated target temperature and pressure on receipt at the RPF will be developed as part of the cask licensing activity. Detailed design of the dissolver basket has not been completed. However, a preliminary calculation indicates that a dissolver basket with a lifting post diameter of [Proprietary Information]. The dissolver basket is not currently anticipated to be a completely enclosed vessel with the potential to build pressure on heating. The estimated dissolver basket temperature indicates that the containers of irradiated target material have the potential to achieve relatively high equilibrium temperatures.

5.1.3 Heat Load and Thermal Flux

The volumetric heat load contained by process vessels varies throughout the RPF system as radioisotopes decay, selected radioisotopes are separated, and solution compositions are adjusted by the unit operations. Conservatism is included in the thermal flux estimate by assuming heat transfer is limited to a radial direction and neglecting heat loss from solution evaporation. Table 5-2 provides estimates of the volumetric heat load and radial thermal flux at the containment apparatus wall for selected vessels where cooling water is used to control the process solution temperature shown in Table 5-1. Table 5-3 provides similar estimates for selected vessels where cooling water is not provided to control the process solution temperature. The vessels listed in Table 5-2 and Table 5-3 were selected to indicate the range of conditions experienced as process solution is transferred through the RPF process equipment.

Table 5-2. Heat Load and Thermal Flux for Selected Water-Cooled Vessels

Process Location	Thermal characteristics: Uranium g U/L	Decay time after EOI	Heat load W/L (W/g U)	Vessel diameter cm (in.)	Radial thermal flux W/cm^2 (BTU/hr-ft^2)
Dissolver 1/2 (DS-D-100/200) – Start of dissolution cycle[a]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Dissolver 1/2 (DS-D-100/200) – End of dissolution cycle	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Mo system feed tank 1A/1B (MR-TK-100/140)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Input from Mo recovery	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Output to uranium recovery	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Ion exchange feed Tank 1 (UR-TK-200)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
High dose waste concentrate collection tank (WH-TK-240)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

Source: NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

[a] Not evaluated by this calculation. The simplified evaluation methodology was not considered applicable.

[b] High-dose waste vessels collect waste from multiple weeks of process operation that is dominated by a [Proprietary Information] time period. Current plans are based on collecting high-dose waste as concentrate from [Proprietary Information] Optimization may allow extension of the waste collection time period. Evaluation indicates the accumulated high-dose waste heat load approaches an asymptote of [Proprietary Information] after accumulating waste from [Proprietary Information].

[c] Based on high-dose waste concentrate tank that is 80% full, containing [Proprietary Information] of heat-generating isotopes.

EOI = end of irradiation.
Mo = molybdenum.
N/A = not applicable.
TBD = to be determined.

Table 5-3. Heat Load and Thermal Flux for Selected Vessels without Water Cooling

Process location	Thermal characteristics: Uranium g U/L	Decay time after EOI	Heat load W/L (W/g U)	Vessel diameter cm (in.)	Radial thermal flux W/cm^2 (BTU/hr-ft^2)
Uranium decay tank (e.g., UR-TK-700A) – Input from separation	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Uranium decay tank (e.g., UR-TK-700A) – Output to target fabrication	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
High-dose waste disposal container	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

Source: NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

[a] Not evaluated by this calculation. The simplified evaluation methodology was not considered applicable.

[b] High-dose waste vessels collect waste from multiple weeks of process operation that is dominated by a [Proprietary Information] time period. Current plans are based on collecting high-dose waste as concentrate from [Proprietary Information] Optimization may allow extension of the waste collection time period. Evaluation indicates the accumulated high-dose waste heat load approaches an asymptote of [Proprietary Information] after accumulating waste from [Proprietary Information].

[c] Based on high-dose waste concentrate tank that is 80% full, containing [Proprietary Information] of heat-generating isotopes.

EOI = end of irradiation.
N/A = not applicable.

The heat load of process solutions in unit operations prior to the start of separating uranium from other radionuclides can be characterized by the solution uranium concentration. Planned operating conditions are used to support characterization of the thermal heat load. The process solution uranium concentration at the end of the dissolution cycle [Proprietary Information] is estimated based on the mass of uranium input by a dissolver basket, combined with the volume of acid charged to the dissolver. The resultant dissolver solution is transferred to the molybdenum (Mo) system feed tank, and thermal characteristics are evaluated neglecting mixing with dissolver vessel flush solutions. The uranium concentration in the impure uranium collection tanks [Proprietary Information], ion exchange feed tank [Proprietary Information], and uranium decay tanks [Proprietary Information] represent goal compositions for process solutions during operation.

Three radionuclide decay times, summarized below, are used to describe the RPF thermal characteristics based on currently planned decay limits within the process operation:

- A decay time of [Proprietary Information] (minimum decay time for targets in transfer casks received at the RPF outer door) is used to describe process solutions in the dissolver, Mo system feed tanks, and solution transferred into the impure uranium collection tanks, neglecting the time required for cask receipt, target disassembly, and target dissolution.
- A decay time of [Proprietary Information] (minimum decay time required to control in-growth of plutonium-239 [^{239}Pu] in recycled uranium after separations) is used to describe process solution at the end of the impure uranium collection tank storage period, solution in ion exchange feed tank 1, solution transferred into the uranium decay tanks, and waste entering the high-dose waste concentrate collection tank, neglecting time required to complete separation activities.
- A decay time of [Proprietary Information] in recycled uranium, allowing contact operation and maintenance in the target fabrication system, is used to describe the process solution at the end of the storage period in the uranium decay tanks.

Target heat generation (shown by Figure 5-1) is placed on a unit uranium mass basis to support the estimate of heat load in the selected vessels. The unit uranium mass input is modified to approximate the impact of radionuclide separations that occur in unit operations. The unit mass heat generation shown in Table 5-2 for a dissolver vessel at the start of dissolution cycle [Proprietary Information] represents material containing all radionuclides in an irradiated target [Proprietary Information]. All isotopes of krypton (Kr), xenon (Xe), and iodine (I) are assumed to be evolved to the dissolver offgas system during dissolution, reducing the unit mass heat generation to [Proprietary Information]. The molybdenum isotopes are assumed to be separated from the dissolver solution by the Mo recovery and purification system, reducing the unit mass heat generation to [Proprietary Information] for solution entering the impure uranium collection tanks. The unit mass heat generation is reduced to [Proprietary Information] U after solution in the impure uranium collection tanks is decayed to [Proprietary Information].

The thermal characteristics of recycled uranium process solution after separation in the uranium recovery and recycle system are shown in Table 5-3. Minimal separation of neptunium from uranium is projected to be obtained by the process, and the heat load is approximated by a unit mass heat generation dominated by the isotopes of neptunium and uranium [Proprietary Information] entering the uranium decay tanks. The unit mass heat generation of recycled uranium solution transferred to target fabrication is reduced to [Proprietary Information].

The thermal characteristics of waste handling vessels are not characterized by the process solution uranium concentration and are expected to collect solution containing radionuclides from multiple weeks of operation. The waste handling vessel thermal characteristics are described by the high-dose waste vessels that contain a majority of the waste radionuclides. Weekly input to the high-dose waste vessels is dominated by wastes from the uranium recovery and recycle separation system and described by radionuclides in a target decayed to [Proprietary Information]with isotopes of Kr, Xe, I, and Mo removed.

Accumulation of waste from a week of operations increases the waste vessel heat load that decreases by decay while awaiting waste input from a subsequent week of operation. Current plans are based on accumulating waste from [Proprietary Information]. However, system optimization may increase the goal high-dose waste accumulation time period. Evaluation of the heat load sequence indicates that the waste heat load approaches an asymptote of [Proprietary Information]. Therefore, the waste vessel heat loads were characterized by [Proprietary Information] contained in the vessel capacity using current estimates of the vessel dimensions.

5.1.4 Maximum Vessel Temperature and Pressure Estimates

An estimate of vessel temperature has been obtained using an overall heat transfer coefficient obtained from handbook values for a tank on legs containing water and an assumed cell air temperature of 35°C (95°F). Temperatures are estimated assuming no water-cooling system is active, and pressures are estimated assuming each vessel is unvented to approximate maximum values. Note that the preliminary estimate assumes that radial temperature variations within the generating heat material are not significant, which may be appropriate for vessels containing liquids, but could be questionable for containers of heat-generating solids.

The vapor pressure of water at the estimated vessel temperature is used to approximate the maximum pressure. The vapor pressure of water was considered a conservative estimate of the pressure developed within a process apparatus, as the total vapor pressure of a solution is decreased by the addition of nitric acid or uranyl nitrate to the liquid phase.

Table 5-4 provides estimates of the maximum temperature and pressure predicted for selected vessels where cooling water is used to control the process solution temperature shown in Table 5-1. Table 5-5 provides similar estimates for uncooled vessels and the high-dose waste disposal container.

The maximum temperature and pressure that could be observed in representative vessels without operation of the coolant system is shown in Table 5-4 as [Proprietary Information], absolute for the Mo system feed tanks. However, the evaluation approach was not considered applicable to the vessel configuration representing a dissolver at the start of the dissolver cycle. This configuration has the potential to produce higher temperatures and pressures than the vessels that could be evaluated using the current approach.

Table 5-4. Estimate of Maximum Temperature and Pressure in Water-Cooled Vessels

Process location	Radial thermal flux,[a] BTU/hr-ft^2	Maximum heat transfer surface temperature[b] °C (°F)	Estimated maximum unvented vessel pressure[c] lb/in^2, absolute
Dissolver 1/2 (DS-D-100/200) – Start of dissolution cycle[d]	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Dissolver 1/2 (DS-D-100/200) – End of dissolution cycle	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Mo system feed tank 1A/1B (MR-TK-100/140)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Input from Mo recovery	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Impure uranium collection tanks (e.g., UR-TK-100A/B) – Output to U recovery	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Ion exchange feed tank 1 (UR-TK-200)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
High-dose waste concentrate collection tank (WH-TK-240)	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

Source: NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015

[a] Radial thermal flux from Table 5-2.

[b] Maximum heat transfer surface temperature assuming overall heat transfer coefficient at walls of 1.8 BTU/hr-ft^2-°F and ambient cell air temperature of 35°C (95°F).

[c] Unvented vessel pressure based on water vapor pressure at the maximum heat transfer surface temperature. Actual estimated water vapor pressure shown in parentheses for pressures less than 14.7 lb/in.^{2}, absolute.

[d] Not evaluated by this calculation. The simplified methodology was not considered applicable.

Mo = molybdenum.
TBD = to be determined.
U = uranium.

Table 5-5. Estimate of Maximum Temperature and Pressure in Vessels without Water Cooling

Process location	Radial thermal flux[a] BTU/hr-ft^2	Maximum heat transfer surface temperature[b] °C (°F)	Estimated maximum unvented vessel pressure[c] lb/in^2, absolute
Uranium decay tank (e.g., UR-TK-700A) – Input from separation	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
Uranium decay tank (e.g., UR-TK-700A) – Output to target fabrication	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]
High-dose waste disposal container	[Proprietary Information]	[Proprietary Information]	[Proprietary Information]

Source: NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

[a] Radial thermal flux from Table 5-3.

[b] Maximum heat transfer surface temperature assuming overall heat transfer coefficient at walls of 1.8 BTU/hr-ft^2-°F and ambient cell air temperature of 35°C (95°F).

[c] Unvented vessel pressure based on water vapor pressure at the maximum heat transfer surface temperature. Actual estimated water vapor pressure shown in parentheses for pressures less than 14.7 lb/in.^{2}, absolute.

5.1.5 Potential Impact of Overcooling Process Solutions

Overcooling of uranium-bearing process solutions has the potential to [Proprietary Information]. Precipitation as a solid form effectively increases the uranium concentration of material contained by a process vessel and potentially results in a nuclear criticality. The [Proprietary Information]. Criticality evaluations are described in the following three documents for current equipment configurations of the irradiated target disassembly/dissolution, target fabrication, and uranium recycle separation systems, respectively.

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

The impact of uranium precipitation upset conditions on nuclear criticality calculations was evaluated by interspersing selected tanks containing [Proprietary Information]among vessels containing uranium at a conservative nominal process concentration. The results indicate that precipitation upset conditions are predicted to remain below an upper subcritical limit of [Proprietary Information] for the configurations evaluated. Therefore, overcooling process solutions is not predicted to pose a nuclear criticality hazard for the current RPF equipment configuration.

5.1.6 Potential Impact on Gas Management System

Coolant system operation has the potential to impact the performance of the gas management system cooled sections. The primary gas management system cooled section controls the decay time provided for noble gases (isotopes of Kr and Xe) by holdup in the dissolver offgas system. The maximum hypothetical accident evaluated in Chapter 13.0, "Accident Analysis," Section 13.2.1 indicates the dose consequences from a bounding release of Kr and Xe isotopes alone is less than 0.15 roentgen equivalent in man (rem). The bounding release of noble gases is less than the performance requirement of 5 rem for an intermediate consequence event defined in Title 10, *Code of Federal Regulations*, Part 70.61, "Performance Requirements" (10 CFR 70.61). Therefore, the cooling water system is not considered to be an item relied on for safety (IROFS) based on the potential impact on the gas management systems.

5.1.7 Conclusion

The evaluation focused on vessels equipped with water cooling jackets. Typical process vessels of a pencil tank configuration are anticipated to be constructed from material similar to Schedule 40 stainless steel pipe. The pressure rating of seamless standard stainless steel pipe ranges from:

- 4-in., Schedule 40 – ~1,500 to 900 lb/in^2, gauge for 37.8 to 398.9°C (100 to 750°F), respectively
- 5-in., Schedule 40 – ~1,350 to 800 lb/in^2, gauge for 37.8 to 398.9°C (100 to 750°F), respectively
- 6-in., Schedule 40 – ~1,200 to 725 lb/in^2, gauge for 37.8 to 398.9°C (100 to 750°F), respectively

The maximum temperature and pressure in vessels without cooling and ventilation is estimated at [Proprietary Information] in Table 5-4 in the Mo system feed tanks, which are projected to be 5-in. diameter pencil tanks.

The high-dose concentrate collection tank is a standard tank design such that the stainless steel pipe comparison is not applicable. Maximum temperature and pressure for this vessel is estimated at [Proprietary Information]. Standard tank designs are capable of containing process solution at the high-dose concentrate collection tank conditions.

Based on the above comparisons, the maximum temperature and pressure within RPF vessels are anticipated to not result in failure of a process apparatus, and the cooling water system is not selected as an IROFS. The approach used to evaluate vessels was not considered applicable to a dissolver at the start of a dissolver cycle (non-uniform distribution of the heat-generating material). Future evaluation of this vessel configuration has the potential to impact the importance of the coolant system.

5.2 COOLANT SYSTEMS DESCRIPTION

The above analysis and description show that the cooling water system is designed such that the system will function in a manner, whether operational or not, consistent with occupational safety and protection of the public and environment. Therefore, the cooling function is not considered an IROFS. A description of the coolant systems for the RPF is provided in Chapter 9.0, "Auxiliary Systems," Section 9.7.

5.3 REFERENCES

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

NWMI-2013-CALC-003, *Interim Approximation of MURR Irradiated Target Isotope Inventory*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, May 2013.

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NWMI-2015-CRITCALC-002, *Irradiated Target Low-Enriched Uranium Material Dissolution*, Rev. 0, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2014.

NWMI-2015-CRITCALC-005, *Target Fabrication Tanks, Wet Processes, and Storage*, Rev. 0, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2014.

NWMI-2015-CRITCALC-006, *Tank Hot Cell*, Rev. 0, Northwest Medical Isotopes, LLC, Corvallis, Oregon, February 2015.

OSU-RC-1301, *Preliminary Isotope Inventory Estimate of the ^{99}Mo Targets After Irradiation In the Oregon State TRIGA® Reactor*, Rev. 1, Oregon State University, Corvallis, Oregon, March 2013.

Chapter 6.0 – Engineered Safety Features

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0
June 2015

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Chapter 6.0 – Engineered Safety Features

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0

Date Published:
June 29, 2015

Document Number: **NWMI-2013-021**	*Revision Number*: **0**
Title: **Chapter 6.0 – Engineered Safety Features Construction Permit Application for Radioisotope Production Facility**	
Approved by: **Carolyn Haass**	*Signature*: Carolyn C Haass

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REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required

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FIGURES

TABLES

TERMS

Acronyms and Abbreviations

^{99}Mo	molybdenum-99
^{235}U	uranium-235
ADUN	acid-deficient uranium nitrate
AEC	active engineered control
ANECF	average neutron energy causing fission
ANS	American Nuclear Society
ANSI	American National Standards Institute
CAAS	criticality accident alarm system
CFR	Code of Federal Regulations
CSE	criticality safety evaluation
DBE	design basis earthquake
HEGA	high-efficiency gas adsorber
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air conditioning
IEU	intermediate-enriched uranium
IX	ion exchange
IROFS	item relied on for safety
Kr	krypton
LEU	low-enriched uranium
Mo	molybdenum
NO_2	nitrogen dioxide
NO_x	nitrogen oxide
NWMI	Northwest Medical Isotopes, LLC
PEC	passive engineered control
PHA	preliminary hazards analysis
RPF	radioisotope production facility
SSC	structures, systems, and components
SPL	single parameter limit
UN	uranium nitride
[Proprietary Information]	[Proprietary Information]
USL	upper subcritical limits
Xe	xenon

Units

°C	degrees Celsius
°F	degrees Fahrenheit
atm	atmosphere
cm	centimeter
cm^3	cubic centimeter
ft	feet
ft^2	square feet
ft^3	cubic feet
g	gram
hr	hour
in.	inch
L	liter
m	meter
m^2	square meter
min	minute
mL	milliliter
mol	mole
rad	radiation absorbed dose
wt%	w percent
yr	year

6.0 ENGINEERED SAFETY FEATURES

6.1 SUMMARY DESCRIPTION

Engineered safety features are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to workers, the public, and environment within acceptable values. The engineered safety features associated with confinement of the process radionuclides and hazardous chemicals for the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) are summarized in Table 6-1, including the accidents mitigated; structures, systems, and components (SSC) used to provide the engineered safety features; and references to subsequent sections providing a more detailed engineered safety feature description.

Confinement is a general engineered safety feature that is credited as being in place as part of the preliminary hazards analysis (PHA) described in Chapter 13.0, "Accident Analysis." Additional items relied on for safety (IROFS) associated with the confinement system were derived from the accident analyses in Chapter 13.0. The derived IROFS are also listed in Table 6-1, with reference to more detailed descriptions in Section 6.2.1.

The current design approach does not anticipate requiring containment or an emergency cooling system as engineered safety features, as discussed in Sections 6.2.2 and 6.2.3.

Nuclear criticality safety is discussed in Section 6.3. Criticality safety controls are described in Section 6.3.1. The currently defined criticality safety controls are derived from a combination of preliminary criticality safety evaluations (CSE) and accident analyses, which are described in Chapter 13.0. The criticality safety analyses produce a set of features needed to satisfy the double-contingency requirements for nuclear criticality control. These features are evaluated by major systems within the RPF and listed by major system in Section 6.3.1.1, Table 6-6 through Table 6-13. The accident analyses in Chapter 13.0 identify IROFS for the prevention of nuclear criticality, which are summarized in Table 6-2, with reference to more detailed descriptions in Section 6.3.1.2.

Table 6-1. Summary of Confinement Engineered Safety Features (2 pages)

Engineered safety feature	IROFS	Accident(s) mitigated	SSCs providing engineered safety features	Detailed description section
Confinement includes: • Hot cell liquid confinement boundary • Hot cell secondary confinement boundary • Hot cell shielding boundary	 RS-01 RS-03 RS-04	• Equipment malfunction and/or maintenance • Hazardous chemical spills	• Confinement enclosures including penetration seals • Zone I exhaust ventilation system, including ducting, filters, and exhaust stack • Zone I inlet ventilation system, including ducting, filters, and bubble-tight isolation dampers • Ventilation control system • Secondary iodine removal bed • Berms	6.2.1.1 through 6.2.1.6
Confinement IROFS Derived from Accident Analyses and Potential Technical Specifications				
Primary offgas relief system	RS-09	Dissolver offgas failure during dissolution operation	• Pressure relief device • Pressure relief tank	6.2.1.7.1
Active radiation monitoring and isolation of low-dose waste transfer	RS-10	Transfer of high-dose process liquid outside the hot cell shielding boundary	Radiation monitoring and isolation system for low-dose liquid transfers	6.2.1.7.2
Cask local ventilation during closure lid removal and docking preparations	RS-13	Target cladding leakage during shipment	Local capture ventilation system over closure lid during lid removal	6.2.1.7.3
Cask docking port enabler	RS-15	Cask not engaged in cask docking port prior to opening docking port door	Sensor system controlling cask docking port door operation	6.2.1.7.4
Process vessel emergency purge system	FS-03	SSC damage due to hydrogen deflagration or detonation	Backup bottled nitrogen gas supply	6.2.1.7.5
Irradiated target cask lifting fixture	FS-04	Dislodging the target cask shield plug while workers present during target unloading activities	• Cask lifting fixture design that prevents cask tipping • Cask lifting fixture design that prevents lift from toppling during a seismic event	6.2.1.7.6
Exhaust stack height	FS-05	• Equipment malfunction resulting in liquid spill or spray • Carbon bed fire	• Zone I exhaust stack	6.2.1.7.7

Table 6-1. Summary of Confinement Engineered Safety Features (2 pages)

Engineered safety feature	IROFS	Accident(s) mitigated	SSCs providing engineered safety features	Detailed description section
Double-wall piping	CS-09	Solution spill in facility area where spill containment berm is neither practical nor desirable for personnel chemical protection purposes	Double-wall piping for selected transfer lines	6.2.1.7.7
Backflow prevention devices	CS-18	High worker exposure from backflow of high-dose solution	Backflow prevention devices located on process lines crossing the hot cell shielding boundary	0
Safe geometry day tanks	CS-19			
Dissolver offgas iodine removal unit[a]	–	• Potential limiting control for operation • Primary iodine control system during normal operation	Dissolver offgas iodine removal units (DS-SB-600A/B/C)	6.2.1.8
Dissolver offgas primary adsorber[a]	–	• Potential limiting control for operation • Primary noble gas control system during normal operation	Dissolver offgas primary adsorber units (DS-SB-620A/B/C)	6.2.1.8.2
Dissolver offgas vacuum receiver or vacuum pump[a]	–	• Potential limiting control for operation • Motive force for dissolver offgas	• Dissolver offgas vacuum receiver tanks (DS-TK-700A/B) • Dissolver offgas vacuum pumps (DS-P-710A/B)	6.2.1.8.3

[a] Examples of candidate technical specification rather than engineered safety feature.

IROFS = item relied on for safety. SSC = structures, systems, and components.

Table 6-2. Summary of Criticality Engineered Safety Features (2 pages)

Engineered safety feature	IROFS	SSC features providing engineered safety features	Detailed description section
Interaction control spacing provided by passively designed fixtures and workstation placement	CS-04	Defines spacing between SSC components using geometry to prevent nuclear criticality	6.3.1.2.1
Pencil tank, vessel, or piping safe geometry confinement using the diameter of tanks, vessels, or piping	CS-06	Defines dimensions of SSCs using geometry to prevent nuclear criticality	6.3.1.2.2
Pencil tank geometry control on fixed interaction spacing of individual tanks	CS-07	Defines spacing between different SSCs using geometry to prevent nuclear criticality	6.3.1.2.3
Floor and sump geometry control on slab depth, and sump diameter or depth for floor dikes	CS-08	Defines sump geometry and dimensions for SSCs using geometry to prevent nuclear criticality	6.3.1.2.4

Table 6-2. Summary of Criticality Engineered Safety Features (2 pages)

Engineered safety feature	IROFS	SSC features providing engineered safety features	Detailed description section
Double-wall piping	CS-09	Defines transfer line leak confinement in locations where sumps under piping are neither feasible nor desirable	6.3.1.2.5
Closed safe-geometry heating or cooling loop with monitoring and alarm	CS-10	Closed-loop heat transfer fluid systems to prevent nuclear criticality or transfer of high-dose material across shielding boundary in the event of a leak into the heat transfer fluid	6.3.1.2.6
Simple overflow to normally empty safe-geometry tank with level alarm	CS-11	Overflow to prevent nuclear criticality from fissile solution entering non-geometrically favorable ventilation equipment	6.3.1.2.7
Condensing pot or seal pot in ventilation vent line	CS-12	Seal pots to prevent nuclear criticality from fissile solution entering non-geometrically favorable ventilation equipment	6.3.1.2.8
Simple overflow to normally empty safe geometry floor with level alarm in the hot cell containment boundary	CS-13	Overflow to prevent nuclear criticality from fissile solution entering non-geometrically favorable ventilation equipment	6.3.1.2.9
Active discharge monitoring and isolation	CS-14	Information to be provided in the Operating License Application	6.3.1.2.10
Independent active discharge monitoring and isolation	CS-15	Information will be provided in the Operating License Application	6.3.1.2.11
Backflow prevention device	CS-18	Backflow prevention to preclude fissile or high dose solution from crossing shielding boundary to non-geometrically favorable chemical supply tanks and prevent nuclear criticality	6.3.1.2.12
Safe geometry day tanks	CS-19	Alternate backflow prevention device	6.3.1.2.13
Evaporator or concentrator condensate monitoring	CS-20	Prevent nuclear criticality from high-volume transfer to non-geometrically favorable vessels in solutions with normally low fissile component concentrations	6.3.1.2.14
Processing component safe volume confinement	CS-26	Defines volume of SSCs to prevent nuclear criticality	6.3.1.2.15
Closed heating or cooling loop with monitoring and alarm	CS-27	Closed-loop, high-volume heat transfer fluid systems to prevent nuclear criticality or transfer of high-dose material across shielding boundary in the event of a leak into the heat transfer fluid with normally low fissile component concentrations	6.3.1.2.16

IROFS = item relied on for safety. SSC = structures, systems, and components.

6.2 DETAILED DESCRIPTIONS

The PHA used to identify accidents in Chapter 13.0, Section 13.1.3, assumed the following known and credited safety features, or IROFS, are in place for normal operations:

- Hot cell shielding boundary, credited for shielding workers and the public from direct exposure to radiation (a normal hazard of the operation)
- Hot cell confinement boundaries, credited for confining the fissile and high-dose solids, liquids, and gases, and controlling gaseous releases to the environment
- Administrative and passive design features on uranium batch, volume, geometry, and interaction controls on the activities, credited for maintaining normal operations involving the handling of fissile material subcritical (the PHA identified initiators for abnormal operations that require further evaluation for IROFS satisfying the double-contingency principle)

This section provides detailed descriptions of the engineered safety features identified by the accident analyses shown in Chapter 13.0.

6.2.1 Confinement

The PHA was based on a definition for confinement, as follows:

> **Confinement** – An enclosure of the facility (e.g., the hot cell area in the RPF) that is designed to limit the exchange of effluents between the enclosure and its external environment to controlled or defined pathways. A confinement should include the capability to maintain sufficient internal negative pressure to ensure inleakage (i.e., prevent uncontrolled leakage outside the confined area), but need not be capable of supporting positive internal pressure or significantly shielding the external environment from internal sources of direct radiation. Air movement in a confinement area could be integrated into the heating, ventilation, and air conditioning (HVAC) systems, including exhaust stacks or vents to the external environment, filters, blowers, and dampers (ANSI/ANS-15.1, *The Development of Technical Specifications for Research Reactors*).

Confinement describes the low-leakage boundary surrounding radioactive or hazardous chemical materials released during an accident to facility regions surrounding the physical process equipment containing process materials. The confinement systems localize releases of radioactive or hazardous materials to controlled areas and mitigate the consequences of accidents.

The principal design and safety objective of the confinement system is to protect on-site workers, the public, and environment. Personnel protection control features (e.g., adequate shielding and ventilation control) will minimize hazards normally associated with radioactive or chemical materials.

The second design objective is to minimize the reliance on administrative or complex active engineering controls and provide a confinement system that is as simple and fail-safe as reasonably possible.

This subsection describes the confinement systems for the RPF. The RPF confinement areas will consist of hot cell and glovebox enclosures housing process operations, tanks, and piping. Confinement will be provided by a combination of the enclosure boundaries (e.g., walls, floor, and ceiling), enclosure ventilation, and ventilation control system. The enclosure boundaries will restrict bulk quantities of process materials, potentially present in solid or liquid forms, to the confinement and limit in-leakage of gaseous components controlled by the ventilation system. The ventilation and ventilation control systems will restrict the gaseous components (including gas phase components and solid/liquid dispersions) to the confinement. Figure 6-1 provides a simplified schematic of the confinement ventilation system, which is described in more detail as the Zone I ventilation system in Chapter 9.0, "Auxiliary Systems."

[Proprietary Information]

Source: Figure 2-5 of NWMI-2015-SDD-013, *System Design Description for Ventilation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

Figure 6-1. Simplified Zone I Ventilation Schematic

The enclosure boundary of the hot cells will also function as biological shielding for operating personnel. Shielding functions of the hot cells are discussed in Chapter 4.0, "Radioisotope Production Facility Description."

Hazardous chemical confinement will be provided by berms located within the RPF to confine spilled material to the vicinity where a spill may originate.

6.2.1.1 Confinement System

Confinement system enclosure structures, ventilation ducting, isolation dampers, and Zone I exhaust filter trains are designated as IROFS. Table 6-3 provides a description of the system component safety functions. Figure 6-2, Figure 6-3, and Figure 6-4 indicate the general location of confinement structure boundaries to the facility ground level, mechanical level, and lower level layouts, respectively. The confinement system is an engineered safety feature that performs the functions identified by IROFS RS-01, RS-03, and RS-04 in Chapter 13.0.

Table 6-3. Confinement System Safety Functions

System, structure, component	Description	Classification
Zone I enclosure inlet isolation dampers and ducting leading from isolation dampers to enclosures	Provide confinement isolation at Zone I/Zone II enclosure boundaries	IROFS
Zone I enclosure exhaust ducting leading from enclosures to the exhaust stack, filters, and exhaust stack	Provides confinement to the confinement exhaust boundary	IROFS
Process vessel vent exhaust ducting leading from process vessels to Zone I exhaust plenum	Provides confinement to the confinement exhaust boundary	IROFS
Ventilation control system	Provides stack monitoring and interlocks to monitor discharge and signal changing on service filter trains during normal and abnormal operation	IROFS
Secondary iodine removal bed	Mitigates a release of the iodine inventory in the dissolver offgas treatment system	IROFS
Hot cells, tank vaults, and glovebox enclosure structures	Provide solid, liquid, gas confinement	IROFS

IROFS = item relied on for safety.

[Proprietary Information]

Source: Figure 2-1 of NWMI-2015-SDD-013, *System Design Description for Ventilation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

Figure 6-2. Ground Level Confinement Boundary

[Proprietary Information]

Source: Figure 2-2 of NWMI-2015-SDD-013, *System Design Description for Ventilation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

Figure 6-3. Mechanical Level Confinement Boundary

[Proprietary Information]

Source: Figure 2-3 of NWMI-2015-SDD-013, *System Design Description for Ventilation*, Rev. A, Northwest Medical Isotopes, LLC, Corvallis, Oregon, March 2015.

Figure 6-4. Lower Level Confinement Boundary

During normal operation, passive confinement is provided by the contiguous boundary between the hazardous materials and the surrounding environment and is credited with confining the hazards generated as a result of accident scenarios. The boundary includes the enclosure structures and extension of the structures through the Zone I ventilation components. The intent of the passive boundary is to confine hazardous materials while also preventing disturbance of the hazardous material inventory by external energy sources. This passive confinement boundary extends from the isolation valve downstream of the intake high-efficiency particulate air (HEPA) filter to the exhaust stack.

An event that results in a release of process material to a confinement enclosure will be confined by the enclosure structural components. Each process line that connects with vessels located outside of a confinement boundary with vessels located inside a confinement boundary will be provided with backflow prevention devices to prevent releases of gaseous or liquid material. The backflow prevention devices on piping penetrating the confinement boundary are designed as passive devices and will be located as near as practical to the confinement boundary or take a position that provides greater safety on loss of actuating power.

The consequences of an uncontrolled release within a confinement enclosure, and the off-site consequences of releasing fission products through the ventilation system, will be mitigated by use of an active component in the form of bubble-tight isolation dampers as IROFS on the inlet ventilation ducting to each enclosure.

This engineered safety feature reduces the ducting to the confinement volume that needs to remain intact to achieve enclosure confinement. The dampers will close automatically (fail-closed) on loss of power, and the ventilation system will automatically be placed into the passive ventilation operating mode.

Overall performance assurance of the active confinement components will be achieved through factory testing and in-place testing. Duct and housing leak tests will be performed in accordance with minimum acceptance criteria, as specified in ASME AG-1, *Code on Nuclear Air and Gas Treatment.* Specific owner requirements with respect to acceptable leak rates will be based on the safety analysis.

Berms will employ a passive confinement methodology. Passive confinement will be achieved through a continuous boundary between the hazardous materials and the surrounding area. In the event of an accidental release, the hazardous liquid will be confined to limit the exposed surface area of the liquid.

6.2.1.2 Accidents Mitigated

The hot cell confinement system and shielding boundary are credited as being in place by the accident analysis in Chapter 13.0, Section 13.1.3.1. Accidents mitigated consist of equipment malfunction events that result in the release of radioactive material or hazardous chemicals to a confinement enclosure. The confinement system is also credited with mitigating the impact of a non-specific initiating event resulting in release of the iodine inventory in the dissolver offgas treatment system.

6.2.1.3 Functional Requirements

Functional requirements of the confinement structural components include:

- Capturing and containing liquid or solid releases to prevent the material from exiting the boundary and causing high dose to a worker or member of the public or producing significant environment contamination
- Preventing spills or sprays of radioactive solution that are acidic or caustic from causing adverse exposure to personnel through direct contact with skin, eyes, and mucus membranes where the combination of chemical exposure and radiological contamination would lead to serious injury and long-lasting effects

Functional requirements of the confinement ventilation components include:

- Providing negative air pressure in the hot cell (Zone I) relative to lower zones outside of the hot cell using exhaust fans equipped with HEPA filters and high-efficiency gas adsorbers (HEGA) to reduce the release of radionuclides (both particulate and gaseous) outside the primary confinement boundary to below Title 10, *Code of Federal Regulations*, Part 20, "Standards for Protection Against Radiation" (10 CFR 20) release limits during normal and abnormal operations.
- Mitigating high-dose radionuclide releases to maintain exposure to acceptable levels to workers and the public in a highly reliable and available manner. The hot cell secondary confinement boundary will perform this function using a system of passive and active engineered features to ensure a high level of reliability and availability.
- Removing iodine isotopes present in the process vessel vent under accident conditions to comply with 10 CFR 70.61, "Performance Requirements," for an intermediate consequence release.

Berms confining potential hazardous chemical spills are designed to hold the entire contents of the container in the event the container fails.

6.2.1.4 Confinement Components

The following components are associated with the confinement barriers of the hot cells, tank vaults, and gloveboxes. The specific materials, construction, installation, and operating requirements of these components are evaluated based on the safety analysis.

Confinement structural components include the following.

- Sealed flooring will provide multiple layers of protection from release to the environment.
- Diked areas will contain specific releases. Sumps of appropriate design will be provided with remote operated pumps to mitigate liquid spills by capturing the liquid in appropriate safe-geometry tanks.
- In the molybdenum-99 (^{99}Mo) purification clean room, smaller confinement catch basins will be provided under points of credible spill potential in addition to the sealed floor.
- Entryway doors into a designated liquid confinement area will be sealed against credible liquid leaks to outside the boundary.
- Piping penetrations and air ducts will be located to minimize the potential for liquid leaks across the confinement boundary.

Ventilation system components that are credited include the following.

- Zone I inlet HEPA filters will provide an efficiency of greater than 99.9 percent for removal of radiological particulates from the air that may reverse flow from Zone I to Zone II.
- Zone I ducting will ensure that negative air pressure can be maintained by conveying exhaust air to the stack.
- Bubble-tight dampers will be provided to comply with the requirements of ASME AG-1, Section DA-5141. Ventilation ductwork and ductwork support materials will meet the requirements of ASME AG-1. Supports will be designed and fabricated in accordance with the requirements of ASME AG-1.
- Zone I exhaust train HEPA filters will provide an efficiency of greater than 99.95 percent for removal of radiological particulates from the air that flows to the stack.
- Zone I exhaust train HEGA filters will provide an efficiency of greater than 90% for removal of iodine.
- The Zone I exhaust stack will provide dispersion of radionuclides in normal and abnormal releases at a discharge point of 23 meters (m) (75 feet [ft]) above the building ground level.
- Stack monitoring and interlocks will monitor discharge and signal changing of service filter trains during normal and abnormal operations.

Secondary process offgas treatment iodine removal beds (VV-SB-520) will mitigate an iodine release.

6.2.1.5 Test Requirements

Engineered safety features will be tested to ensure that components maintain operability and can provide adequate confidence that the safety system performs satisfactorily during postulated events. The confinement engineered safety features that initiate the system interlocks are designed to permit testing during plant operation.

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.6 Design Basis

Codes and standards are discussed in Chapter 3.0, "Design of Structures, Systems, and Components." The design bases for Zone I and Zone II ventilation systems are described in Chapter 9.0. The design basis of confinement enclosure structures is described in Chapter 4.0. Chapter 7.0, "Instrumentation and Control Systems," identifies the engineered safety feature-related design basis of the ventilation control system.

The following information was developed for the Construction Permit Application to describe the process offgas secondary iodine removal bed:

- Sorbent bed of [Proprietary Information]
- Iodine removal efficiency greater than [Proprietary Information]
- Nominal superficial gas flow velocity of [Proprietary Information]
- Nominal sorbent bed operating temperature of less than [Proprietary Information]
- Nominal sorbent bed depth of [Proprietary Information]
- Nominal gas relative humidity less than [Proprietary Information]

Additional detailed information on the process offgas iodine retention bed design basis will be developed for the Operating License Application.

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the RPF confinement systems and components are discussed in Chapter 14.0, "Technical Specifications."

6.2.1.7 Derived Confinement Items Relied on for Safety

The following subsections describe additional engineered safety features that are derived from the accident analyses described in Chapter 13.0 and are projected technical specifications defining limited conditions for operation.

6.2.1.7.1 IROFS RS-09, Primary Offgas Relief System

IROFS RS-09, "Primary Offgas Relief System," is identified by the accident analysis in Chapter 13.0. As an active engineered control (AEC), the primary offgas relief system will be a component included in the offgas train for the two irradiated target dissolvers. The dissolver offgas system is intended to operate at a pressure that is less than the confinement enclosures to maintain gaseous components generated during dissolution within the vessels and route the gaseous components through the offgas treatment unit operations. The primary offgas relief system, or pressure relief tank, will be used to confine gases to the dissolver and a portion of the dissolver offgas equipment, if the offgas motive force (vacuum pumps) ceases operation during dissolution of a dissolver batch.

Figure 6-5 is a diagram of the dissolver offgas system process, which shows the pressure relief tank position in the offgas treatment equipment train. Figure 6-6 shows the location of the pressure relief tank within the RPF hot cell (identified as "pressure relief").

[Proprietary Information]

Figure 6-5. Dissolver Offgas System Engineered Safety Features

[Proprietary Information]

Figure 6-6. Dissolver Offgas Hot Cell Equipment Location

The pressure relief tank will be evacuated to a specified, subatmospheric pressure prior to initiating dissolution of a target batch and selected valves (indicated as 2, 3, and 4 on Figure 6-5) closed. Valve 1 will be open during normal dissolver operation. An upset during the dissolver operation (e.g., loss of vacuum pump operation) will result in closing Valve 1 and opening Valve 2 to contain dissolver offgas within the dissolver and offgas vessels. Due to the short duration of dissolver operation, dissolution is assumed to go to completion independent of an offgas system upset. The pressure relief tank will contain the offgas as dissolution is completed.

Valves 3, 4, and 5 are provided for upset recovery. After correction of the upset cause, gases collected in the pressure relief tank will be routed to the downstream treatment unit operations via Valve 3 or returned to a caustic scrubber via Valve 4. Liquid condensed in the pressure relief tank as a result of activation will be routed to the dissolver offgas liquid waste collection tank via Valve 5 for disposal.

Accident Mitigated

- Irradiated target dissolver offgas system malfunctions, including loss of power during target dissolution operations

System Components

- Pressure relief valves
- Pressure relief tank (DS-TK-500)

Functional Requirements

- As an AEC, use relief device to relieve 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 dissolver
- Prevent a failure of the primary confinement system by capturing gaseous effluents in a vacuum receiver tank

Design Basis

The following information was developed for the Construction Permit Application describing the pressure relief tank.

- Pressure-relief tank sizing is based on a maximum dissolver batch of [Proprietary Information] that has just started dissolution when the pressure relief event is initiated.
- The non-condensable gas volume to the pressure relief tank is equivalent to all nitrogen oxide (NO_x) generated by dissolution, plus the sweep gas flow for flammable hydrogen gas mitigation.
- Worst-case reaction stoichiometry of [Proprietary Information] dissolved is used.
- No credit is taken for reaction of NO_2 with water to produce nitric acid.
- Dissolver gas additions, other than the minimum sweep gas flow for hydrogen mitigation, are terminated by the pressure relief event.
- Gas contained by the pressure relief tank and associated dissolver offgas piping is saturated with water vapor.
- The pressure change from [Proprietary Information], absolute activates the pressure relief tank.

Additional detailed information on the pressure relief tank design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS RS-10, "Active Radiation Monitoring and Isolation of Low-Dose Waste Transfer," is identified by the accident analyses described in Chapter 13.0. As an AEC, the recirculating stream and the 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 instrument will monitor the transfer lines to provide an open permissive signal to dedicated isolation valves.

Accident Mitigated

- Transfer of high-dose process liquid solutions outside the hot cell shielding boundary

System Components

Additional detailed information of the radiation monitor and isolation of low-dose waste transfers will be developed for the Operating License Application.

Functional Requirement

- Maintain worker and public exposure rates within approved limits

Design Basis

Additional detailed information of the radiation monitor and isolation of low-dose waste transfers will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS RS-13, "Cask Local Ventilation During Closure Lid Removal and Docking Preparations," is identified by the accident analyses described in Chapter 13.0. As an AEC, a local capture ventilation system will be used over the irradiated target cask closure lid to remove any escaped gases from the worker breathing zone during removal of the closure lid, removal of the shielding block bolts, and installation of the lifting lugs.

Accident Mitigated

- Irradiated target cladding fails during transportation, releasing gaseous radionuclides within the cask containment boundary

System Components

- Use a dedicated evacuation hood over the top of the cask during containment closure lid removal
- Remove gases to the Zone I secondary confinement system for processing

Functional Requirement

- Prevent exposure to workers by evacuating any high-dose gaseous radionuclides from the worker breathing zone and preventing immersion of the worker in a high-dose environment

Design Basis

The following information was developed for the Construction Permit Application describing the cask local ventilation system:

- Use the local capture ventilation 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

Additional detailed information on the cask local ventilation system design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.4 IROFS RS-15, Cask Docking Port Enabling Sensor

IROFS RS-15, "Cask Docking Port Enabling Sensor," is identified by the accident analyses described in Chapter 13.0. 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.

Accident Mitigated

- Cask lift failure occurs after shield plug removal (but before target basket removal) with targets inside the cask

System Components

- Enabling contact signal and positive closure signal when the sensor does not sense a cask mated to the cask docking port, causing the cask docking port door to close

Functional Requirement

- Prevent the cask docking port door from being opened and allowing a streaming radiation path to areas accessible by workers

Design Basis

Detailed information on the system design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.5 IROFS FS-03, Process Vessel Emergency Purge System

IROFS FS-03, "Process Vessel Emergency Purge System," is identified by the accident analyses described in Chapter 13.0. Hydrogen gas will be evolved from process solutions through radiolytic decomposition of water in the high radiation fields. An air purge to the vapor space of selected tanks will be provided by the facility air compressors to control the hydrogen concentration from radiolysis in vessel vapor space to below the flammability limit for hydrogen. As an AEC, an emergency backup set of bottled nitrogen gas will be provided for all tanks that have the potential to evolve significant volumes of hydrogen gas through the radiolytic decomposition of water (in both a short- and long-term storage condition).

Accident Mitigated

- Hydrogen deflagration or detonation in a process vessel

System Components

Information will be provided in the Operating License Application.

Functional Requirement

- Prevent development of an explosive hydrogen-air mixture in the tank vapor spaces to prevent the deflagration or detonation hazard

Design Basis

The following information was developed for the Construction Permit Application describing the process vessel emergency purge system:

- Monitor the purge pressure going into the individual tanks and open an isolation valve on low pressure (setpoint to be determined) to restore the continuous sweep of the system using nitrogen
- Provide sweep gas sufficient for the facility to allow repair of a compressed gas system outage
- Activate by sensing low pressure on the normal sweep air system, introducing a continuous purge of nitrogen from a reliable emergency backup station of bottled nitrogen into each affected vessel near the bottom (e.g., through a liquid level detection leg) of the vessel
- Dilute hydrogen as it rises to the top of the vessel and is vented to the respective vent system

Additional detailed information on the process vessel emergency purge system design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.6 IROFS FS-04, Irradiated Target Cask Lifting Fixture

IROFS FS-04, "Irradiated Target Cask Lifting Fixture," is identified by the accident analyses described in Chapter 13.0. As a passive engineered control (PEC), the irradiated target cask lifting fixture will be designed to prevent the cask from tipping within the fixture and the fixture itself from toppling during a seismic event.

Accident Mitigated

- Dislodged irradiated target shipping cask shield plug in the presence of workers during target unloading activities

System Components

Detailed information on the system components will be developed for the Operating License Application.

Functional Requirements

Detailed information on the system functional requirements will be developed for the Operating License Application.

Design Basis

Detailed information on the system design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.7 IROFS FS-05, Exhaust Stack Height

IROFS FS-05, "Exhaust Stack Height," is identified by the accident analyses described in Chapter 13.0.

Accidents Mitigated

- Process solution spills and sprays
- Carbon bed fire

System Component

- Zone I exhaust stack

Functional Requirement

- Provide an offgas release height for ventilation gases consistent with the stack height used as input to mitigated dose consequence evaluations.

Design Basis

The Zone I exhaust stack height is 23 m (75 ft).

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.8 IROFS CS-09, Double Wall Piping

[Proprietary Information]

Figure 6-7. Proposed Location of Double-Wall Piping (Example)

IROFS CS-09, "Double Wall Piping," is identified by the accident analyses in Chapter 13.0. This IROFS has both a confinement and nuclear criticality prevention function. As a PEC, the piping system conveying fissile solution between credited confinement locations will be provided with a double-wall barrier to contain any spills that may occur from the primary confinement piping. This IROFS will be used at those locations that pass through the facility, where creating a spill containment berm under the piping is neither practical nor desirable for personnel chemical protection purposes. Figure 6-7 provides an example location where IROFS CS-09 will be applied (e.g., the transfer line between the recycle uranium decay tanks and the [Proprietary Information]).

Accident Mitigated

- Leak in piping that passes between confinement enclosures

System Components

The following double-wall piping segments are identified at this time:

- Transfer piping containing fissile solutions traversing between hot cell walls
- Transfer piping connecting the uranium product transfer send tank (UR-TK-720) and uranyl nitrate storage tank (TF-TK-200)
- Other locations to be identified in final design

Functional Requirements

- Double-wall piping prevents personnel injury from exposure to acidic or caustic licensed material solutions conveyed in the piping that runs outside a confinement enclosure
- Double-wall piping routes pipe leaks to a critically-safe leak collection tank or berm as a nuclear criticality control feature

Design Basis

The double-wall piping arrangement is designed to gravity drain to a safe-geometry set of tanks or to a safe geometry berm.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.7.9 IROFS CS-18, Backflow Prevention Devices, and IROFS CS-19, Safe-Geometry Day Tanks

IROFS CS-18, "Backflow Prevention Devices," and IROFS CS-19, "Safe-Geometry Day Tanks," are identified by the accident analyses in Chapter 13.0. As a PEC or AEC, chemical and gas addition ports to fissile process solution systems will enter a confinement enclosure through a backflow prevention device. Backflow prevention devices and safe-geometry day tanks will provide alternatives for preventing process addition backflow across confinement boundaries. The 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. Therefore, these IROFSs have both a confinement and a nuclear criticality prevention function.

Accident Mitigated

- Backflow of process material located inside a confinement boundary to vessel located outside confinement via connected piping due to process upset.

System Components

System component information will be provided in the Operating License Application.

Functional Requirements

- Prevent fissile solutions and/or high dose solutions from backflowing from the tank into systems outside the confinement boundaries that may lead to accidental nuclear criticality or high exposures to workers
- Provide each hazardous location with an engineered backflow prevention device that provides high reliability and availability for that location
- Locate the backflow prevention device features for high-dose product solutions inside the confinement boundaries
- Support the backflow prevention devices with safe-geometry day tanks located inside the confinement boundary
- Direct spills from the backflow prevention device to a safe-geometry confinement berm

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.8 Dissolver Offgas Systems

6.2.1.8.1 Dissolver Offgas Iodine Removal Unit

A significant fraction of iodine entering the RPF in targets is projected to be released to dissolver offgas during target dissolution. The dissolver offgas iodine removal units will be included in the RPF as the primary SSCs for controlling the release of iodine isotopes to the environment or facility areas occupied by workers. Components of the dissolver offgas system, beginning with the iodine removal unit, will also be used to treat vent gas from the target disassembly system. Target disassembly vent gas is treated by dissolver offgas components for the Construction Application Permit configuration as a measure to mitigate the unverified potential for a release of fission gas radionuclides during target transportation.

Figure 6-5 (Section 6.2.1.7.1) shows the iodine removal unit position in the offgas treatment equipment train. The dissolver offgas iodine removal unit location in the facility is shown in Figure 6-6 (identified as "primary fission gas treatment").

Accidents Mitigated

- Projected limiting control for operation
- Required for normal operation and not for accident mitigation

System Components

- Iodine removal unit A (DS-SB-600A)
- Iodine removal unit B (DS-SB-600B)
- Iodine removal unit C (DS-SB-600C)

Functional Requirement

- Remove iodine isotopes from the dissolver offgas during normal operations such that the dose to workers complies with 10 CFR 20.1201, "Occupational Dose Limits for Adults," and the dose to the public complies with 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

Design Basis

The following information was developed for the Construction Permit Application describing each individual iodine removal unit:

- Sorbent bed of [Proprietary Information]
- Iodine removal efficiency greater than [Proprietary Information]
- Nominal superficial gas flow velocity of [Proprietary Information]
- Nominal sorbent bed operating temperature of [Proprietary Information]
- Nominal sorbent bed depth of [Proprietary Information] providing iodine removal capacity of greater than 1 year (yr).

Additional detailed information on the iodine removal unit design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.8.2 Dissolver Offgas Primary Adsorber

Noble gases (krypton [Kr] and xenon [Xe]) entering the RPF in targets are projected to be released to dissolver offgas during target dissolution. The dissolver offgas primary adsorber units will be included in the RPF as the primary SSCs for controlling the release of noble gas isotopes to the environment or facility areas occupied by workers. Components of the dissolver offgas system will also be used to treat vent gas from the target disassembly system. Target disassembly vent gas is treated by dissolver offgas components for the Construction Application Permit configuration as a measure to mitigate the unverified potential for a release of fission gas radionuclides during target transportation.

Figure 6-5 (Section 6.2.1.7.1) shows the primary adsorber position in the offgas treatment equipment train. The dissolver offgas primary adsorber location in the facility is shown in Figure 6-6 (identified as "primary fission gas treatment").

Accidents Mitigated

- Projected limiting control for operation
- Required for normal operation and not for accident mitigation

System Components

- Primary adsorber A (DS-SB-620A)
- Primary adsorber B (DS-SB-620B)
- Primary adsorber C (DS-SB-620C)

Functional Requirement

- Delay the release of noble gas isotopes via the dissolver offgas during normal operations such that the dose to workers complies with 10 CFR 20.1201 and the dose to the public complies with 10 CFR 20.1301.

Design Basis

The following information was developed for the Construction Permit Application describing each individual primary adsorber unit:

- Sorbent bed of [Proprietary Information]
- Nominal sorbent bed operating temperature of [Proprietary Information]
- Nominal gas relative humidity less than [Proprietary Information]
- Average gas flow rate of [Proprietary Information]
- Nominal superficial gas flow velocity of [Proprietary Information]
- Delay time for release of Xe isotopes of 10 days and Kr isotopes of 8 hours (hr) (additional delay time is provided by the secondary adsorber)

Additional detailed information on the primary adsorber unit design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.8.3 Dissolver Offgas Vacuum Receiver/Vacuum Pump

The dissolver offgas vacuum pump will provide the motive force for transferring offgas, generated in the dissolvers and disassembly equipment during operation, through the dissolver offgas equipment train while maintaining dissolver vessels at a pressure less than the equipment enclosure pressure. Vacuum receiver tanks will be provided as part of the motive force system to allow the vacuum pumps to cycle on and off less frequently and accommodate the wide variations in gas flow rate associated with a target dissolution cycle.

Figure 6-5 (Section 6.2.1.7.1) shows the vacuum receiver tank and vacuum pump positions in the offgas treatment equipment train. The vacuum receiver tank and vacuum pump location in the facility is shown in Figure 6-3 in the vicinity of equipment identified for the process offgas secondary iodine removal bed.

Accidents Mitigated

- Projected limiting control for operation
- Required for normal operation and not for accident mitigation

System Components

- Vacuum receiver tank A (DS-TK-700A)
- Vacuum receiver tank B (DS-TK-700B)
- Vacuum pump A (DS-P-710A)
- Vacuum pump B (DS-P-710B)

Functional Requirements

- Maintain the dissolver vessel gas space at a pressure less than the dissolver vessel enclosure pressure throughout the target dissolution cycle
- Accommodate pressure drops associated with dissolver offgas unit operations over the range of gas flow rates generated in both dissolvers and the target disassembly equipment vent throughout a target dissolution cycle

Design Basis

The following information was developed for the Construction Permit Application describing the vacuum receiver tanks and vacuum pump:

- Minimum inlet setpoint pressure of [Proprietary Information]
- Maximum inlet setpoint pressure of [Proprietary Information]
- Outlet pressure of [Proprietary Information]
- Maximum sustained gas flow into [Proprietary Information]
- Receiver tank provides a [Proprietary Information] with the vacuum pump off and inlet at the maximum sustained gas flow

Additional detailed information on the vacuum receiver tank and vacuum pump design basis will be developed for the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.2.1.9 Exhaust System

The ventilation exhaust system is described in Chapter 9.0, Section 9.1.2. Additional detailed information will be developed for the Operating License Application, including:

- Describing changes in operating conditions in response to potential accidents and the mitigation of accident radiological consequences
- Demonstrating how dispersion or distribution of contaminated air to the environment or occupied spaces is controlled
- Identifying the design bases for location and operating characteristics of the exhaust stacks

6.2.1.10 Effluent Monitoring System

Each RPF exhaust stack will include an effluent monitoring system. The monitoring system sample lines are designed to comply with ANSI N13.1, *Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities*. Additional detailed information on the effluent monitoring systems will be developed for the Operating License Application.

6.2.1.11 Radioactive Release Monitoring

The effluent monitoring system will provide flow rate, temperature, and composition inputs for dispersion modeling of releases from the exhaust stacks. These inputs will provide the capability for calculating potential exposures as a basis for actions to ensure that the public is protected during both normal operation and accident conditions. Additional detailed information on radioactive release monitoring will be developed for the Operating License Application.

6.2.1.12 Confinement System Mitigation Effects

Detailed information describing the confinement system mitigation effects will be developed for the Operating License Application. This information will compare the radiological exposures to the facility staff and the public with and without the confinement system engineered safety feature. The comparison will be based on analyses showing airflow rates, reduction in quantities of airborne radioactive material by filter systems, system isolation, and other parameters that demonstrate the effectiveness of the system.

6.2.2 Containment

Containment for the RPF is defined based on NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors - Format and Content*, Part 1 interim staff guidance.

> *Containments are required as an engineered safety feature on the basis of the radioisotope production facility design, operating characteristics, accidents scenarios, and location. A potential scenario for such a release could be a significant loss of integrity of the radioisotope extraction system or the irradiated fuel processing system. The containment is designed to control the release to the environment of airborne radioactive material that is released in the facility even if the accident is accompanied by a pressure surge or steam release.*

The NUREG-1537 Part 1 interim staff guidance has been applied to the RPF target processing systems. The current accident analysis described in Chapter 13.0 has not identified a need for a containment system as an engineered safety feature.

6.2.3 Emergency Cooling System

An emergency cooling system for the RPF is defined by NUREG-1537 Part 1 interim staff guidance.

> *In the event of the loss of any required primary or normal cooling system, an emergency cooling system may be required to remove decay heat from the fuel to prevent the failure or degradation of the gas management system, the isotope extraction system, or the irradiated fuel processing system.*

An evaluation of RPF cooling requirements provided in Chapter 5.0, "Coolant Systems," indicates that an emergency cooling system will not be required to avoid rupture of the primary process vessels. In addition, the current accident analysis described in Chapter 13.0 has not identified a need for an emergency cooling system as an engineered safety feature.

6.3 NUCLEAR CRITICALITY SAFETY IN THE RADIOISOTOPE PRODUCTION FACILITY

The RPF design will provide adequate protection against criticality hazards related to the storage, handling, and processing of SNM outside a reactor. This is accomplished by:

- Including equipment, facilities, and procedures to protect health and minimize danger to life or property
- Ensuring that the design provides for criticality control, including adherence to the double-contingency principle
- Incorporating a criticality monitoring and alarm system into the facility design

For the Construction Permit Application, the design has assumed that a nuclear criticality accident is a high-consequence event independent of whether shielding or other isolation is available between the source of radiation and facility personnel. While not considered likely at this time, justification for considering criticality events as other than a high-consequence event will be provided in the Operating License Application, if this assumption is changed for specific locations by future design activities.

The nuclear criticality safety program defines the programmatic elements that work in concert to maintain criticality controls throughout the operating life of the RPF. The nuclear criticality safety program and facility design are developed based on the following American National Standards Institute/American Nuclear Society (ANSI/ANS) standards, with exceptions described in Regulatory Guide 3.71.

- ANSI/ANS-8.1, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*
- ANSI/ANS-8.3, *Criticality Accident Alarm System*
- ANSI/ANS-8.7, *Nuclear Criticality Safety in the Storage of Fissile Materials*
- ANSI/ANS-8.10, *Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement*
- ANSI/ANS-8.19, *Administrative Practices for Nuclear Criticality Safety*
- ANSI/ANS-8.20, *Nuclear Criticality Safety Training*
- ANSI/ANS-8.22, *Nuclear Criticality Safety Based on Limiting and Controlling Moderators*
- ANSI/ANS-8.23, *Nuclear Criticality Accident Emergency Planning and Response*
- ANSI/ANS-8.24, *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations*
- ANSI/ANS-8.26, *Criticality Safety Engineer Training and Qualification Program*

For the Construction Permit Application, no deviations from standards or requirements have been identified that would require development of equivalent requirements for the RPF.

The nuclear criticality safety program includes the following elements:

- Responsibilities
- Criticality safety evaluations
- Criticality safety control implementation
- Nuclear criticality safety training
- Criticality safety assessments

- Criticality prevention specifications
- Operating procedures and maintenance work
- Criticality safety postings
- Fissile material container labeling, storage, and transport
- Criticality safety nonconformance response
- Criticality safety configuration control
- Criticality safety guidelines for firefighting
- Emergency preparedness plan and procedures

Preliminary descriptions of the nuclear criticality safety program elements developed for the Construction Permit Application are summarized below. Modifications to the nuclear criticality safety program elements are anticipated as the design matures and will be included in the Operating License Application.

Responsibilities

This element describes the responsibilities of management and staff in implementing the nuclear criticality safety program.

- General facility management will ensure that the nuclear safety function is as independent as practical from the facility operating functions.
- A Nuclear Criticality Safety Manager will be assigned and responsible for overall coordination, maintenance, and management of the nuclear criticality safety program.
- A Criticality Safety Representative will be assigned who is qualified to interpret criticality safety requirements and serve as a liaison between custodians of fissionable material and other operations, advising operating personnel and supervisors on questions concerning conformance to criticality safety requirements.
- Qualified Criticality Safety Engineers will responsible for performing criticality analyses and evaluations of systems, maintaining current verified and validated criticality computer codes, advising staff on technical aspects of criticality controls, and supporting/participating in inspections and management assessments.
- Operations management will be responsible for establishing the responsibility for criticality safety throughout the operations organization, communicating criticality safety responsibilities for each individual involved in operations, ensuring that controls identified by CSEs are implemented, ensuring each worker has necessary training and qualifications, and ensuring that procedures that include controls significant to criticality safety are prepared before operations commence.
- Supervisors and workers will be responsible for completing training before performing fissile material operations, understanding and ensuring compliance with all applicable criticality safety controls, and reporting any proposed change in fissile material operations to the Criticality Safety Representative for evaluation and approval before the operation commences.

Criticality Safety Evaluations

This element describes the process for preparing CSEs that demonstrate fissile material operation will be subcritical under both normal and credible abnormal conditions.

- CSEs will determine, identify, and document the controlled parameters and associated limits on which criticality safety depends.
- CSEs will be required to evaluate normal operations, and contingent and upset conditions.

- Preliminary CSEs prepared for the Construction Permit Application, including verification and validation of supporting computer codes, are described in Section 6.3.1.1 and provide examples of the CSEs.
- Design changes impacting criticality will be reviewed by the Criticality Safety Representative.
- CSEs will be independently reviewed to confirm the technical adequacy of the evaluation prior to commencing new or modified fissile material operations.

Criticality Safety Control Implementation

This element describes the process for implementing criticality safety controls defined by the CSEs.

- Implementation includes confirming that:
  - All required engineered criticality safety controls are maintained by a configuration management system.
  - Equipment dimensions, volumes, or other features relied on for controls are with limits documented in the CSEs.
  - Administrative criticality safety controls from CSEs are implemented in written operating and maintenance procedures.
- Fissile material inventories will be monitored and incorporated into implementation of criticality safety controls.
- Access to fissionable material will be controlled.

Nuclear Criticality Safety Training

This element describes the training program for nuclear criticality safety based on the worker's duties and responsibilities.

- This training program is developed and implemented with input from the nuclear criticality safety staff, training staff, and management, with a focus on:
  - Knowledge of the physics associated with nuclear criticality safety
  - Analysis of jobs and tasks to determine the knowledge a worker must have to perform tasks efficiently
  - Design and development of learning objectives based on the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker
  - Implementation of revised or temporary operating procedures
  - Testing methods to demonstrate competence in training materials dependent on an individual's responsibility
  - Training records maintenance
- General training on criticality hazards and alarm responses will be provided to all RPF personnel and visitors.
- Operators responsible for some aspect of nuclear criticality safety will:
  - Satisfy defined minimum initial qualifications
  - Complete an initial criticality safety training course designed for operators
  - Perform periodic requalification training
- Management, operations supervisor, and technical staff responsible for some aspect of nuclear criticality safety will:

  - Satisfy defined minimum initial qualifications
  - Complete an initial criticality safety training course designed for managers and engineers
  - Perform periodic requalification training
- The Criticality Safety Representative will:
  - Satisfy defined minimum initial qualifications
  - Complete an initial criticality safety program designed for the Criticality Safety Representative
  - Demonstrate competence in understanding facility nuclear criticality controls and procedures
  - Perform periodic requalification training
- Criticality Safety Engineers will be trained and qualified in accordance with ANSI/ANS-8.26.

Criticality Safety Assessments

This element describes the periodic criticality safety inspections and assessments conducted to ensure that the criticality safety program is maintained at an adequate level for the RPF.

- Annual criticality safety inspections will be conducted to satisfy the requirement of ANSI/ANS-8.1 and 8.19 for operational reviews to be conducted at least annually.
- Procedures will be developed for performing periodic criticality safety inspections. The facility Criticality Safety Representative and inspection team will comprise individuals (typically from Engineering) who are knowledgeable of criticality safety, and who, to the extent practicable, are not immediately responsible for the operation being inspected.
- Facility inspections are conducted to verify that the facility configuration and activities comply with the nuclear criticality safety program. Facility inspections generally consist of observation of task preparation and verification of field procedures and training.
- Management assessments will be conducted of the nuclear criticality safety program. These assessments will be led by the Nuclear Criticality Safety Manager, with assistance from other members of the criticality safety staff. The criticality safety staff is independent of the operating organization and not directly responsible for the operations.
- Records generated during performance of criticality safety inspections and assessments will be included in a criticality safety inspection report or specialty assessment report.

Criticality Prevention Specifications

This element describes the requirements for the criticality prevention specifications used to implement limits and controls established in the CSEs for safe handling of fissionable material and implements the ANSI/ANS-8 series requirement for clear communication of criticality safety limits and controls.

- Each criticality prevention specification will:
  - Be based on an approved CSE and refer to the CSE used as a specification source
  - Be prepared by either the Criticality Safety Representative of a qualified Criticality Safety Engineer
  - Emphasize limits controllable by the operator
  - Have clear and unambiguous meaning and be written, to the extent practical, using operations terminology with common units of measure

Operating Procedures and Maintenance Work

This element describes the requirements for implementing nuclear criticality controls in written procedures for operations and maintenance work.

- Procedures will meet the intent of ANSI/ANS-8.19.
- Procedures for operations and maintenance work will be prepared according to approved procedure control programs, developed and maintained to reflect changes in operations, and written so that no single inadvertent failure to follow a procedure can cause a criticality accident.
- Operating procedures will include:
  - Controls and limits significant to nuclear criticality safety of the operation
  - Periodic revisions, as necessary
  - Periodic review of active procedures by supervisors
- Operating procedures will be supplemented by criticality safety postings on equipment or incorporated in operating checklists.
- Maintenance work procedures associated with SSCs affecting nuclear criticality safety will be reviewed by the Criticality Safety Representative or a Criticality Safety Engineer for compliance with nuclear criticality safety limits based on current RPF conditions present prior to initiating each maintenance evolution.

Criticality Safety Postings

- Criticality safety postings will be developed for the Operating Permit Application.

Fissile Material Container Labeling, Storage, and Transport

- Fissile material container labeling, storage, and transport will be developed for the Operating Permit Application.

Criticality Safety Nonconformance Response

This element describes the response to deviations from defined nuclear criticality safety controls.

- Deviations from procedures and unforeseen alterations in process conditions that affect criticality safety will be immediately reported to management and the Criticality Safety Representative or a Criticality Safety Engineer.
- NWMI management will provide the required notifications of the deviation to the U.S. Nuclear Regulatory Commission Operations Center.
- The Criticality Safety Representative or a Criticality Safety Engineer will support an investigative team comprising, at a minimum, the Operations Manager and operations personnel familiar with the operation in question during the development of a recovery plan for safely returning to compliance with the procedures.
- The deviation will be corrected per the recovery plan and the incident documented.
- Action is to be taken to ensure that a similar situation does not exist in another part of the facility and to prevent recurrence of the nonconformance.

Criticality Safety Configuration Control

This element describes the criticality safety configuration controls.

- The primary criticality safety control, performed at the start of a proposed activity or equipment change, is for the Criticality Safety Representative to confirm if an existing active CSE is applicable.
- All dimensions, nuclear properties, and other features on which reliance is placed will be documented and verified prior to beginning operations, and control will be exercised to maintain them.
- The nuclear criticality safety staff will provide technical guidance for the design of equipment and processes and for the development of operating procedures.
- All proposed criticality safety-related changes to design or process configuration will be reviewed by a Criticality Safety Representative or Criticality Safety Engineer to ensure that the change can be performed under an approved CSE.
- All operational changes that impact criticality safety will be documented and include proper approval designation.
- The project manager will request a CSE applicability review at the earliest practical stage of a project to determine if there could be criticality safety impacts. If the potential exists for the physical configuration or operating parameters for new or revised equipment to affect criticality safety, the drawings and process control plans will be reviewed and approved by a Criticality Safety Representative or Criticality Safety Engineer, in compliance with standard engineering practices and procedures.
- Facility and process change control will include the following.
  - The change management process will be in accordance with ANSI/ANS-8.19.
  - All dimensions, nuclear properties, and other features on which reliance is placed will be documented and verified prior to beginning operations, and control will be exercised to maintain them.
  - Changes that involve or could affect nuclear criticality controls will be evaluated under 10 CFR 50.59, "Changes, Tests, and Experiments."
  - Changes include new designs, operation, or modification to existing SSCs, computer programs, processes, operating procedures, or management measures.
  - Changes that involve or could affect nuclear criticality controls will be reviewed and approved by the Criticality Safety Representative.
  - Prior to implementing the change, the process will be determined to be subcritical (with an approved margin for safety) under both normal and credible accident scenarios.

Testing and Calibration of Active Engineered Controls

- Testing and calibration of AECs will be developed for the Operating Permit Application.

Criticality Safety Guidelines for Firefighting

- Criticality safety guidelines for firefighting will be developed for the Operating Permit Application.

Emergency Preparedness Plan and Procedures

This element describes the response to criticality accidents.

- The criticality accident alarm system (CAAS) will be used as described in Section 6.3.1.1 and provides for detection and annunciation of criticality accidents.
- Emergency procedures will be prepared and approved by management.
- Facility and off-site organizations expected to respond to emergencies will be informed of conditions that might be encountered.
- Procedures will:
  - Designate evacuation routes that are clearly identified and follow the quickest, most direct routes practical
  - Include assessment of exposure to individuals
  - Designate personnel assembly stations outside the areas to be evacuated.
- A method to account for personnel will be established and arrangements made in advance for the care and treatment of injured and exposed personnel.
- The possibility of personnel contamination by radioactive material will be considered.
- Personnel will be trained in evaluation methods, informed of routes and assembly stations, and drills performed at least annually.
- Instrumentation and procedures will be provided for determining radiation in an evacuated area following a criticality accident and information collected in a central location.
- Emergency procedures will be maintained for each area in which special nuclear material is handled, used, or stored to ensure that all personnel withdraw to an area of safety on sounding the alarm.
- Emergency procedures will include conducting drills to familiarize personnel with the evacuation plan, designation of responsible individuals to determine the cause of the alarm, and placement of radiation survey instruments in accessible locations for use in such an emergency.
- The current emergency procedures for each area will be retained as a record for as long as licensed special nuclear material is handled, used, or stored in the area.
- Superseded sections of emergency procedures will be retained for three years after the section is superseded.
- Fixed and personnel accident dosimeters will be provided in areas that require a CAAS.
- Dosimeters will be readily available to personnel responding to an emergency and a method provided for prompt on-site dosimeter readouts.

6.3.1 Criticality Safety Controls

The following sections describe criticality safety controls based on information developed for the Construction Permit Application. Section 6.3.1.1 summarizes the results of preliminary CSEs that define PECs and AECs credited to satisfy the double-contingency control principle. Section 6.3.1.2 summarizes IROFS related to preventing a nuclear criticality identified by the accident analyses described in Chapter 13.0.

6.3.1.1 Preliminary Criticality Safety Evaluations

A series of calculations were performed to support the Construction Permit Application investigating parameters associated with prevention of nuclear criticality in the current equipment configuration of major process systems. The calculations are described in the following documents:

- NWMI-2015-CRITCALC-001, *Single Parameter Subcritical Limits for 20 wt% ^{235}U – Uranium Metal, Uranium Oxide, and Homogenous Water Mixtures*
- NWMI-2015-CRITCALC-002, *Irradiated Target Low-Enriched Uranium Material Dissolution*
- NWMI-2015-CRITCALC-003, *55-Gallon Drum Arrays*
- NWMI-2015-CRITCALC-005, *Target Fabrication Tanks, Wet Processes, and Storage*
- NWMI-2015-CRITCALC-006, *Tank Hot Cell*

Calculations were performed using the MCNP 6.1 code (LA-CP-13-00634, *MCNP6 User Manual*). Validation of the MCNP 6.1 code used in the calculations is described in [Proprietary Information]. The validation report documents the methodology and results for the bias and bias uncertainty values calculated for homogeneous and heterogeneous uranium systems for the MCNP 6.1 code system. The bias is expressed as upper subcritical limits (USL) calculated using a facility-specific [Proprietary Information]. The primary focus of the validation was to determine the bias and bias uncertainty for intermediate-enriched uranium (IEU) systems. However, sufficient experiments for low-enriched uranium (LEU) and high-enriched uranium were included to demonstrate that there is no variation in the USL with varying enrichment. Similarly, the primary focus of the validation was on thermal neutron energy systems. However, sufficient experiments for intermediate and fast energy experiments were included to demonstrate that there is no variation in the USL with increasing neutron energy.

The purpose of the computer code validation is to determine values of k_{eff} that are demonstrated to be subcritical (at or below the USL) for areas of applicability similar to systems or operations being analyzed. The USL is defined by Equation 6-1.

$$USL = 1.0 - Bias - Bias\ Uncertainty - Margin\ of\ Subcriticality \qquad \text{Equation 6-1}$$

[Proprietary Information] rearranges Equation 6-1 to produce a criterion for model cases that are considered acceptable as subcritical, as shown by Equation 6-2, and incorporates the margin of subcriticality in the USL as required by ANSI/ANS-8.1.

$$k_{eff} + (2 \times \sigma_{calc}) \leq USL \qquad \text{Equation 6-2}$$

where k_{eff} is the MCNP calculated k-effective and σ_{calc} is the MCNP calculation uncertainty.

[Proprietary Information]

[Proprietary Information] indicates the validation is appropriate for homogeneous and heterogeneous IEU systems. A summary of the area of applicability is provided in Table 6-4. For systems outside the validation area of applicability, an increased margin of subcriticality value may be warranted, depending on the specific problem being analyzed. The analyst must document any extrapolation beyond the validation area of applicability, and justification must be documented for no adjustments to the margin of subcriticality when extrapolating.

Table 6-4. Area of Applicability Summary

Parameter	Area of Applicability
Fissile material	[Proprietary Information]
Fissile material form	[Proprietary Information]
$H/^{235}U$ ratio	[Proprietary Information]
Average neutron energy causing fission	[Proprietary Information]
Enrichment	[Proprietary Information]
Moderating materials	[Proprietary Information]
Reflecting materials	[Proprietary Information]
Absorber materials	[Proprietary Information]
Geometry	[Proprietary Information]

[a] Source: [Proprietary Information]

ANECF = average neutron energy causing fission.

The RPF was divided into 13 activity groups for development of preliminary CSEs of the activities and associated equipment. Controlled nuclear criticality safety parameters vary with the activity group and are summarized in Table 6-5. A minimum of two nuclear criticality safety parameters are controlled to satisfy the double-contingency principle.

Table 6-5. Controlled Nuclear Criticality Safety Parameters

Nuclear parameter	NWMI criticality safety evaluation (NWMI-2015-CSE[a])												
	01	02	03	04	05	06	07	08	09	10	11	12	13
Mass	Y	Y	Y	Y	Y	Y	Y	N	Y	Y	Y[b]	Y	Y
Geometry	Y	Y	Y	Y	Y	Y[c]	Y[c]	Y	N	Y	Y	Y	Y
Moderation	Y	N	N	N	N	N	N	N	N	N	N	N	N
Interaction	Y	Y	Y	Y	Y	Y	Y	Y	N	Y	Y	Y	Y
Volume	Y	Y	Y	Y	Y	Y	Y	N	N	N	Y	N	Y
Concentration/ density	N	Y[d]	Y[d]	Y[d]	Y[d]	N	N	N	Y[e]	Y[e]	Y[e]	N	N
Reflection	N	N	N	N	N	N	N	N	N	N	N	N	N
Absorbers	N	N	N	N	N	N	N	N	N	N	N	N	N
Enrichment[f]	N	N	N	N	N	N	N	N	N	N	N	N	N

[a] Derived from the indicated CSE reference document.
[b] Limited by nature of process in the air filtration.
[c] Limited by target design.
[d] Controlled through input fissile mass.
[e] Limited by total uranium mass allowed in the system.
[f] Facility license limited to ≤20 wt% ^{235}U.

^{235}U = uranium-235.
CSE = criticality safety evaluation.
N = no.
NWMI = Northwest Medical Isotopes, LLC.
Y = yes.

The preliminary CSEs define a series of PECs, AECs, and administrative controls that are credited to satisfy the double-contingency control principle for prevention of nuclear criticality events such that at least two changes in process conditions must occur before criticality is possible. PECs, AECs, and administrative controls are described for the 13 activity groups in the following referenced tables:

- NWMI-2015-CSE-01, Irradiated Target Handling and Disassembly (Table 6-6)
- NWMI-2015-CSE-02, Irradiated Low-Enriched Uranium Target Material Dissolution (Table 6-7)
- NWMI-2015-CSE-03, Molybdenum-99 Recovery (Table 6-8)
- NWMI-2015-CSE-04, Low-Enriched Uranium Target Material Production (Table 6-9)
- NWMI-2015-CSE-05, Target Fabrication Uranium Solution Processes (Table 6-9)
- NWMI-2015-CSE-06, Target Finishing (Table 6-9)
- NWMI-2015-CSE-07, Target and Can Storage and Carts (Table 6-9)
- NWMI-2015-CSE-08, Hot Cell Uranium Purification (Table 6-10)
- NWMI-2015-CSE-09, Waste Liquid Processing (Table 6-11)
- NWMI-2015-CSE-10, Solid Waste Collection, Encapsulation, and Staging (Table 6-11)
- NWMI-2015-CSE-11, Offgas and Ventilation (Table 6-12)
- NWMI-2015-CSE-12, Target Transport Cask or Drum Handling – The shipping package dictate design features that must be properly implemented for legal over the road transport. This CSE does not impose or credit additional passive controls other than those already incorporated in the respective shipping packages.
- NWMI-2015-CSE-13, Analytical Laboratory

Criticality controls are selected based on the following order of preference:

- Passive engineered controls
- Active engineered controls
- Enhanced administrative controls
- Administrative controls

Note that a number of features listed in the preliminary CSEs are duplicated in multiple activity groups (e.g., the floor of cells is verified to be flat, with no collection points deeper than 3.5 centimeters [cm]). Duplications are included in the current listings to clearly identify minor dimension variations that may exist in the defined features for different activity groups.

Table 6-6. [Proprietary Information] Double-Contingency Controls

Identifier[a]	Feature description and basis
CSE-01-PDF1	[Proprietary Information]
CSE-01-PDF2	[Proprietary Information]
CSE-01-PDF3	[Proprietary Information]
CSE-01-AC1	[Proprietary Information]
CSE-01-AC2	[Proprietary Information]
CSE-01-AC3	[Proprietary Information]
CSE-01-AC4	[Proprietary Information]

[a] [Proprietary Information]

HEPA = high-efficiency particulate air.
SPL = single parameter limit.

Table 6-7. [Proprietary Information] Double-Contingency Controls (2 pages)

Identifier[a]	Feature description and basis
CSE-02-PDF1	[Proprietary Information]
CSE-02-PDF2	[Proprietary Information]
CSE-02-PDF3	[Proprietary Information]
CSE-02-PDF4	[Proprietary Information]
CSE-02-PDF5	[Proprietary Information]
CSE-02-PDF6	[Proprietary Information]
CSE-02-PDF7	[Proprietary Information]
CSE-02-PDF8	[Proprietary Information]
CSE-02-AEF1	[Proprietary Information]
CSE-02-AC1	[Proprietary Information]
CSE-02-AC2	[Proprietary Information]

[a] [Proprietary Information]

[Proprietary Information$_2$ =[Proprietary Information].

Table 6-8. [Proprietary Information] Double-Contingency Controls (2 pages)

Identifier[a]	Feature description and basis
CSE-03-PDF1	[Proprietary Information]
CSE-03-PDF2	[Proprietary Information]
CSE-03-PDF3	[Proprietary Information]
CSE-03-PDF4	[Proprietary Information]
CSE-03-PDF5	[Proprietary Information]
CSE-03-PDF6	[Proprietary Information]
CSE-03-PDF7	[Proprietary Information]
CSE-03-PDF8	[Proprietary Information]
CSE-03-PDF9	[Proprietary Information]
CSE-03-PDF10	[Proprietary Information]
CSE-03-PDF11	[Proprietary Information]
CSE-03-PDF12	[Proprietary Information]
CSE-03-AEF1	[Proprietary Information]
CSE-03-AC1	[Proprietary Information]

[a] [Proprietary Information]

IX = ion exchange.
Mo = molybdenum.
UO_2 = uranium dioxide.

[Proprietary Information]

Table 6-9. [Proprietary Information] Double-Contingency Controls (8 pages)

Identifier	Feature description and basis
CSE-04-PDF1[a]	[Proprietary Information]
CSE-04-PDF2[a]	[Proprietary Information]
CSE-04-PDF3[a]	[Proprietary Information]
CSE-04-PDF4[a]	[Proprietary Information]
CSE-04-PDF5[a]	[Proprietary Information]
CSE-04-PDF6[a]	[Proprietary Information]
CSE-04-PDF7[a]	[Proprietary Information]
CSE-04-PDF8[a]	[Proprietary Information]
CSE-04-PDF9[a]	[Proprietary Information]
CSE-04-PDF10[a]	[Proprietary Information]
CSE-04-PDF11[a]	[Proprietary Information]
CSE-04-PDF12[a]	[Proprietary Information]
CSE-04-PDF13[a]	[Proprietary Information]
CSE-04-PDF14[a]	[Proprietary Information]
CSE-04-PDF15[a]	[Proprietary Information]
CSE-04-PDF16[a]	[Proprietary Information]
CSE-04-AEF1[a]	[Proprietary Information]
CSE-04-AC1[a]	[Proprietary Information]
CSE-04-AC2[a]	[Proprietary Information]
CSE-04-AC3[a]	[Proprietary Information]
CSE-04-AC4[a]	[Proprietary Information]
CSE-04-AC5[a]	[Proprietary Information]
CSE-04-AC6[a]	[Proprietary Information]
CSE-04-AC7[a]	[Proprietary Information]
CSE-05-PDF1[b]	[Proprietary Information]
CSE-05-PDF2[b]	[Proprietary Information]
CSE-05-PDF3[b]	[Proprietary Information]
CSE-05-PDF4[b]	[Proprietary Information]
CSE-05-PDF5[b]	[Proprietary Information]
CSE-05-PDF6[b]	[Proprietary Information]
CSE-05-PDF7[b]	[Proprietary Information]

Table 6-9. [Proprietary Information] Double-Contingency Controls (8 pages)

Identifier	Feature description and basis
CSE-05-PDF8[b]	[Proprietary Information]
CSE-05-AEF1[b]	[Proprietary Information]
CSE-05-AEF2[b]	[Proprietary Information]
CSE-05-AEF3[b]	[Proprietary Information]
CSE-05-AC1[b]	[Proprietary Information]
CSE-05-AC2[b]	[Proprietary Information]
CSE-05-AC3[b]	[Proprietary Information]
CSE-06-PDF1[c]	[Proprietary Information]
CSE-06-PDF2[c]	[Proprietary Information]
CSE-06-AC1[c]	[Proprietary Information]
CSE-06-AC2[c]	[Proprietary Information]
CSE-06-AC3[c]	[Proprietary Information]
CSE-06-AC4[c]	[Proprietary Information]
CSE-06-AC5[c]	[Proprietary Information]
CSE-06-AC6[c]	[Proprietary Information]
CSE-07-PDF1[d]	[Proprietary Information]
CSE-07-PDF2[d]	[Proprietary Information]
CSE-07-PDF3[d]	[Proprietary Information]
CSE-07-PDF4[d]	[Proprietary Information]
CSE-07-AC1[d]	[Proprietary Information]
CSE-07-AC2[d]	[Proprietary Information]
CSE-07-AC3[d]	[Proprietary Information]
CSE-07-AC4[d]	[Proprietary Information]
CSE-07-AC5[d]	[Proprietary Information]
CSE-07-AC6[d]	[Proprietary Information]
CSE-07-AC7[d]	[Proprietary Information]

[a] [Proprietary Information]
[b] [Proprietary Information]
[c] [Proprietary Information]
[d] [Proprietary Information]

ADUN = acid-deficient uranium nitrate.
DBE = design basis earthquake.
U = uranium.
UN = uranium nitride.
[Proprietary Information] =[Proprietary Information].

[Proprietary Information]

[Proprietary Information]

[Proprietary Information]

[Proprietary Information]

[Proprietary Information]

[Proprietary Information]

Table 6-10. [Proprietary Information] Double-Contingency Controls (2 pages)

Identifier[a]	Feature description and basis
CSE-08-PDF1	[Proprietary Information]
CSE-08-PDF2	[Proprietary Information]
CSE-08-PDF3	[Proprietary Information]
CSE-08-PDF4	[Proprietary Information]
CSE-08-PDF5	[Proprietary Information]
CSE-08-PDF6	[Proprietary Information]
CSE-08-PDF7	[Proprietary Information]
CSE-08-PDF8	[Proprietary Information]
CSE-08-PDF9	[Proprietary Information]
CSE-08-PDF10	[Proprietary Information]
CSE-08-PDF11	[Proprietary Information]
CSE-08-PDF12	[Proprietary Information]
CSE-08-AEF1	[Proprietary Information]
CSE-08-AC1	[Proprietary Information]
CSE-08-AC2	[Proprietary Information]

[a] [Proprietary Information]

DBE = design basis earthquake.
IX = ion exchange.

[Proprietary Information]

Table 6-11. [Proprietary Information] Double-Contingency Controls (3 pages)

Identifier	Feature description and basis
CSE-09-AEF1[a]	[Proprietary Information]
CSE-09-AC1[a]	[Proprietary Information]
CSE-09-AC2[a]	[Proprietary Information]
CSE-09-AC3[a]	[Proprietary Information]
CSE-10-PDF1[b]	[Proprietary Information]
CSE-10-AEF1[b]	[Proprietary Information]
CSE-10-AC1[b]	[Proprietary Information]
CSE-10-AC2[b]	[Proprietary Information]
CSE-10-AC3[b]	[Proprietary Information]
CSE-10-AC4[b]	[Proprietary Information]
CSE-10-AC5[b]	[Proprietary Information]
CSE-10-AC6[b]	[Proprietary Information]
CSE-10-AC7[b]	[Proprietary Information]
CSE-10-AC8[b]	[Proprietary Information]
CSE-10-AC9[b]	[Proprietary Information]

[a] [Proprietary Information]
[b] [Proprietary Information]

^{235}U = uranium-235.
HIC = high-integrity container.
RPF = Radioisotope Production Facility.
SPL = single parameter limit.
U = uranium.

[Proprietary Information]

[Proprietary Information]

Table 6-12. [Proprietary Information] Double-Contingency Controls (2 pages)

Identifier[a]	Feature description and basis
CSE-11-PDF1	[Proprietary Information]
CSE-11-PDF2	[Proprietary Information]
CSE-11-PDF3	[Proprietary Information]
CSE-11-PDF4	[Proprietary Information]
CSE-11-PDF5	[Proprietary Information]
CSE-11-PDF6	[Proprietary Information]
CSE-11-PDF7	[Proprietary Information]
CSE-11-PDF8	[Proprietary Information]
CSE-11-AEF1	[Proprietary Information]
CSE-11-AC1	[Proprietary Information]

[a] [Proprietary Information]

DBE = design basis earthquake.
HEPA = high-efficiency particulate air.
Mo = molybdenum.
NO_x = nitrogen oxide.

Table 6-13. [Proprietary Information] Double-Contingency Controls (2 pages)

Identifier[a]	Feature description and basis
CSE-13-PDF1	[Proprietary Information]
CSE-13-PDF2	[Proprietary Information]
CSE-13-PDF3	[Proprietary Information]
CSE-13-AC1	[Proprietary Information]
CSE-13-AC2	[Proprietary Information]
CSE-13-AC3	[Proprietary Information]
CSE-13-AC4	[Proprietary Information]
CSE-13-AC5	[Proprietary Information]
CSE-13-AC6	[Proprietary Information]

[a] [Proprietary Information]

R&D = research and development.
RPF = Radioisotope Production Facility.
SPL = single parameter limit.
U = uranium.

[Proprietary Information]

Each of the preliminary CSEs indicates that the process areas evaluated will be within the detector and alarm coverage of the CAAS. Evaluation of the CAAS coverage will be performed after final design is complete and prior to facility startup. The CAAS will be designed to meet the following.

- The facility CAAS:
  - Will be capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 radiation dose absorbed (rad) of combined neutron and gamma radiation at an unshielded distance of 2 m from the reacting material within 1 minute; two detectors will cover each area needing CAAS coverage
  - Will use gamma and neutron sensitive radiation detectors that energize clearly audible alarm signals if an accidental criticality occurs
  - Will comply with ANSI/ANS-8.3, as modified by Regulatory Guide 3.71
  - Will be appropriate for the type of radiation detected, the intervening shielding, and the magnitude of the minimum accident of concern
  - Will be designed to remain operational during design basis accidents
  - Will be clearly audible in areas that must be evacuated or there will be alternative notification methods that are documented to be effective in notifying personnel that evaluation is necessary
- Operations will be rendered safe, by shutdown and quarantine, if necessary, in any area where CAAS coverage has been lost and not restored within a specified number of hours. The number of hours will be determined on a process-by-process basis, because shutting down certain processes, even to make them safe, may carry a larger risk than being without a CAAS for a short time. Compensatory measures (e.g., limiting access, halting SNM movement, or restoring CAAS coverage with an alternate instrument) when the CAAS is not functional will be determined for inclusion in the Operating Permit Application.
- Emergency power will be provided to the CAAS by the uninterruptable power supply system.

6.3.1.2 Derived Nuclear Criticality Safety Items Relied on for Safety

The following subsections describe engineered safety features that are derived from the accident scenarios that could result in a nuclear criticality, as described in Chapter 13.0.

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

IROFS CS-04, "Interaction Control Spacing Provided by Passively Designed Fixtures and Workstation Placement," is identified by the accident analyses in Chapter 13.0. During handling of uranium solids and solutions outside of processing systems under normal conditions, the material will be handled in safe masses controlled by either physical measurement or batch limits on well characterized devices. Solid uranium will be handled outside of processing systems during:

- Receipt and processing of fresh uranium (and presumably shipment of spent uranium back to the supplier)
- [Proprietary Information]
- Fabrication of targets using [Proprietary Information] LEU target material (including movement of LEU target material to and from the fabrication workstation and handling of the completed targets)

- Disassembly of targets following irradiation
- Laboratory sampling and analysis activities (albeit in smaller quantities).

Each activity is assigned a mass or batch limit for safe handling.

Accident Mitigated

The accident occurs when a safe mass or batch limit is exceeded beyond some bounding extent based on the management measures on the control. Note that this accident involves normal condition criticality controlled limits for safe handling, and the upset represents failure of an associated administrative control. The most limiting activity would involve processing the LEU target material from [Proprietary Information]. If the IROFS fails, accidental nuclear criticality is possible without additional control.

System Components

As a PEC, fixed interaction control fixtures or workstations will be provided for holding or processing approved containers containing approved quantities of uranium metal, [Proprietary Information], batches of targets, and batches of samples.

Functional Requirements

The fixtures are designed to hold only the approved container or batch and are fixed with 2-ft edge-to-edge spacing from all other fissile material containers, workstations, or fissile solution tanks, vessels, and ion exchange (IX) columns. Where LEU target material is handled in open containers, the design will prevent spills from readily spreading to an adjacent workstation or storage location.

Design Basis

Final workstation and fixture spacing will be determined in final design when all process upsets are evaluated. Workstations with interaction controls include the following (not an all-inclusive listing):

- [Proprietary Information]
- [Proprietary Information]
- Target basket fixture that provides safe spacing of a batch of targets from one another in the target receipt cell

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.2 IROFS CS-06, Pencil Tank, Vessel, or Piping Safe Geometry Confinement Using the Diameter of Tanks, Vessels, or Piping

IROFS CS-06, "Pencil Tank, Vessel, or Piping Safe Geometry Confinement using the Diameter of Tanks, Vessels, or Piping," is identified by the accident analyses in Chapter 13.0. The PHA in Chapter 13.0 identified a number of individual potential initiating events that could lead to a spill of fissile solution from the geometrically safe confinement tanks, vessels, or piping that provide the primary safety functions of the processes. Four processing systems will handle fissile solutions:

- Target fabrication (from the [Proprietary Information])

- Target dissolution system
- First stage of molybdenum recovery and purification
- Entire uranium recovery and recycle system

Three of these systems will be at least partially located within the hot cell wall boundary due to the high-dose of the fission products. Initiating events include the general categories of tank, vessel, or piping failure due to operator error (valves out of position), valves leaking, equipment leaking (pumps, piping, vessels, etc.), high pressure events from various causes including high temperature solutions (locked in boundary valves), hydrogen detonation, and exothermic reactions with the wrong resins or reagents used in the respective systems. Some of the initiators result in small leaks that are identified and mitigated (e.g., pump seal and small valve leaks). Over the life of the facility, these types of leaks are to be expected, but do not challenge the overall safety of RPF operations.

Accident Mitigated

The accident of concern involves fissile process solution in quantities necessary to sustain accidental nuclear criticality. Larger catastrophic leaks or ruptures of equipment must occur for enough material to be released. Such leaks would represent a failure of the safe-geometry confinement IROFS for the respective equipment. Thus, scenarios leading to this accident sequence involve the failure of these IROFS. Due to the nature of the process, the worst-case accident involves the tanks with the largest capacity and the highest normal case concentrations.

System Components

As a PEC, pencil tanks and other standalone vessels are designed and will be fabricated with a safe-geometry diameter for safe storage and processing of fissile solutions. The safe diameters of various tanks, vessels, or components will be provided in the Operating License Application.

Functional Requirements

The safety function of safe diameter vessels is also one of confinement of the contained solution. The safe-geometry confinement of fissile solutions will prevent accidental nuclear criticality, a high consequence event. The safe-geometry confinement diameter will conservatively include the outside diameter of the tank wall or out to the outside diameter of any heating or cooling jackets (or any other void spaces that may inadvertently capture fissile solution) on the vessels. Where insulation is used on the outside wall of a vessel, 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.

Design Basis

The safe-geometry diameter of tanks, vessels, and piping will be determined in final design after finalizing the reference CSEs. Note that preliminary vessel sizes for activity groups are listed in the double-contingency parameters described in Section 6.3.1.1.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.3 IROFS CS-07, Pencil Tank Geometry Control on Fixed Interaction Spacing of Individual Tanks

IROFS CS-07, "Pencil Tank Geometry Control on Fixed Interaction Spacing of Individual Tanks," is identified by the accident analyses in Chapter 13.0 (see description in Section 6.3.1.2.2).

Accident Mitigated

See description in Section 6.3.1.2.2.

System Components

As a PEC, pencil tanks and other standalone vessels (controlled with safe geometry or volume constraints) are designed and will be fabricated with a fixed interaction spacing for safe storage and processing of the fissile solutions. Tanks, vessels, and components requiring fixed interaction control spacing of the barrels within each set of pencil tanks and between various tanks, vessels, or components will be provided in the Operating License Application.

Functional Requirements

The safety function of fixed interaction spacing of individual tanks 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 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 spacing will be measured from the outside of the respective tanks, vessels, or component or from the outside of any heating or cooling jackets (or any other void spaces that may inadvertently capture fissile solution) on the vessels or component. The fixed interaction control distance from the safe slab depth spill containment berm will also be specified where applicable.

Design Basis

Actual interaction control parameters will be defined during final design. In addition, the following generic interaction control parameters apply during design.

- Connecting piping between fissile material components will not exceed a cross-sectional density to be determined during final evaluation of systems.
- Edge-to-edge spacing between fissile material-bearing vessels and components and the concrete reflector presented by the hot cell shielding walls will be fixed at a distance to be determined during final evaluation of all components.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.4 IROFS CS-08, Floor and Sump Geometry Control on Slab Depth, Sump Diameter or Depth for Floor Dikes

IROFS CS-08, "Floor and Sump Geometry Control on Slab Depth, Sump Diameter or Depth for Floor Dikes," is identified by the accident analyses described in Chapter 13.0 (see description in Section 6.3.1.2.2).

Accident Mitigated

See description in Section 6.3.1.2.2.

System Components

As a PEC, the floor under designated tanks, vessels, and workstations will be constructed with a spill containment berm using a safe-geometry slab depth, and one or more collection sumps with diameters or depths, to be determined in final design.

Functional Requirements

The safety function of a spill containment berm is to 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 overhead systems. 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 is operable if it contains reserve volume for the largest single credible spill. Spill containment berm sizes and locations will be determined during final design.

Design Basis

The safe-geometry slab depth under designated tanks, vessels, and workstations will be determined during final design after finalizing the reference CSEs. Note that the preliminary slab depth for the activity groups are listed in the double-contingency parameters described in Section 6.3.1.1.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.5 IROFS CS-09, Double-Wall Piping

IROFS CS-09, "Double Wall Piping," is identified by the accident analyses described in Chapter 13.0. As a PEC, a piping system for conveying fissile solution between confinement structures will be provided with a double-wall barrier to contain any spills that may occur from the primary piping.

Accident Mitigated

- Leak in piping that passes between confinement enclosures

System Components

IROFS CS-09 is used at the locations listed below that pass through the facility where creating a spill containment berm under the piping is neither practical nor desirable for personnel chemical protection purposes. The following double-wall piping segments are identified for criticality safety:

- Transfer piping containing fissile solutions traversing between hot cell walls
- Transfer piping connecting the uranium product transfer send tank (UR-TK-720) and the uranyl nitrate storage tank (TF-TK-200)
- Any other locations in final design where fissile solution piping exits a safe-slab spill containment berm and enters another

Functional Requirements

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 double-wall piping will also function as a barrier to prevent fissile solution from soaking into the concrete from lines passing through concrete walls where required by the criticality safety analysis (e.g., see PDF2 of Table 6-9). The secondary safety function of double-wall piping is to prevent personnel injury from exposure to acidic or caustic licensed material solutions that are conveyed in the piping.

Design Basis

The double-wall piping arrangement is designed to gravity-drain to a safe-geometry set of tanks or a safe-geometry containment berm.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.6 IROFS CS-10, Closed Safe Geometry Heating/Cooling Loop with Monitoring and Alarm

IROFS CS-10, "Closed Safe Geometry Heating or Cooling Loop with Monitoring and Alarm," is identified by the accident analyses in Chapter 13.0. 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 the heat transfer fluid boundary if the primary boundary fails.

Accidents Mitigated

The dual-purpose safety function of the closed-loop system is to prevent (1) fissile process solution from causing accidental nuclear criticality, and (2) high-dose process solution from exiting the hot cell containment, confinement, or shielded boundary (or to prevent low-dose solution from exiting the facility, for systems located outside of the hot cell containment, confinement, or shielded boundary), and causing excessive dose to workers and the public, and/or causing a release to the environment.

System Components

The closed loop steam and cooling water loop design is described in Chapter 9.0.

Functional Requirements

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, or uranium concentration) 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 prior to discharge to a non-safe geometry.

Design Basis

The closed loop steam and cooling water loop design is described in Chapter 9.0.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS CS-11, "Simple Overflow to Normally Empty Safe Geometry Tank with Level Alarm," is identified by the accident analyses described in Chapter 13.0. As a PEC, 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) for each vented tank containing fissile or potentially fissile process solution for which this IROFS is assigned.

Accident Mitigated

The overflow drain will prevent the process solution from entering the respective non-geometrically favorable sections of the process ventilation system and any chemical addition ports (where chemical addition ports enter through anti-siphon devices).

System Components

Locations of the overflow and overflow collection tanks will be provided with the final design.

Functional Requirements

The safety function of this feature is to prevent accidental nuclear criticality in non-geometrically favorable sections of the process ventilation system. The overflow will be directed to a safe-geometry storage tank. The overflow storage tank will normally be maintained 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 can be taken to restore operability of the safety feature by emptying the tank.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS CS-12, "Condensing Pot or Seal Pot in Ventilation Vent Line," is identified by the accident analyses described in Chapter 13.0. As a PEC, a safe-geometry condensing pot or seal pot will be installed downstream of each tank for which this IROFS is assigned 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.

Accident Mitigated

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 over-capacity condition from defeating both.

System Components

Locations of the condensing pots or seal pots and associated drain points will be provided with the final design.

Functional Requirements

The safety function of the condensing or seal pots is to prevent accidental nuclear criticality in non-geometrically favorable sections 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 overflow capacity 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 can 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.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS CS-13, "Simple Overflow to Normally Empty Safe Geometry Floor with Level Alarm in the Hot Cell Containment Boundary," is identified by the accident analyses described in Chapter 13.0. As a PEC, a simple overflow line will be installed above the high alarm setpoint for each vented tank containing fissile or potentially fissile process solution for which this IROFS is assigned. The overflow will be directed to one or more safe-geometry flooring configurations with safe-geometry sumps.

Accident Mitigated

This IROFS prevents accidental criticality by ensuring that overflowing fissile solutions are captured in a safe-geometry slab configuration with safe-geometry sumps.

System Components

System component information will be provided in the Operating License Application.

Functional Requirements

The floor areas (separated as needed to support operations in different hot cell areas) will normally be maintained empty. The floor area(s) will be equipped with a sump level alarm to inform the operator when use of the IROFS has been initiated.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.10 IROFS CS-14, Active Discharge Monitoring and Isolation

IROFS CS-14, "Active Discharge Monitoring and Isolation," is identified by the accident analyses described in Chapter 13.0. Additional detailed information describing active discharge monitoring and isolation will be developed for the Operating License Application.

System Components

System component information will be provided in the Operating License Application.

Functional Requirements

Functional requirements information will be provided in the Operating License Application.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS CS-15, "Independent Active Discharge Monitoring and Isolation," is identified by the accident analyses described in Chapter 13.0. Additional detailed information describing independent active discharge monitoring and isolation will be developed for the Operating License Application.

System Components

System component information will be provided in the Operating License Application.

Functional Requirements

Functional requirements information will be provided in the Operating License Application.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.12 IROFS CS-18, Backflow Prevention Device

IROFS CS-18, "Backflow Preventions Device," is identified by the accident analyses described in Chapter 13.0.

See description in Section 0.

Accident Mitigated

See description in Section 0.

System Components

See description in Section 0.

Functional Requirements

See description in Section 0.

Design Basis

See description in Section 0.

Test Requirements

See description in Section 0.

6.3.1.2.13 IROFS CS-19, Safe-Geometry Day Tanks

IROFS CS-19, "Safe Geometry Day Tanks," is identified by the accident analyses described in Chapter 13.0. See description in Section 0.

Accident Mitigated

See description in Section 0.

System Components

See description in Section 0.

Functional Requirements

See description in Section 0.

Design Basis

See description in Section 0.

Test Requirements

See description in Section 0.

6.3.1.2.14 IROFS CS-20, Evaporator/Concentrator Condensate Monitoring

IROFS CS-20, "Evaporator/Concentrator Condensate Monitoring," is identified by the accident analyses described in Chapter 13.0. As an AEC, the condensate tanks will use a continuous 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 discharged to a non-favorable geometry system or in the condenser for leaking to the non-safe geometry cooling loop.

Accident Mitigated

The safety function of this IROFS is to prevent an accidental nuclear criticality because of excessive uranium in the condensate carryover to a non-geometrically favorable waste collection tank.

System Components

System components consist of:

- Condensate sample tank 1A (UR-TK-340)
- Condensate delay tank 1 (UR-TK-360)
- Condensate sample tank 1B (UR-TK-370)
- Condensate sample tank 2A (UR-TK-540)
- Condensate delay tank 2 (UR-TK-560)
- Condensate sample tank 2B (UR-TK-570)
- Condensate sampling systems
- Condensate monitors

Functional Requirements

The detection system works 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 secures 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. Locations where these IROFS are used will be determined during final design.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.1.2.15 IROFS CS-26, Processing Component Safe Volume Confinement

IROFS CS-26, "Processing Component Safe Volume Confinement," is identified by the accident analyses described in Chapter 13.0 (see description in Section 6.3.1.2.2).

Accident Mitigated

See description in Section 6.3.1.2.2.

System Components

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 the fissile solutions. Components that may be controlled to a safe volume will be described in the Operating License Application.

Functional Requirements

The safety function of a 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 will conservatively include 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.

Design Basis

The safe-volume confinement components will be determined in final design after finalizing the referenced CSEs.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

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

IROFS CS-27, "Closed Heating or Cooling Loop with Monitoring and Alarm," is identified by the accident analyses in Chapter 13.0. As a PEC, closed cooling water loops with monitoring for breakthrough of process solution will be provided on the evaporator or concentrator condensers to contain process solution that leaks across this boundary, if the boundary fails. This IROFS will be 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. The inherent characteristics of the leak path will reduce back-leakage into the closed loop system, and the risk of product solutions entering the condenser will be very low by evaporator and concentrator design.

System Components

The purpose of this safety function is to monitor the health 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.

Functional Requirements

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, or uranium concentration) 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. 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 prior to discharge to a non-safe geometry.

Design Basis

Design basis information will be provided in the Operating License Application.

Test Requirements

The above analysis is based on information developed for the Construction Permit Application. Additional detailed information on test requirements will be developed for the Operating License Application.

6.3.2 Surveillance Requirements

A review of surveillance requirements to ensure the availability and reliability of safety controls when required to perform safety functions will be included in the Operating License Application.

6.3.3 Technical Specifications

The technical specifications will be provided in the Operating License Application.

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Public
Chapter 7

NWMI
NORTHWEST MEDICAL ISOTOPES

Chapter 7.0 – Instrumentation and Control Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0
June 2015

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Chapter 7.0 – Instrumentation and Control Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0

Date Published:
June 29, 2015

***Document Number*: NWMI-2013-021**		***Revision Number*: 0**
***Title*: Chapter 7.0 – Instrumentation and Control Systems Construction Permit Application for Radioisotope Production Facility**		
***Approved by*: Carolyn Haass**	***Signature*:** Carolyn C. Haass	

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REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required

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CONTENTS

FIGURES

TABLES

TERMS

Acronyms and Abbreviations

^{99}Mo	molybdenum-99
ADUN	acid-deficient uranyl nitrate
ALARA	as low as reasonably achievable
BMS	building management system
CAAS	criticality accident alarm system
CAM	continuous air monitor
CFR	Code of Federal Regulations
CGD	commercial grade dedication
COTS	commercial off-the-shelf
DCS	digital control system
ESF	engineered safety feature
FPC	facility process control
HMI	human-machine interface
I	iodine
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
IROFS	items relied on for safety
ISA	integrated safety analysis
IX	ion exchange
Kr	krypton
LEU	low-enriched uranium
Mo	molybdenum
NAVLAP	National Voluntary Laboratory Accreditation
NO_x	nitrogen oxide
NRC	U.S. Nuclear Regulatory Commission
NWMI	Northwest Medical Isotopes, LLC
OIT	operator interface terminal
PLC	programmable logic controller
RAM	radiation area monitor
RPF	Radioisotope Production Facility
SDOE	secure development and operational environment
SIF	safety instrumented function.
SIL	safety integrity level.
SIS	safety instrumented system
SNM	special nuclear material
SSC	structures, systems, and components
TCE	trichloroethylene
U.S.	United States
UPS	uninterruptible power supply
V&V	verification and validation
Xe	xenon

Units

m	meter
min	minute
rad	radiation absorbed dose

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 SUMMARY DESCRIPTION

The Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) preliminary instrumentation and control (I&C) configuration includes the special nuclear material (SNM) preparation and handling processes (e.g., target fabrication, and uranium recovery and recycle), radioisotope extraction and purification processes (e.g., target receipt and disassembly, target dissolution, molybdenum [Mo] recovery and purification, and waste handling), process utility systems, criticality accident alarm system (CAAS), and systems associated with radiation monitoring.

The SNM processes will be enclosed predominately by hot cells and glovebox designs except for the target fabrication area. The facility process control (FPC) system will provide monitoring and control of the process systems within the RPF. In addition, the FPC system will provide monitoring of safety-related components within the RPF. The process strategy for the RPF involves the use of batch or semi-batch processes with relatively simple control steps.

The building management system (BMS) (a subset of the FPC system) will monitor the RPF ventilation system and mechanical utility systems. The BMS primary functions will be to monitor the facility ventilation system and monitor and control (turn on and off) the mechanical utility systems.

Engineered safety feature (ESF) systems will operate independently from the FPC system or BMS. Each ESF safety function will use hard-wired analog controls/interlocks to protect workers, the public, and environment. The ESF parameters and alarm functions will be integrated into and monitored by the FPC system or BMS.

The preliminary concept for the RPF I&C system configuration is shown in Figure 7-1. The green circles identify the FPC and the BMS distributed process control or programmable logic controller (PLC) systems. The solid lines and dashed lines show how the SNM processes, support systems, utilities, radiation and criticality systems, and building functions relate to the FPC and BMS and to local human-machine interface (HMI) stations. Solid lines indicate the control functions, and dashed lines indicate the monitoring functions.

The FPC system will perform as the overall production process controller. This system will monitor and control the process instrumented functions within the RPF, including monitoring of process fluid transfers and controlled inter-equipment pump transfers of process fluids. Process control systems are described further in Section 7.3.

The fire protection system will have its own central alarm panel (green circle). The fire protection system will report the status of the fire protection equipment to the central alarm station and the RPF control room. The fire protection system is discussed further in Section 7.2.3.3.

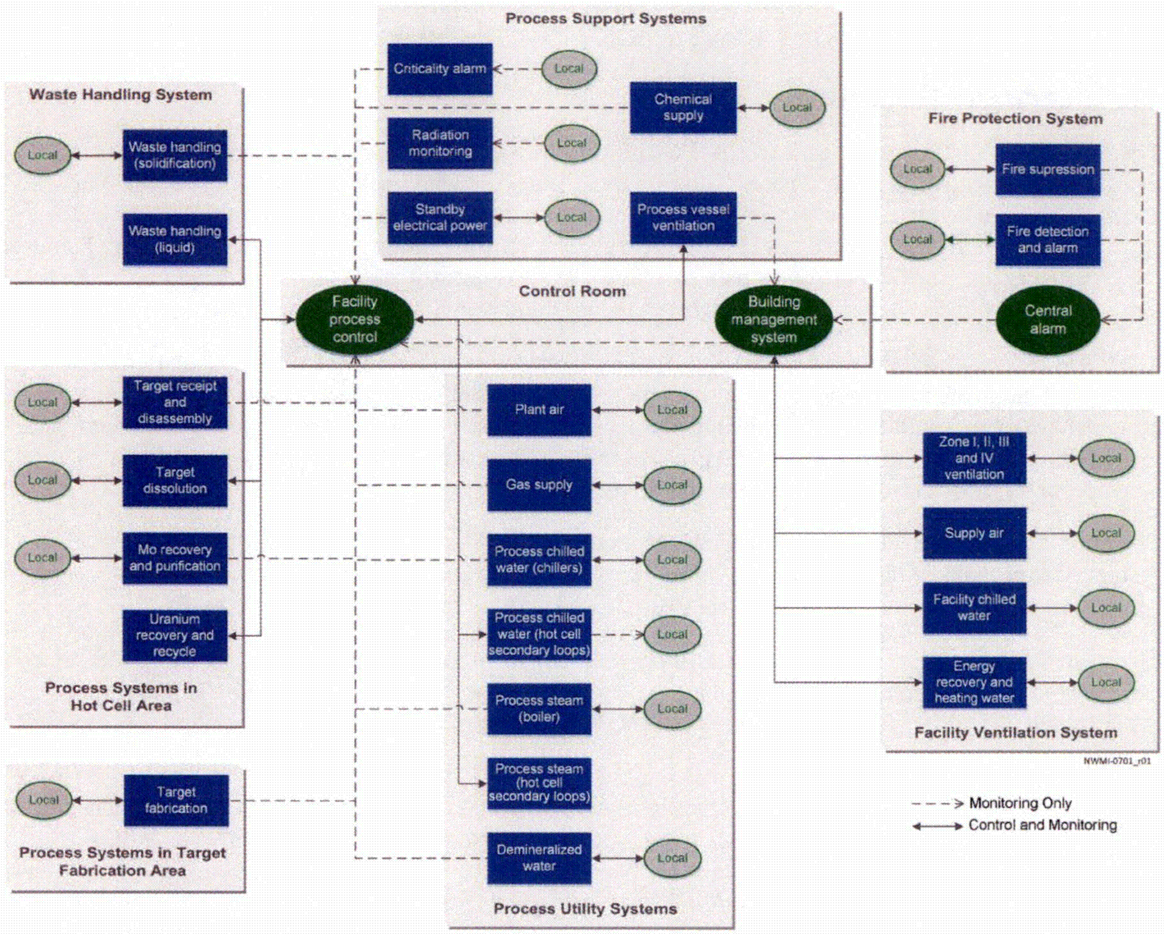

Figure 7-1. Radioisotope Production Facility Instrumentation and Control System Configuration

Special nuclear material preparation and handling processes – The FPC system will control and/or monitor the SNM preparation and handling processes, the following.

- **Target fabrication** – Batch processes located in the target fabrication area will be controlled by operators at local HMIs, with surveillance monitoring in the control room.
- **Uranium recovery and recycle** – Batch processes located inside the hot cell area will be controlled by operators in the control room.

Radioisotope extraction and purification processes – The FPC system will control and/or monitor the radioisotope production processes, including the following.

- **Target receipt and disassembly** – Hardware/target movement located in irradiated target basket receipt bay area, target cask preparation airlock, target receipt hot cell, and target disassembly hot cell will normally be controlled by operators at local HMIs, with surveillance monitoring in the control room.
- **Target dissolution** – Batch process located inside the dissolution hot cell will occur at local HMIs in the operating gallery, and offgas operations in the tank hot cell will be controlled by operators in the control room, with surveillance monitoring at both locations.

- **Mo recovery and purification** – Batch processes located inside the Mo hot cells will be controlled by operators at a local HMI in the operating gallery, with surveillance monitoring in the control room.
- **Waste handling** – This system includes liquid waste handling, liquid waste solidification, and solid waste handling. Operators in the control room will control liquid waste handling, while operators at local HMIs in the low-dose liquid solidification room (W107) will monitor and control liquid waste solidification, and solid waste nondestructive examination and solidification.

Process utility and support systems – The FPC system will control and monitor the process utility and process support systems. Operators in the control room will control the following subsystems:

- Process chilled water hot cell secondary loops
- Process steam hot cell secondary loops
- Process vessel ventilation system

Operators at local HMIs will control the following subsystems, with surveillance monitoring in the control room using the FPC system or BMS.

- Plant air system
- Gas supply system
- Process chilled water chillers
- Process steam boilers
- Demineralized water system
- Chemical supply system
- Standby electrical power system

Criticality accident alarm system – The CAAS will be provided as an integrated vendor package. The detectors and alarm response are integral to the individual units/locations. The FPC system will monitor the CAAS status in the control room. The CAAS is described further in Section 7.3.

Radiation monitoring system – The FPC system will monitor the various radiation monitoring systems, including continuous air monitors (CAM), air samplers, radiation area monitors (RAM), and exhaust stack monitors. The CAMs and RAMs will be strategically placed throughout the RPF to alert personnel of any potential radiation hazards. The CAMs and RAMs will alarm in the control room and locally at locations throughout the RPF. The radiation monitoring systems are described further in Section 7.6.

Facility ventilation system and mechanical utility systems – The control function for most of the RPF ventilation system and mechanical utility systems will be local HMIs and hard-wired interlocks for the ESF functions. The BMS will monitor the systems and provide ventilation and mechanical utility system status as an input to the FPC process controls.

The following subsystems will be monitored by the BMS:

- Facility ventilation Zones I, II, III, and IV
- Supply air system
- Facility chilled water system
- Energy recovery and heating water

Safety-Related Components and Engineering Safety Features

The ESF safety functions will operate independently from the FPC systems as hard-wired analog controls or interlocks. The FPC system will be a digital control system (DCS) that monitors safety-related components within the RPF. The ESFs will be integrated into the FPC systems and provide a common point of HMI, monitoring, and alarming at the control room and, as necessary, local HMI workstations.

Control Console and Display Instruments

The control room will be the primary interface location for the RPF support systems and provide centralized process controls, monitoring, alarms, and acknowledgement. Mechanical utility systems with vendor packages and integrated controls will be controlled at associated local HMIs. The BMS will provide primarily on/off control and system monitoring from the control room.

The tank hot cell processes will be controlled primarily in the control room, with surveillance monitoring of the FPC subsystems. The FPC system will have annunciation, alarms, and operator interface displays. From the consoles, operators will view and trend essential measurement values from the operator interface display, and evaluate real-time data from the essential measurements used to control and monitor the RPF process. This system is further described in Section 7.5.

Process utility and support systems with vendor package and integrated controls will be operated at associated local HMIs. These systems are discussed further in Section 7.5. Local HMIs are anticipated in the following locations:

- Irradiated target basket receipt bay A/B (R102A/B)
- Cask preparation airlock (R012)
- Operating gallery (G101 A/B/C)
- Target fabrication (T104 A/B)
- Low-dose liquid waste solidification (W107)
- Chemical supply room (L102)
- Local to equipment with integrated control systems

7.2 DESIGN OF INSTRUMENTATION AND CONTROL SYSTEMS

The design criteria and the codes and standards for I&C systems are outlined in Chapter 3.0, "Design of Structures, Systems, and Components," and discussed below.

7.2.1 Design Criteria

The applicable design criteria and guidelines that apply to the RPF I&C systems are summarized in column one of Table 7-1. Additional, design criteria for I&C systems are provided in Chapter 3.0. The detailed and specific design criteria for I&C systems will be confirmed in the Operating License Application.

7.2.2 Design Basis and Safety Requirements

The design basis for I&C systems used in the RPF are presented in the second column of Table 7-1. The second column maps the criteria to I&C systems or components and how compliance will be ensured. Note that the FPC system callouts may also apply to the BMS. The design basis requirements for facility and process systems are described in Chapter 4.0, "Radioisotope Production Facility Description," and Chapter 9.0, "Auxiliary Systems."

The I&C system will use hard-wired interlocks for actuated engineered safety functions. Section 7.4 summarizes the I&C ESFs.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 379-2014, *IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems* **Description**: Application of the single-failure criterion to electrical power, instrumentation, and control portions of nuclear power generating safety systems. **Keywords**: Actuator, cascaded failure, common-cause failure, design basis event, detectable failure, effects analysis, safety system, single-failure criterion, system actuation, system logic	**Application:** • Design of FPC system, ESFs, and other instrumentation SSCs that are identified as IROFS **Compliance:** • Ensure FPC system is a DCS designed, rated, and approved for use in safety instrumented systems, as determined by ANSI/ISA 84.00.01 • Use a safety PLC, as recognized by IEC 61508, in the FPC system with redundant power supplies, processors, and input/output channels • Evaluate controls that are classified as IROFS in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002, against single-failure criteria **Exception:** • NUREG-1537 allows for sharing and combining of systems and components with justification • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
IEEE 577-2012, *IEEE Standard Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Facilities* **Description**: Sets minimum acceptable requirements for the performance of reliability analyses for safety systems when used to address the reliability considerations discussed in industry standards and guidelines. The requirement that a reliability analysis be performed does not originate with this standard. However, when reliability analysis is used to demonstrate compliance with reliability requirements, this standard describes an acceptable response to the requirements. **Keywords**: Nuclear facilities, reliability analysis, safety systems	**Application:** • Use for design of FPC system, ESFs, and other instrumentation SSCs that are identified as IROFS **Compliance:** • Perform a reliability analysis of the proposed design solution for IROFS functions, as identified in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002. The analysis can be qualitative or quantitative in nature, as described in the standard

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 603-2009, ***IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*** **Description**: Establishes minimum functional and design criteria for the power, instrumentation, and control portions of nuclear power generating station safety systems. Criteria are to be applied to those systems required to protect public health and safety by functioning to mitigate the consequences of design basis events. The intent is to promote appropriate practices for design and evaluation of safety system performance and reliability. The standard is limited to safety systems; many of the principles may have applicability to equipment provided for safe shutdown, post-accident monitoring display instrumentation, preventive interlock features, or any other systems, structures, or equipment related to safety. **Keywords**: Actuated equipment, associated circuits, Class 1E, design, failure, maintenance bypass, operating bypass, safety function, sense and command features, sensor	**Application:** • Use for design of FPC system, ESFs, and other instrumentation SSCs that are identified as IROFS • Apply minimum functional and design criteria to safety systems **Compliance:** • Ensure design conforms to the practices detailed in the standard for the IROFS functions identified in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002 **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
IEEE 384-2008, ***IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits*** **Description**: Describes independence requirements of circuits and equipment comprising or associated with Class 1E systems. Identifies criteria for independence that can be achieved by physical separation, and electrical isolation of circuits and equipment that are redundant. The determination of what is to be considered redundant is not addressed. **Keywords**: Associated circuit, barrier, Class 1E, independence, isolation, isolation device, raceway, separation	**Application:** • Use for design of FPC system, ESFs, and other instrumentation SSCs that are identified as IROFS • Apply minimum criteria for separation and independence of systems in a physical way **Compliance:** • Ensure design conforms to the practices detailed in the standard for the IROFS functions identified in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002 **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 323-2003, ***IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*** **Description**: Identifies requirements for qualifying Class 1E equipment and interfaces that are to be used in nuclear power generating stations. The principles, methods, and procedures are intended for use in qualifying equipment, maintaining and extending qualification, and updating qualification, as required, if the equipment is modified. The qualification requirements of the standard demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions, including design basis events, reducing the risk of common-cause equipment failure. **Keywords**: Age conditioning, aging, condition monitoring, design basis event, equipment qualification, qualification methods, harsh environment, margin, mild environment, qualified life, radiation, safety-related function, significant aging mechanism, test plan, test sequence, type testing	**Application:** • Use for equipment qualification when needed to qualify equipment for applications or environments to which the equipment may be exposed • Use for qualification of Class 1E equipment located in harsh environments and for certain post-accident monitoring equipment; may also be used for the qualification of equipment in mild environments **Compliance:** • Ensure design conforms to the practices detailed in the standard for those systems determined to be Class IE and located in harsh environments for safety functions identified in Chapters 6.0 and 13, or NWMI-2015-SAFETY-002 • Apply to SSCs within the hot cell area; not all safety components reside in the hot cell area • Apply standard using a graded approach **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 344-2004, *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations* **Description**: Identifies recommended practices for establishing procedures that will yield data to demonstrate that the Class 1E equipment can meet performance requirements during and/or following one safe shutdown earthquake event, preceded by a number of operating basis earthquake events. This recommended practice may be used to establish tests, analyses, or experience-based evaluations that will yield data to demonstrate Class 1E equipment performance claims or to evaluate and verify performance of devices and assemblies as part of an overall qualification effort. Common methods currently in use for seismic qualification by test are presented. Two approaches to seismic analysis are described: one based on dynamic analysis, and the other on static coefficient analysis. Two approaches to experience-based seismic evaluation are described, one based on earthquake experience and the other on test experience. **Keywords**: Class 1E, earthquake, earthquake experience, equipment qualification, inclusion rules, nuclear, operating basis earthquake, prohibited features, qualification methods, required response spectrum, response spectra, safe shutdown earthquake, safety function, seismic, seismic analysis, test response spectrum, test experience	**Application:** • Apply seismic design requirements for equipment used in Class 1E systems **Compliance:** • Use in design of FPC system, ESFs, and other instrumentation SSCs that are identified as a Class 1E system **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
IEEE 338-2012, *IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems* **Description**: Provides criteria for the performance of periodic surveillance testing of nuclear power generating station safety systems. The scope of periodic surveillance testing consists of functional tests and checks, calibration verification, and time response measurements, as required, to verify that the safety system performs its defined safety function. Post-maintenance and post-modification testing are not covered by this document. This standard amplifies the periodic surveillance testing requirements of other nuclear safety-related IEEE standards. **Keywords**: Functional tests, IEEE 338, periodic testing, risk-informed testing, surveillance testing	**Application:** • Use for design of FPC system, ESFs, and other instrumentation SSCs that are identified as IROFS • Use methods and criteria to establish a periodic surveillance program **Compliance:** • Ensure design conforms to the practices detailed in the standard for the IROFS functions identified in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002 **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 497-2010, ***IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations*** **Description**: Establishes criteria for variable selection, performance, design, and qualification of accident monitoring instrumentation, and includes the requirements for display alternatives for accident monitoring instrumentation, documentation of design bases, and use of portable instrumentation. **Keywords**: Accident monitoring, display criteria, selection criteria, type variables	**Application:** • Use as selection, design, performance, qualification, and display criteria for accident monitoring instrumentation • Apply guidance on the use of portable instrumentation and for examples of accident monitoring display configurations **Compliance:** • Ensure design conforms to standard for the monitoring functions determined to be required for health and safety of workers or the public during normal operation and design basis accidents **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
IEEE 7-4.3.2-2010, ***IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations*** **Abstract**: Specifies additional computer-specific requirements to supplement IEEE 603-2009. The standard defines the term computer as a system that includes computer hardware, software, firmware, and interfaces, and establishes minimum functional and design requirements for computers used as components of a safety system. **Keywords**: Commercial-grade item, diversity, safety systems, software, software tools, software verification and validation	**Application:** • In conjunction with IEEE 603-2009, use to establish minimum functional and design requirements for computers that are components of a safety system • Design FPC system as a DCS, and apply this standard to system development, specifically software development • Apply standard to CGD and implement an approach **Compliance:** • Develop FPC system software using this standard **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
IEEE 828-2012, ***IEEE Standard for Configuration Management in Systems and Software Engineering*** **Description**: Establishes minimum requirements for configuration management in systems and software engineering. This standard applies to any form, class, or type of software or system, and explains configuration management, including identifying and acquiring configuration items, controlling changes, reporting the status of configuration items, and performing software builds and release engineering. This standard addresses what configuration management activities are to be done, when they are to happen in the life-cycle, and what planning and resources are required. The content areas for a configuration management plan are also identified. The standard supports IEEE STD 12207 and ISO/IEC/IEEE 15288, and adheres to the terminology in ISO/IEC/IEEE STD 24765 and the information item requirements of IEEE STD 15939. **Keywords**: Change control, configuration accounting, configuration audit, configuration item, IEEE 828, release engineering, software builds, software configuration management, system configuration management	**Application:** • Use to establish configuration management processes, define how configuration management is to be accomplished, and identify who is responsible for performing specific activities, when the activities are to happen, and what specific resources are required • Design FPC system as a DCS, and apply standard during the development of software for systems with IROFS functions **Compliance:** • Develop FPC system software using this standard for safety function implementation
IEEE 1028-2008, ***IEEE Standard for Software Reviews and Audits*** **Description**: Identifies five types of software reviews and audits, together with procedures required for the execution of each type. This standard is concerned only with reviews and audits; procedures for determining the necessity of a review or audit are not defined, and the disposition of the results of the review or audit is not specified. Types included are management reviews, technical reviews, inspections, walk-throughs, and audits. **Keywords**: Audit, inspection, review, walk-through	**Application:** • Use to identify minimum acceptable requirements for systematic software reviews • Identify organizational means for conducting a review and documenting the findings • Design FPC system as a DCS, and apply standard during the development of software for systems with IROFS functions **Compliance:** • Develop FPC system using this standard
ANS 10.4-2008, ***Verification and Validation of Non-Safety-Related Scientific and Engineering Computer Programs for the Nuclear Industry*** **Description**: Provides guidelines for V&V of non-safety-related scientific and engineering computer programs developed for use by the nuclear industry. Scope is restricted to research and other non-safety-related, noncritical applications. **Keywords**: Software integrity level, software life-cycle, validation, verification, V&V	**Application:** • Perform software V&V to build quality into the software during the software life-cycle • Use to verify and validate software development for non-safety-related systems • Use for software development in the RPF that is not safety significant (e.g., not safety-related or IROFS) **Compliance:** • Develop non-safety-related software using this standard

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
ANSI/ISA 67.04.01-2006, *Setpoints for Nuclear Safety-Related Instrumentation* **Description**: Defines requirements for assessing, establishing, and maintaining nuclear safety-related and other important instrument setpoints associated with nuclear power plants or nuclear reactor facilities. **Keywords**: Setpoint, drift, analog channel, reliability analysis	**Application:** • Use methods and criteria to establish setpoints for safety systems and to maintain the documentation • Apply to the design of the FPC system and other instrumentation SSCs that are identified as IROFS for the RPF **Compliance:** • Ensure design conforms to the practices detailed in the standard for IROFS functions with inherent setpoints identified in Chapters 6.0 and 13.0, or NWMI-2015-SAFETY-002
ANSI/ISA 84.00.01-2004, *Functional Safety: Safety Instrumented Systems for the Process Industry Sector* **Part 1: "Framework, Definitions, System, Hardware and Software Requirements"** **Part 2: "Guidelines for the Application of ANSI/ISA-84.00.01-2004 Part 1 (IEC 61511-1 Mod) – Informative"** **Part 3: "Guidance for the Determination of the Required Safety Integrity Levels – Informative"** **Description**: Provides requirements for the specification, design, installation, operation, and maintenance of a safety instrumented system, so the system can be confidently entrusted to place and/or maintain the process in a safe state. This standard has been developed as a process sector implementation of IEC 61508. **Keywords**: Safety instrumented system (SIS), safety integrated level (SIL), safety instrumented function (SIF)	**Application:** • Apply to the design of safety systems (standard specifically designed for industrial processes) • Standard is made up of three parts: – Use Part 1 to lay the groundwork for the safety system life-cycle, overall structure of safety systems, definitions used, and to implement safety system design engineering – Use Part 2 guidance for the specification, design, installation, operation, and maintenance of safety instrumented functions and related safety instrumented systems, as defined in Part 1 – Use Part 3 to develop underlying concepts of risk in relation to safety integrity, identify tolerable risk, and determine the safety integrity levels of the safety functions • Design physical hardware of the FPC system based on this standard and IEC 61508 • Evaluate the IROFS functions required to be implemented by the FPC system using Parts 1, 2, and 3 of this standard • Use to demonstrate reliability and risk reduction of the FPC system, while having similar or higher documented and tested ability to reduce risk as fulfillment through other channels **Compliance:** • Use for the design and implementation for IROFS functions that are required of the FPC system

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
NUREG-0700, *Human-System Interface Design Review Guidelines* **Description**: Provides guidance to the NRC on the evaluation of human factors engineering aspects of nuclear power plants in accordance with NUREG-0800. Detailed design review procedures are provided in NUREG-0711. As part of the review process, the interfaces between plant personnel and the plant systems and components are evaluated for conformance with human factors engineering guidelines. **Keywords**: Display, HMI, human-interface system, human-system interface	**Application:** • Use comprehensive design review guidance to develop information displayed in human-interface systems • Develop informative and effective designs that will assist operators in the performance of their duties **Compliance:** • Design FPC system to provide information to operators in a display format • Display development used in connection with the FPC system will be provided in the Operating License Application
NUREG/CR-6463, *Review Guidelines on Software Languages for Use in Nuclear Power Plant Safety Systems* **Description**: Provides guidance to the NRC on auditing programs for safety systems written in the following six high-level languages: Ada, C and C++, PLC Ladder Logic, Sequential Function Charts, Pascal, and PL/M. The guidance could also be used by those developing safety significant software as a basis for project-specific programming guidelines. **Keywords**: Pascal, C, Ladder Logic, PL/M, Ada, C++, PLC, programming, sequential function charts	**Application:** • Use guidance to review high-integrity software in a nuclear facility • Develop FPC system as a DCS, with associated programming development needs for the RPF • Use guideline as a means to review FPC system programming code **Compliance:** • Develop FPC system software programs using this guidance **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
NUREG/CR-6090, *The Programmable Logic Controller and Its Application in Nuclear Reactor Systems* **Abstract**: Outlines recommendations for review of the application of PLCs to the control, monitoring, and protection of nuclear reactors. **Keywords**: PLC, programming, protection systems	**Application:** • Use guidance to implement PLCs for nuclear application and as a forum for what constitutes good practices of previously installed systems • Use guidance during selection process for hardware, failure analysis, and product life-cycle within the facility **Compliance:** • Design FPC system to use a PLC-type DCS • Select design and implement PLCs based on this guide, as applicable **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
EPRI TR-106439, ***Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications*** **Description**: Provides a consistent, comprehensive approach for the evaluation and acceptance of commercial digital equipment for nuclear safety systems. **Keywords**: Commercial off-the-shelf (COTS), programming, software, commercial grade dedication	**Application:** • Use to identify appropriate critical characteristics with subsequent verification through testing, analysis, vendor assessments, and careful review of operating experience • Use guidance for digital upgrades to safety-related systems and for non-safety-related applications that require high reliability or are compatible with utility-specific change processes, including graded approaches for quality assurance **Compliance:** • Ensure that digital systems components that require CGD apply the guidance of this standard, as applicable
Regulatory Guide 1.152, ***Criteria for Use of Computers in Safety Systems of Nuclear Power Plants*** **Description**: Describes a method that the NRC staff deems acceptable for complying with NRC regulations for promoting high functional reliability, design quality, and a secure development and operational environment for the use of digital computers in the safety systems of nuclear power plants. **Keywords**: Secure development and operational environment (SDOE), computers	**Application:** • Use for I&C system designs with computers in safety-related systems that make extensive use of advanced technology • Use for RPF designs (that are expected to be significantly and functionally different from current day process designs) with microprocessors, digital systems and displays, fiber optics, multiplexing, and different isolation techniques to achieve sufficient independence and redundancy **Compliance:** • Develop FPC system and associated HMI using this guidance **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
Regulatory Guide 1.53, ***Application of the Single-Failure Criterion to Safety Systems*** **Description**: Provides methods acceptable to the NRC staff for satisfying NRC regulations with respect to the application of the single-failure criterion to the electrical power and I&C portions of nuclear power plant safety systems. **Keywords**: IEEE 379-2014, single-failure criterion	**Application:** • Apply single-failure criterion to safety-related I&C systems • Apply to end-devices used by the FPC system that are identified as IROFS **Compliance:** • Evaluate FPC system, ESFs, and IROFS end-devices using this guidance

Table 7-1. Instrumentation and Control System Design Criteria (10 pages)

Design criteria description[a]	Design bases as applied to RPF
Regulatory Guide 1.97, ***Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*** **Description**: Provides a method that the NRC staff considers acceptable for use in complying with NRC regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. **Keywords**: IEEE 497-2010, accident monitoring	**Application:** • Use this guidance for development of accident monitoring for the RPF **Compliance:** • Design FPC system, CAAS, CAMs, and RAMs using this guidance **Exception:** • The RPF is not considered a nuclear power reactor but a production facility. The facility will not have all of the systems detailed in this standard and guidance will be applied as appropriate.
Regulatory Guide 5.71, ***Cyber Security Programs for Nuclear Facilities*** **Description**: Provides an approach that the NRC staff deems acceptable for complying with NRC regulations regarding the protection of digital computers, communications systems, and networks from a cyberattack, as defined by 10 CFR 73.1. **Keywords**: Cybersecurity, 10 CFR 73.54(a)(2), design basis threat	**Application:** • Use this guidance for development of cybersecurity protections **Compliance:** • Design the FPC system and associated HMI based on this guidance

[a] Full references provided in Section 7.7.

CAAS	=	criticality accident alarm system.
CAM	=	continuous air monitor.
CFR	=	Code of Federal Regulations.
CGD	=	commercial grade dedication.
COTS	=	commercial off-the-shelf.
DCS	=	digital control system.
ESF	=	engineered safety feature.
FPC	=	facility process control.
HMI	=	human-machine interface.
I&C	=	instrumentation and control.
IEEE	=	Institute of Electrical and Electronics Engineers.
IROFS	=	items relied on for safety.
NRC	=	U.S. Nuclear Regulatory Commission.
PLC	=	programmable logic controller.
RAM	=	radiation alarm monitor.
RPF	=	Radioisotope Production Facility.
SDOE	=	secure development and operational environment.
SIF	=	safety instrumented function.
SIL	=	safety integrity level.
SIS	=	safety instrumented system.
SSC	=	structures, systems, and components.
V&V	=	verification and validation.

Specific requirements will be developed during the next stages of design for the Operating License Application. The I&C design will be expanded and analyzed to document fulfillment of the design criteria and design basis requirements for the Operating License Application.

7.2.3 System Description

As described in Section 7.1, the RPF I&C system basic components include the FPC system, ESF actuation systems, control console and display instruments, and BMS. These systems provide an interface for the operator to monitor and control those systems. The FPC system will be a DCS that functions independently and electrically isolated from power systems. The items relied on for safety (IROFS)/ESF safety functions will be activated via hardwire interlocks.

7.2.3.1 Facility Process Control System

The FPC system controls and monitors the target fabrication system, hot cell area (e.g., Mo recovery and purification, uranium recovery and recycle system), process utility and support systems, and waste handling activities. The FPC system functions also include radiation monitoring, CAAS, HMIs, safe shutdown control and initiation, supervisory information, and alarms. The BMS is a subsystem to the FPC system and monitors the facility ventilation system.

The primary control location of the FPC system is in the control room. The control room FPC system operates with a synchronized hot standby redundant system structure. The hot standby workstations provide redundant hardware with identical PLC software systems as automatic backup control systems. The primary and backup PLC systems monitor each other. On loss of synchronizing signal from one system, the other system continues with control and monitoring. This automatic backup control system minimizes the likelihood of downtime during Mo production processing.

7.2.3.2 Control Room/Operator Interface Description

The operator will have direct visualization of critical values and the ability to input control functions into the FPC system. The FPC system dedicated displays will perform the following functions:

- **Static display** – This display will show critical measurement values and perform the function of an annunciator panel. This fixed display panel will not provide any interactive control functionality.
- **Alarm/event annunciator display panel** – This panel will display any event or alarm that is defined for the process. The display will enable the operator to acknowledge current events and alarms, and will provide a historical record of events.
- **Dynamic interface display panel or HMI** – This panel will enable the operator to perform tasks, change modes, enable/disable overrides, and other tasks that require operator input to allow, perform, or modify a task or event.

The set of displays will be arranged in a workstation. This workstation will also include a keyboard and mouse that will be used to interface with the system.

7.2.3.3 Fire Protection System

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, including 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. As an example, signals from the fire control panel may initiate actions such as shutdown of the ventilation equipment or actuating the deluge valves. The fire protection system is described in Chapter 9.0, Section 9.3.

7.2.3.4 Facility Communication Systems

The RPF communication systems will relay information within the facility during normal and emergency conditions. The 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. Facility communications system is described in Chapter 9.0, Section 9.4.

7.2.3.5 Analytical Laboratory System

The analytical laboratory will support the production of the 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. The analytical laboratory equipment will be provided as vendor package units. Control room monitoring of the analytical laboratory will be limited to the facility systems, including ventilation and radiation monitoring systems. Analytical laboratory system is described in Chapter 9.0, Section 9.7.3.

7.2.4 System Performance Analysis

The RPF I&C system will monitor the processes and ESFs when required. The IROFS will be managed by the FPC system. The FPC system will provide the central decision-making processor that evaluates monitored parameters from the various plant instrumentation and from the radiation monitoring systems of the CAMs, CAAS, and RAMs. The analysis herein discusses safety as it relates to the IROFS design criteria and design basis. Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the RPF I&C systems are provided in Chapter 14.0, "Technical Specifications."

7.2.4.1 Facility Trip and Alarm Design Basis

The design basis information for the FPC system trip functions is based on the following two requirements from Title 10, *Code of Federal Regulations*, Part 70 (10 CFR 70), "Domestic Licensing of Special Nuclear Material."

- **Double-contingency principle** – Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible (baseline design criteria of 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," paragraph [9]).
- The safety program will ensure that each IROFS will be available and reliable to perform its intended function when needed and in the context of the performance requirements of this section (10 CFR 70.61, "Performance Requirements," paragraph [e]).

The FPC system trip and alarm annunciation are protective functions and will be part of the overall protection and safety monitoring systems for the RPF. The specific equipment design basis for the instrumentation and equipment used for the FPC system trip and alarming functions is discussed in Section 7.2.2.

The following discussion relates to the design basis used for monitoring specific signal values for RPF trips and alarms, requirements for performance, requirements for specific modes of operation of the RPF and the FPC system, and the general design criteria noted in Table 7-1.

7.2.4.1.1 Safety Functions Corresponding Protective or Mitigative Actions for Design Basis Events

IEEE 603-2009, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations* (Sections 4a and 4b). The results of the integrated safety analysis (ISA) for the RPF structures, systems, and components (SSC) are discussed in Chapter 13.0, "Accident Analysis." Conditions that require monitoring and the subsequent action to be taken are described in Chapter 13.0.

7.2.4.1.2 Variable Monitored to Control Protective or Mitigative Action

IEEE 603-2009 (Section 4d). The list of variables to be monitored in the RPF to eliminate or reduce the exposure for the operator will be provided in the Operating License Application.

7.2.4.1.3 Functional Degradation of Safety System Performance

IEEE 603-2009 (Section 4h). These design requirements will be factored in and will be evaluated in the Operating Licensing Application.

7.2.4.2 Analysis

7.2.4.2.1 Facility Process Control System Trip Function Conformance to Applicable Criteria

The FPC system will perform a trip as a protective function as part of the RPF safety analysis. The associated design criteria are discussed in Sections 7.2.1 and 7.2.2. The following discussions relate to conformance to the criteria for the FPC system trip function.

7.2.4.2.2 General Functional Requirement Conformance

IEEE 603-2009 (Section 5). The FPC system will initiate and control ESF activation and isolation when the system detects an off-normal event appropriate for activation. The FPC system trips are discussed in Section 7.2.4.1. These monitored values and subsequent trips are a result of the preliminary accident analysis in Chapter 13.0 and provide a means to mitigate or reduce the consequences from the design basis accident to acceptable levels.

7.2.4.2.3 Requirements on Bypassing Trip Functions Conformance

IEEE 603-2009 (Sections 5.8, 5.9, 6.6, and 6.7). Trip override or bypass is recognized as a design requirement. Channel bypass will be allowed based on the nature of the signal. No channel bypass will be allowed without a visual indication on the FPC system display and recording the bypass event in the historical log.

7.2.4.2.4 Requirements on Setpoint Determination and Multiple Setpoint Conformance

IEEE 603-2009 (Section 6.8). Table 7-1 discusses the criteria to be used for setpoint derivation. Setpoints will be calculated in accordance with ISA-RP-67.04.02, *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation*.

7.2.4.2.5 Requirements for Completion of Trip Conformance

IEEE 603-2009 (Section 5.2). The ESF and the interaction of a mitigative action going to completion will be provided in the design. The FPC system will monitor for a complete trip of the ESF. This information will be available on the operator display for the FPC system and at the local operator interface terminals (OIT) near the hot cell. An alarm/event annunciation will be displayed to the operator. Section 7.4.1 describes the activation of the ESF, alarm/event strategy, and operator requirements to manually reset the system after a facility trip.

7.2.4.2.6 Requirements for Manual Control of Trip Conformance

IEEE 603-2009 (Section 6.2). The FPC system will have the ability to perform a manual activation of the ESF. Section 7.4.1 describes the activation of the ESF, alarm/event strategy, and operator requirements to manually reset the system after a facility trip.

7.2.4.3 Conclusion

The I&C systems for the RPF will meet the stated design criteria and design basis requirements outlined in NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors – Format and Content*. A crosswalk of the I&C subsystems, along with a cross-reference to specific design criteria, is presented in Table 7-2.

Table 7-2. Instrumentation and Control Criteria Crosswalk with Design Basis Applicability and Function Means (4 pages)

Criteria[a]	Design basis applicability	Functional means
IEEE 379 Single failure criterion	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Safety DCS preapproved platform • Redundant independent isolation components • Redundant operator interface workstations • Redundant sensors • Alternative manual means for ESF initiation
IEEE 577 Reliability analysis criterion	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Safety DCS pre-approved platform for an SIS • Redundant independent isolation components • Redundant operator interface workstations • Redundant sensors • Alternative manual means for ESF initiation
IEEE 603 Standard criteria safety system	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	See Section 7.3 for details.
IEEE 384 Independence of Class 1E equipment and circuits	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• IEEE 603 and IEEE 379 were used during development of the Construction Permit Application. • Additional details will be developed for the Operating License Application.
IEEE 323 Qualifying Class 1E Equipment	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Standard supports selection and qualification of equipment to be Class 1E use qualified. • This standard will be reevaluated in the Operating License Application for applicability.
IEEE 344 Recommended practice for seismic qualification	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Standard supports selection and qualification of equipment to be Class 1E use qualified. • Standard will be reevaluated in the Operating License Application for applicability.

Table 7-2. Instrumentation and Control Criteria Crosswalk with Design Basis Applicability and Function Means (4 pages)

Criteria[a]	Design basis applicability	Functional means
IEEE 338 Criteria for the periodic surveillance testing of safety systems	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Standard supports selection of equipment; which resulted in the use of general design criteria (presented in Chapter 3.0) during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 497 Criteria for accident monitoring instruments	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • CAAS • RAMs • CAMs	• Standard supports selection of accident monitoring equipment (e.g., radiation monitoring, annunciation), which resulted in the use of general design criteria (presented in Chapter 3.0) during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 7-4.3.2 Criteria for digital computers in safety systems	• FPC system • FPC system display • OIT displays	• Programming software must comply with this criteria and with the NWMI Software Quality Assurance Plan (prepared during development of the Operating License Application), which will be developed per the design criteria outlined in Chapter 3.0 and this standard. • Software and hardware used for the displays for the FPC system and OIT must also follow guidelines set forth in this standard. • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 828 Configuration management in systems and software engineering	• FPC system • FPC system display • OIT displays	• Complies with IEEE 7-4.3.2 and the NWMI Software Quality Assurance Plan • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 829 Software and system test documentation	• FPC system • FPC system display • OIT displays	• Complies with IEEE 7-4.3.2 and the NWMI Software Quality Assurance Plan • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 1012 Criteria for software verification and validation	• FPC system • FPC system display • OIT displays	• Complies with IEEE 7-4.3.2 and the NWMI Software Quality Assurance Plan • Standard will be reevaluated in the Operating License Application for applicability.
IEEE 1028 Software reviews and audits	• FPC system • FPC system display • OIT displays	• Complies with IEEE 7-4.3.2 and the NWMI Software Quality Assurance Plan • Standard will be reevaluated in the Operating License Application for applicability.
ANS-10.4 Verification and validation for non-safety software	• FPC system • FPC system display • OIT displays	• Complies with IEEE 7-4.3.2 and the NWMI Software Quality Assurance Plan • Standard will be reevaluated in the Operating License Application for applicability.

Table 7-2. Instrumentation and Control Criteria Crosswalk with Design Basis Applicability and Function Means (4 pages)

Criteria[a]	Design basis applicability	Functional means
ANSI/ISA 67.04.01 Setpoints for nuclear safety-related instruments	• FPC system • FPC system IROFS end devices	• Incorporated into overall design and the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
ANSI/ISA 84.00.01, Parts 1, 2, and 3 Functional safety: safety instrumented systems for the process industry sector	• FPC system • FPC system display • OIT displays	• Standard supports the design and development of non-safety-related systems that rely on safety, reliability, and functionality and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
NUREG-0700 Human-system interface design review guidelines	• FPC system • FPC system display • OIT displays	• Standard supports the design and development of non-safety-related systems that pertain to control room arrangement, screen developments, and operator interface, and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
NUREG/CR-6463 Review guidelines on software languages for use in nuclear power plant safety systems	• FPC system	• Standard supports the design, development, and review of safety-related software and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
NUREG/CR-6090 PLC and applications in nuclear reactor systems	• FPC system	• Standard supports the design, development, and review of safety-related and non-safety-related software and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
EPRI TR-106439 Guideline on evaluation/acceptance of commercial grade digital equipment for nuclear safety applications	• FPC system display • OIT displays	• Standard supports the design, development, and review of safety-related systems that pertain to obtaining software or hardware for the FPC system, operator interface displays, and data acquisition systems, and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.

Table 7-2. Instrumentation and Control Criteria Crosswalk with Design Basis Applicability and Function Means (4 pages)

Criteria[a]	Design basis applicability	Functional means
Regulatory Guide 1.152 Criteria for use of computers in safety systems	• FPC system • FPC system display • OIT displays	• Standard supports the design and development of redundant safety PLC platforms, FPC system redundant HMI workstations, and operator interface workstations, and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
Regulatory Guide 1.53 Single failure criterion evaluation for safety systems	• FPC system • FPC system display • FPC system IROFS end devices • ESFs • ESFs manual isolation	• Standard supports the design and development of high-integrity safety PLCs, redundant channels for ESFs, redundant operator interface workstations, redundant sensors, and alternative manual means for ESF initiation, and was used during development of the Construction Permit Application. • Standard will be reevaluated in the Operating License Application for applicability.
Regulatory Guide 5.71 Cybersecurity programs for nuclear facilities	• FPC system • FPC system display • OIT display	• Criteria requires the development of a design approach and implementation for cybersecurity. • Standard will be reevaluated in the Operating License Application for applicability.

[a] Full references are provided in Section 7.7.

CAAS = criticality accident alarm system.
CAM = continuous air monitor.
DCS = digital control system.
ESF = engineered safety feature.
FPC = facility process control.
HMI = human-machine interface.
IROFS = items relied on for safety.
NWMI = Northwest Medical Isotopes, LLC.
OIT = operator interface terminal.
PLC = programmable logic controller.
RAM = radiation alarm monitor.
SIS = safety instrumented system.

7.3 PROCESS CONTROL SYSTEMS

The process control systems for the RPF will include SNM preparation and handling processes and radioisotope production processes. SNM preparation and handling processes include uranium recovery and recycle, and target fabrication. Radioisotope production processes include target receipt and disassembly, target dissolution, Mo recovery and purification, and waste handling.

The RPF process control will be administered by the FPC system and is described in Section 7.2.3. The FPC system will perform the following high-level process functions.

- **Monitor the remote valve position for routing process fluid for inter-equipment process fluid transfers** – For specific transfers identified by the operator, the FPC system will provide a permissive to allow for the active pump in that circuit to be energized once the operator has manually configured the routing.
- **Monitor and control inter-equipment process fluid transfers in the RPF** – For transport requiring a pump, the FPC system will control the ability of the pump to be energized. For specific transfers, the FPC system will provide controlled fluid flow transfers based on a closed-loop flow control. The operator will initialize the transfer of fluids.
- **Other process fluid transfers**, including:
  - Dissolved low-enriched uranium (LEU) solution to the Mo recovery and purification system
  - Uranium solution to the uranium recovery and recycle system
  - Liquid wastes to the waste handling system

The I&C system for process utilities and support systems and for the ventilation systems will be described in more detail in the Operating License Application. The process systems described below provide for reliable control of the SNM preparation and handling process and the radioisotope production processes, and include:

- Range of operation of the sensor that is sufficient to cover the expected range of variation of the monitored variable during normal and transient process operation
- Reliable information about the status and magnitude of the process variable necessary for the full operating range of the radioisotope production and SNM recovery and recycle processes
- Reliable operation in the normal range of environmental conditions anticipated within the facility
- Safe state during loss of electrical power

Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the RPF process control systems are discussed in Chapter 14.0.

7.3.1 Uranium Recovery and Recycle System

The uranium recovery and recycle system will process raffinate from the Mo recovery and purification system for recycle to the target fabrication system. Two cycles of uranium purification will be included to separate uranium from unwanted fission products using ion exchange. The first ion exchange cycle will separate the bulk of the fission product contaminant mass from the uranium product. Product will exit the ion exchange column as a dilute uranium stream that is concentrated to control the stored volume of process solutions. Uranium from the first cycle will then be purified by a nearly identical second cycle system to further reduce fission product contaminants to satisfy product criteria. Each ion exchange system feed tank will include the capability of adding a reductant and modifying the feed chemical composition such that adequate separations are achieved, while minimizing uranium losses.

Due to the variety of process activities performed during uranium recovery and recycle, the system description is divided into the following subsystems:

- Impure uranium collection
- Primary ion exchange
- Primary concentration
- Secondary ion exchange
- Secondary concentration
- Uranium recycle
- Uranium decay and accountability
- Spent ion exchange resin
- Waste collection

7.3.1.1 Design Criteria

Design criteria for the uranium recovery and recycle I&C systems are described in Section 7.2.

7.3.1.2 Design Basis and Safety Requirements

The design basis and safety requirements for the uranium recovery and recycle I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0, "Engineered Safety Features."

7.3.1.3 System Description

The uranium recovery and recycle I&C system will be defined in the Operating License Application. The strategy and associated parameters for the system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

Normal operating functions will be performed remotely using the FPC system in the control room. Table 7-3 lists the anticipated control parameters, monitoring parameters, and primary control locations for each subsystem. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. Details of the control system (e.g., interlocks and permissive signals), nuclear and process instruments, control logic and elements, indication, alarm, and control features will be developed for the Operating License Application.

Table 7-3. Uranium Recovery and Recycle Control and Monitoring Parameters (2 pages)

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
Impure uranium collection	• Flowrate (A) • Pump actuation (M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Level • Pressure • Temperature • Valve position	Control room
Primary ion exchange	• Flowrate (A) Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Analyzer, uranium • Density • Differential pressure • Flowrate • Flowrate totalizer • Level • Pressure • Temperature • Valve position	Control room
Primary concentration	• Density (A) • Flowrate (A) • Level (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Analyzer, uranium • Density • Differential pressure • Flowrate • Level • Pressure • Temperature • Valve position	Control room
Secondary ion exchange	• Flowrate (A) Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Analyzer, uranium • Density • Differential pressure • Flowrate • Flowrate totalizer • Level • Pressure • Temperature • Valve position	Control room
Secondary concentration	• Density (A) • Flowrate (A) • Level (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Analyzer, uranium • Density • Differential pressure • Flowrate • Level • Pressure • Temperature • Valve position	Control room
Uranium recycle	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Level • Pressure • Temperature • Valve position	Control room

Table 7-3. Uranium Recovery and Recycle Control and Monitoring Parameters (2 pages)

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
Uranium decay and accountability	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Level • Pressure • Temperature • Valve position	Control room
Spent ion exchange resin	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Valve actuation (A/M)	• Analyzer, uranium • Differential pressure • Flowrate • Level • Pressure • Valve position	Control room
Waste collection	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Level • Pressure • Valve position	Control room

Table 7-4 provides a preliminary listing of the interlocks and permissive signals that have been identified. These devices will be further developed and detailed information will be provided in the Operating License Application.

Table 7-4. Uranium Recycle and Recovery System Interlocks and Permissive Signals (4 pages)

Interlock or permissive input	Hard-wired or PLC	Safety Interlock
Impure uranium collection tank (UR-TK-100A) low-level switch (typical of eight tanks)	PLC	N/A
Impure uranium collection tank (UR-TK-100A) high-level switch (typical of eight tanks)	PLC	N/A
Impure uranium collection tank (UR-TK-100A) high-temperature switch (typical of eight tanks)	PLC	N/A
IX feed tank 1 (UR-TK-200) low-level switch	PLC	N/A
IX feed tank 1 (UR-TK-200) high-level switch	PLC	N/A
IX feed tank 1 (UR-TK-200) high-temperature switch	PLC	N/A
IX column 1A (UR-IX-240) high-uranium alarm (AAH-252)	PLC	N/A
IX column 1A U solution filter (UR-F-250) high-differential pressure alarm	PLC	N/A
IX column 1A waste filter (UR-F-255) high-differential pressure alarm	PLC	N/A
IX column 1B (UR-IX-260) high-uranium alarm (AAH-272)	PLC	N/A

Table 7-4. Uranium Recycle and Recovery System Interlocks and Permissive Signals (4 pages)

Interlock or permissive input	Hard-wired or PLC	Safety Interlock
IX column 1B U solution filter (UR-F-270) high-differential pressure alarm	PLC	N/A
IX Column 1B Waste Filter (UR-F-275) high-differential pressure alarm	PLC	N/A
Concentrator 1 feed tank (UR-TK-300) low-level switch	PLC	N/A
Concentrator 1 feed tank (UR-TK-300) high-level switch	PLC	N/A
Concentrator 1 (UR-Z-320) low-liquid level alarm	PLC	N/A
Concentrator 1 (UR-Z-320) high-liquid level alarm	PLC	N/A
Concentrator 1 (UR-Z-320) demister high-differential pressure alarm	PLC	N/A
Concentrator 1 (UR-Z-320) condenser high-differential pressure alarm	PLC	N/A
Concentrator 1 (UR-Z-320) condenser high-offgas temperature alarm	PLC	N/A
Condensate sample tank 1A (UR-TK-340) high-liquid level alarm	PLC	N/A
Condensate sample tank 1A (UR-TK-340) high-uranium switch (AE-356)	Hard-wired	Reroute condensate transfer to UR-TK-300 (position V-396, close V-397) Close IX column eluent addition control valves (V-244 and V-264)
Condensate delay tank 1 (UR-TK-370) high-liquid level alarm	PLC	N/A
Condensate sample tank 1B (UR-TK-340) high-liquid level alarm	PLC	N/A
Condensate sample tank 1B (UR-TK-370) high-uranium switch (AE-386)	Hard-wired	Permissive to route condensate to WH-TK-420 (position V-496, open V-397) Permissive to open IX column eluent addition control valves (V-244 and V-264)
IX feed tank 2A (UR-TK-400) low-level switch	PLC	N/A
IX feed tank 2A (UR-TK-400) high-level switch	PLC	N/A
IX feed tank 2A (UR-TK-400) high-temperature switch	PLC	N/A
IX feed tank 2B (UR-TK-420) low-level switch	PLC	N/A
IX feed tank 2B (UR-TK-420) high-level switch	PLC	N/A
IX feed tank 2B (UR-TK-420) high-temperature switch	PLC	N/A
IX column 2A (UR-IX-460) high-uranium alarm (AAH-472)	PLC	N/A
IX column 2A U solution filter (UR-F-470) high-differential pressure alarm	PLC	N/A

Table 7-4. Uranium Recycle and Recovery System Interlocks and Permissive Signals (4 pages)

Interlock or permissive input	Hard-wired or PLC	Safety Interlock
IX column 2A waste filter (UR-F-475) high-differential pressure alarm	PLC	N/A
IX column 2B (UR-IX-480) high-uranium alarm (AAH-492)	PLC	N/A
IX column 2B U solution filter (UR-F-490) high-differential pressure alarm	PLC	N/A
IX column 2B waste filter (UR-F-495) high-differential pressure alarm	PLC	N/A
Concentrator 2 feed tank (UR-TK-500) low-level switch	PLC	N/A
Concentrator 2 feed tank (UR-TK-500) high-level switch	PLC	N/A
Concentrator 2 (UR-Z-520) low-liquid level alarm	PLC	N/A
Concentrator 2 (UR-Z-520) high-liquid level alarm	PLC	N/A
Concentrator 2 (UR-Z-520) demister high-differential pressure alarm	PLC	N/A
Concentrator 2 (UR-Z-520) condenser high-differential pressure alarm	PLC	N/A
Concentrator 2 (UR-Z-520) condenser high-offgas temperature alarm	PLC	N/A
Condensate sample tank 2A (UR-TK-540) high-liquid level alarm	PLC	N/A
Condensate sample tank 2A (UR-TK-540) high-uranium switch (AE-556)	Hard-wired	Reroute condensate transfer to UR-TK-500 (position V-596, close V-597) Close IX column eluent addition control valves (V-464 and V-484)
Condensate delay tank 2 (UR-TK-560) high-liquid level alarm	PLC	N/A
Condensate sample tank 2B (UR-TK-570) high-liquid level alarm	PLC	N/A
Condensate sample tank 2B (UR-TK-570) high-uranium switch (AE-586)	Hard-wired	Permissive to route condensate to WH-TK-420 (position V-596, open V-597) Permissive to open IX column eluent addition control valves (V-464 and V-484)
Concentrate receiver tank (UR-TK-600) high-liquid level alarm	PLC	N/A
Concentrate receiver tank (UR-TK-600) high-temperature alarm	PLC	N/A
Product sample tank (UR-TK-620) high-liquid level alarm	PLC	N/A
Product sample tank (UR-TK-620) high-temperature alarm	PLC	N/A
Uranium rework tank (UR-TK-660) high-liquid level alarm	PLC	N/A
Uranium rework tank (UR-TK-660) high-temperature alarm	PLC	N/A

Table 7-4. Uranium Recycle and Recovery System Interlocks and Permissive Signals (4 pages)

Interlock or permissive input	Hard-wired or PLC	Safety Interlock
Uranium decay tank (UR-TK-700A) high-liquid level alarm (typical of 17 tanks)	PLC	N/A
Uranium decay tank (UR-TK-700A) high-temperature alarm (typical of 17 tanks)	PLC	N/A
Uranium accountability tank (UR-TK-720) high-liquid level alarm	PLC	N/A
Uranium accountability tank (UR-TK-720) high-temperature alarm	PLC	N/A
Spent resin tank A (UR-TK-820A) high-liquid level alarm	PLC	N/A
Spent resin tank A (UR-TK-820A) high-temperature alarm	PLC	N/A
Spent resin tank B (UR-TK-820B) high-liquid level alarm	PLC	N/A
Spent resin tank B (UR-TK-820B) high-temperature alarm	PLC	N/A
Resin transfer liquid tank (UR-TK-850) high-liquid level alarm	PLC	N/A
IX waste collection 1 tank (UR-TK-900) high-liquid level alarm	PLC	N/A
IX waste collection 1 tank (UR-TK-900) high-temperature alarm	PLC	N/A
IX waste collection 2 tank (UR-TK-920) high-liquid level alarm	PLC	N/A
IX waste collection 2 tank (UR-TK-920) high-temperature alarm	PLC	N/A

IX = ion exchange.
PLC = programmable logic controller.
TBD = to be determined.

7.3.1.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.2 Target Fabrication System

The target fabrication system will produce LEU targets from fresh LEU material and recycled uranyl nitrate. The system will commence with the receipt of fresh LEU from the U.S. Department of Energy, and end with packaging new targets for shipment to the university research reactor facilities.

Due to the variety of process activities performed during target fabrication, the system description is divided into the following subsystems.

- Fresh uranium receipt and dissolution
- Nitrate extraction
- Acid-deficient uranyl nitrate (ADUN) concentration
- [Proprietary Information]
- [Proprietary Information]
- [Proprietary Information]

- Target fabrication waste
- Target assembly
- [Proprietary Information]
- New target handling

7.3.2.1 Design Criteria

Design criteria for the target fabrication I&C systems are described in Section 7.2.

7.3.2.2 Design Basis and Safety Requirements

The design basis and safety requirements for the target fabrication I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.3.2.3 System Description

The target fabrication I&C system will be defined in the Operating License Application. The strategy and associated parameters for the I&C system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

Normal operating functions will be performed remotely using the FPC system HMI in the target fabrication area. Table 7-5 lists the anticipated control parameters, monitoring parameters, and primary control location for each subsystem. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. Details of the control system (e.g., interlocks and permissive signals), nuclear and process instruments, control logic and elements, indication, alarm, and control features will be developed for the Operating License Application.

Table 7-5. Target Fabrication System Control and Monitoring Parameters (2 pages)

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
Fresh uranium receipt and dissolution (100-series tag numbers)	• Current (A) • Conductivity (A) • Flow totalizer (A) • Heater actuation (A/M) • Level (A) • Pump actuation (A/M) • Temperature (A) • Valve actuation (A/M)	• Conductivity • Density • Differential pressure • Flowrate • Level • Pressure • Temperature	Local
Nitrate extraction (200-series tag numbers)	• Analyzer, pH (A) • Contactor actuation (M) • Flow totalizer (A) • Flowrate (A) • Level (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Analyzer, pH • Density • Differential pressure Flowrate • Level • Pressure • Pump motor speed • Temperature	Local

Table 7-5. Target Fabrication System Control and Monitoring Parameters (2 pages)

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
ADUN concentration (300-series tag numbers)	• Conductivity (A) • Density (A) • Flowrate (A) • Level (A) • Pump actuation (A/M) • Pump motor speed (A) • Valve actuation (A/M)	• Conductivity • Density • Flowrate • Level • Pressure • Temperature	Local
[Proprietary Information] (400-series tag numbers)	• Level (A) • Pump actuation (A/M) • Tank agitator actuation (A/M) • Tank agitator speed (A) • Temperature (A) • Valve actuation (A/M)	• Flowrate • Level • Pressure • Temperature	Local
[Proprietary Information] (500-series tag numbers)	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M) • Vibration dispersion assembly actuation (M)	• Density • Differential pressure • Pressure • Level • Temperature • Vibration	Local
[Proprietary Information] (600-series tag numbers)	• Analyzer, hydrogen (A) • Analyzer, oxygen (A) • Flow totalizer (A) • Level (A) • Tank agitator speed (M) • Temperature (A) • Valve actuation (A/M)	• Analyzer, hydrogen • Analyzer, oxygen • Flowrate • Level • Pressure • Temperature	Local
Target fabrication waste (700-series tag numbers)	• Flowrate (A) • Level (A) • Pump actuation (A/M) • Pump motor speed (A) • Valve actuation (A/M)	• Density • Flowrate • Level • Pressure • Temperature	Local
Target assembly	TBD	TBD	Local
[Proprietary Information]	TBD	TBD	Local
New target handling	TBD	TBD	Local

ADUN = acid-deficient uranyl nitrate.
LEU = low-enriched uranium.
TBD = to be determined.

Table 7-6 provides a listing of the target fabrication I&C system interlocks and permissive signals that have been identified. These devices will be further developed and detailed information will be provided in the Operating License Application.

Table 7-6. Target Fabrication System Interlocks and Permissive Signals (2 pages)

Interlock or permissive input	Hard-wired or PLC	Safety interlock
Dissolver column (TF-D-100) high-temperature switch	PLC	N/A
Uranium dissolution heat exchanger (TF-E-120) chilled water return high-conductivity switch	Hard-wired	Close chilled water return control valve (XV-122) on high conductivity
Uranium dissolution heat exchanger (TF-E-120) low-differential pressure alarm	PLC	N/A
Uranyl nitrate storage tank (TF-TK-200) level switch	PLC	N/A
ADUN evaporator condenser (TF-E-350) chilled water return high-conductivity switch	Hard-wired	Close chilled water return control valve (HV-352) on high conductivity
ADUN product heat exchanger (TF-E-360) low-differential pressure alarm	PLC	N/A
ADUN product heat exchanger (TF-E-360) chilled water return high-conductivity switch	Hard-wired	Close chilled water return control valve (HV-361) on high conductivity
ADUN evaporator reboiler (TF-E-330) steam condensate high-conductivity switch	Hard-wired	Close steam condensate control valve (XV-333) on high conductivity
ADUN storage tank (TF-TK-400) low-level switch	PLC	N/A
ADUN storage tank (TF-TK-405) low-level switch	PLC	N/A
ADUN storage tank (TF-TK-410) low-level switch	PLC	N/A
ADUN storage tank (TF-TK-415) low-level switch	PLC	N/A
ADUN storage tank (TF-TK-400) high-level switch	PLC	N/A
ADUN storage tank (TF-TK-405) high-level switch	PLC	N/A
ADUN storage tank (TF-TK-401) high-level switch	PLC	N/A
ADUN storage tank (TF-TK-415) high-level switch	PLC	N/A
[Proprietary Information] (TF-TK-480) high-level switch	PLC	N/A
[Proprietary Information] (TF-C-500) high-temperature switch	PLC	N/A
Silicone oil heater (TF-E-550) outlet high-temperature switch	Hard-wired	N/A
[Proprietary Information] furnace #1 (TF-Z-660) high-temperature switch	Hard-wired	N/A
[Proprietary Information] furnace #2 (TF-Z-661) high-temperature switch	Hard-wired	N/A
[Proprietary Information] furnace #3 (TF-Z-662) high-temperature switch	Hard-wired	N/A
[Proprietary Information] furnace #4 (TF-Z-663) high-temperature switch	Hard-wired	N/A

Table 7-6. Target Fabrication System Interlocks and Permissive Signals (2 pages)

Interlock or permissive input	Hard-wired or PLC	Safety interlock
[Proprietary Information] furnace #1 (TF-Z-660) door closed switch	PLC	N/A
[Proprietary Information] furnace #2 (TF-Z-661) door closed switch	PLC	N/A
[Proprietary Information]furnace #3 (TF-Z-662) door closed switch	PLC	N/A
[Proprietary Information]furnace #4 (TF-Z-663) door closed switch	PLC	N/A
Reduction furnace offgas heat exchanger (TF-E-670) outlet high-oxygen concentration	PLC	N/A
Reduction furnace offgas heat exchanger (TF-E-670) outlet high-hydrogen concentration	PLC	N/A
Aqueous waste pencil tank (TF-TK-700) high-level alarm	PLC	N/A
Aqueous waste pencil tank (TF-TK-705) high-level alarm	PLC	N/A
TCE tank (TF-TK-760) high-level switch	PLC	N/A
Target fabrication overflow tank (TF-TK-770) high-high-level switch	PLC	N/A

ADUN = acid-deficient uranyl nitrate.
PLC = programmable logic controller.
TBD = to be determined.
TCE = trichloroethylene.

7.3.2.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.3 Target Receipt and Disassembly System

The target receipt and disassembly system will include the delivery and receipt of the irradiated target cask, introduction of the irradiated targets into the hot cell, disassembly of the targets, and retrieval and transfer of the irradiated target material for processing. This system will feed the target dissolution system by the transfer of recovered irradiated target material through the dissolver 1 hot cell (DS-EN-100) and dissolver 2 hot cell (DS-EN-200) isolation door interfaces.

Due to the variety of activities performed during target receipt and disassembly, the system description is divided into the following subsystems:

- Cask receipt
- Target receipt
- Target disassembly

7.3.3.1 Design Criteria

Design criteria for the target receipt and disassembly I&C systems are described in Section 7.2.

7.3.3.2 Design Basis and Safety Requirements

The design basis and safety requirements for the target receipt and disassembly I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.3.3.3 System Description

The target receipt and disassembly I&C system will be defined in the Operating License Application. The strategy and associated parameters for the I&C system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

Normal operating functions will be performed remotely using the FPC system HMI in the truck bay, cask preparation airlock, and the operating gallery. Redundant control functions will be provided in the control room. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. Details of the control system (e.g., interlocks and permissive signals), nuclear and process instruments, control logic and elements, indication, alarm, and control features will be developed for the Operating License Application.

Prior to the start of disassembly operations, the following process control permissive signals will be required.

- Ventilation inside the hot cell is operable.
- Fission gas capture hood is on and functional.
- Irradiated target material collection container is in position under the target cutting assembly collection bin.
- Waste drum transfer port is open and there is physical space to receive the waste target hardware after disassembly and irradiated target material recovery.

The control parameters and monitoring parameters will be defined during design development for the Operating License Application.

7.3.3.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.4 Target Dissolution System

The target dissolution system process will receive the LEU target material from the target receipt and disassembly system and dissolve the uranium and molybdenum-99 (^{99}Mo) in the solid irradiated target material in hot nitric acid. The concentrated uranyl nitrate solution will then be transferred to the Mo recovery and purification system for further processing.

The target dissolution process will be operated in a batch f[Proprietary Information] transferred to a collection container. The collection container will move through the pass-through to a dissolver basket positioned over a dissolver, the target material will then be dissolved and the resulting solution transferred to the Mo recovery and purification system.

Target dissolution of irradiated LEU will result in gaseous fission products (iodine [I], krypton [Kr], and xenon [Xe]) with very high radiation fields. A primary function of the process offgas systems will be to control release of these gases both internal and external to the facility. The dissolver offgas treatment system will include the nitrogen oxide (NO_x) treatment and fission gas treatment subsystems.

Due to the variety of process activities performed during target dissolution, the system description is divided into the following subsystems:

- Target dissolution 1 and target dissolution 2
- NO_x treatment 1 or NO_x treatment 2
- Pressure relief
- Primary fission gas treatment
- Secondary fission gas treatment
- Waste collection

7.3.4.1 Design Criteria

Design criteria for the target dissolution I&C systems are described in Section 7.2.

7.3.4.2 Design Basis and Safety Requirements

The design basis and safety requirements for the target dissolution I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.3.4.3 System Description

The target dissolution I&C system will be defined in the Operating License Application. The strategy and associated parameters for the I&C system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

Loading of [Proprietary Information] into the dissolver will involve mechanical handling of the transfer containers. Operators using remote in-cell cranes and manipulators will perform these functions. Other normal operating functions will be performed remotely using the FPC system HMI in the operating gallery. Redundant control functions will be provided in the control room. Table 7-7 lists the anticipated control parameters, monitoring parameters, and primary control locations for each subsystem. Details of the control system (e.g., interlocks and permissive signals), control logic, indication, alarm, and control features will be defined in the Operating License Application.

Table 7-7. Target Dissolution System Control and Monitoring Parameters

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
Target dissolution 1 and 2	• Dissolver agitator actuation (A/M) • Dissolver agitator speed (A) • Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Dissolver agitator speed • Flowrate • Flowrate totalizer • Level • Pressure • Radiation • Temperature • Valve position	Operating gallery
NO_x treatment 1 or 2	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Differential pressure • Flowrate • Flowrate totalizer • Level • Pressure • Radiation • Temperature • Valve position	Operating gallery
Pressure relief	• Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Flowrate • Level • Pressure • Valve position	Operating gallery
Primary fission gas treatment	• Temperature (A) • Valve actuation (A/M)	• Differential pressure • Flowrate • Pressure • Radiation • Temperature • Valve position	Operating gallery
Secondary fission gas treatment	• Valve actuation (A/M)	• Differential pressure • Flowrate • Pressure • Radiation • Temperature • Valve position	Operating gallery
Waste collection	• Pump actuation (A/M) • Pump motor Speed (A) • Temperature (A) • Valve actuation (A/M)	• Differential pressure • Flowrate • Level • Temperature • Pressure • Radiation • Valve position	Operating gallery

NO_x = nitrogen oxide.

Table 7-8 provides a preliminary listing of the target dissolution I&C system interlocks and permissive signals that have been identified. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. These devices will be further developed and detailed information will be provided in the Operating License Application.

Table 7-8. Target Dissolution System Interlocks and Permissive Signals (2 pages)

Interlock or permissive input	Hard-wired or PLC	Safety interlock
Dissolver 1 (DS-D-100) high-liquid level alarm	PLC	N/A
Dissolver 1 (DS-D-100) low-liquid level alarm	PLC	N/A
Dissolver 1 (DS-D-100) high liquid temperature alarm	PLC	N/A
Dissolver 1 Condenser (DS-E-130) high gas temperature alarm	PLC	N/A
Dissolver 2 (DS-D-200) high-liquid level alarm	PLC	N/A
Dissolver 2 (DS-D-200) low-liquid level alarm	PLC	N/A
Dissolver 2 (DS-D-200) high liquid temperature alarm	PLC	N/A
Dissolver 2 condenser (DS-E-230) high gas temperature alarm	PLC	N/A
Primary caustic scrubber 1 (DS-C-310) high-liquid level alarm	PLC	N/A
Caustic scrubber 1 (DS-C-310) high gas temperature	PLC	N/A
NO_x oxidizer 1 (DS-C-340) high-liquid level alarm	PLC	N/A
NOx oxidizer 1 (DS-C-340) high gas temperature	PLC	N/A
NO_x absorber 1 (DS-C-370) high-liquid level alarm	PLC	N/A
NOx absorber 1 (DS-C-370) high gas temperature	PLC	N/A
Primary caustic scrubber 2 (DS-C-410) high-liquid level alarm	PLC	N/A
Caustic scrubber 2 (DS-C-410) high gas temperature	PLC	N/A
NO_x oxidizer 2 (DS-C-440) high-liquid level alarm	PLC	N/A
NOx oxidizer 2 (DS-C-440) high gas temperature	PLC	N/A
NO_x absorber 2 (DS-C-470) high-liquid level alarm	PLC	N/A
NOx absorber 2 (DS-C-470) high gas temperature	PLC	N/A
Pressure relief tank (DS-TK-500) high-pressure alarm	Hard-wired	Opens valve to capture dissolver gases
Pressure relief tank (DS-TK-500) high-liquid level alarm	PLC	N/A
Pressure relief tank (DS-TK-500) low-liquid level alarm	PLC	N/A
Dryer A (DS-E-610A) high gas temperature alarm	PLC	N/A
Primary adsorber A (DS-SB-620A) high gas temperature alarm	PLC	N/A
Filter A (DS-F-630A) high-pressure differential alarm	PLC	N/A
Dryer B (DS-E-610B) high gas temperature alarm	PLC	N/A
Primary adsorber B (DS-SB-620B) high gas temperature alarm	PLC	N/A
Filter B (DS-F-630B) high-pressure differential alarm	PLC	N/A
Dryer C (DS-E-610C) high gas temperature alarm	PLC	N/A

Table 7-8. Target Dissolution System Interlocks and Permissive Signals (2 pages)

Interlock or permissive input	Hard-wired or PLC	Safety interlock
Primary adsorber C (DS-SB-620C) high gas temperature alarm	PLC	N/A
Filter C (DS-F-630C) high-pressure differential alarm	PLC	N/A
Secondary adsorber A (DS-SB-730A) high gas temperature alarm	PLC	N/A
Secondary adsorber B (DS-SB-730B) high gas temperature alarm	PLC	N/A
Secondary adsorber C (DS-SB-730C) high gas temperature alarm	PLC	N/A
Waste collection and sampling tank 1 (DS-TK-800) high-liquid level alarm	PLC	N/A
Waste collection and sampling tank 1 (DS-TK-800) high-liquid temperature alarm	PLC	N/A
Waste collection and sampling tank 2 (DS-TK-820) high-liquid level alarm	PLC	N/A
Waste collection and sampling tank 2 (DS-TK-820) high-liquids temperature alarm	PLC	N/A

N/A = not applicable.
NO_x = nitrogen oxide.
PLC = programmable logic controller.

7.3.4.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.5 Molybdenum Recovery and Purification System

The Mo recovery and purification system will receive the impure Mo/uranium solution from the target dissolution system into feed tank 1A and feed tank 1B (MR-TK-100 and MR-TK-140) located in the tank hot cell. The Mo/uranium solution will then be transferred to process hot cells and processed through three separate ion exchange unit operations to achieve the desired product criteria. A collection container holding the separated and purified Mo product material will be used for final chemical adjustment and sampling for verification of batch acceptance. The product will be sampled and weighed, placed in stainless steel bottles with lids applied and tightened, loaded into shielded containers, and then shipped in an approved cask.

Due to the variety of activities performed during Mo recovery and purification, the system description is divided into the following subsystems:

- Primary ion exchange
- Secondary ion exchange
- Tertiary ion exchange
- Mo product

7.3.5.1 Design Criteria

Design criteria for the Mo recovery and purification I&C systems are described in Section 7.2.

7.3.5.2 Design Basis and Safety Requirements

The design basis and safety requirements for the Mo recovery and purification I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.3.5.3 System Description

The Mo recovery and purification I&C system will be defined in the Operating License Application. The strategy and associated parameters for the I&C system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

Operators using remote in-cell manipulators will perform the product transfer and packaging functions. All other normal operating functions will be performed remotely using the FPC system HMI in the operating gallery. Redundant control functions will be provided in the control room. Table 7-9 lists the anticipated control parameters, monitoring parameters, and primary control locations for each subsystem. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. Details of the control system (e.g., interlocks and permissive signals), nuclear and process instruments, control logic and elements, indication, alarm, and control features will be developed for the Operating License Application.

Table 7-9. Molybdenum Recovery and Purification System Control and Monitoring Parameters

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
Primary ion exchange	• Temperature (A) • Valve actuation (A/M)	• Density • Flowrate • Level • Temperature • Pressure • Radiation • Valve position	Operating gallery
Secondary ion exchange	• Pumps (M)	• Temperature	Operating gallery
Tertiary ion exchange	• Pumps (M)	• Density • Flowrate • Level • Pressure • Temperature	Operating gallery
Molybdenum product	• Actuate capping unit (M)	• Weight	Operating gallery

Table 7-10 provides a preliminary listing of the Mo recovery and purification system interlocks and permissive signals that have been identified. These devices will be further developed and detailed information will be provided in the Operating License Application.

Table 7-10. Molybdenum Recovery and Purification System Interlocks and Permissive Signals

Interlock or permissive input	Hard-wired or PLC	Safety Interlock
Feed tank 1A (MR-TK-100) high-liquid level alarm	PLC	N/A
Feed tank 1A (MR-TK-100) low-liquid level alarm	PLC	N/A
Feed tank 1A (MR-TK-100) high-temperature alarm	PLC	N/A
Feed tank 1A (MR-TK-100) high-pressure alarm	PLC	N/A
Feed tank 1B (MR-TK-140) high-liquid level alarm	PLC	N/A
Feed tank 1B (MR-TK-140) low-liquid level alarm	PLC	N/A
Feed tank 1B (MR-TK-140) high-temperature alarm	PLC	N/A
Feed tank 1B (MR-TK-140) high-pressure alarm	PLC	N/A
U solution collection tank (MR-TK-180) high-liquid level alarm	PLC	N/A
U solution collection tank (MR-TK-180) low-liquid level alarm	PLC	N/A
U solution collection tank (MR-TK-180) high-pressure alarm	PLC	N/A
Waste collection tank (MR-TK-340) high-liquid level alarm	PLC	N/A
Waste collection tank (MR-TK-340) low-liquid level alarm	PLC	N/A
Waste collection tank (MR-TK-340) high-pressure alarm	PLC	N/A

N/A = not applicable.
PLC = programmable logic controller.
U = uranium.

7.3.5.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.6 Waste Handling System

The waste handling system will consist of storage tanks for accumulating waste liquids and adjusting the waste composition, and the equipment needed for handling and encapsulating solid waste. Liquid waste will be split into high-dose and low-dose streams by concentration. The high-dose fraction will be further concentrated and adjusted. Liquid waste will then be mixed with an adsorbent material. The solid waste streams will be placed in a waste drum and encapsulated by adding a cement material to fill voids remaining within the drum. All high-dose waste streams will be held for decay and shipped to a disposal facility.

Due to the variety of activities performed during waste handling, the system description is divided into the following subsystems:

- High-dose liquid waste collection
- Low-dose liquid waste collection
- Low-dose waste evaporation
- High-dose liquid waste solidification
- Low-dose liquid waste solidification

- Spent resin dewatering
- Solid waste encapsulation
- High-dose waste decay
- High-dose waste handling

7.3.6.1 Design Criteria

Design criteria for the waste handling I&C systems are described in Section 7.2.

7.3.6.2 Design Basis and Safety Requirements

The design basis and safety requirements for the waste handling I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.3.6.3 System Description

The waste handling I&C system will be defined in the Operating License Application. The strategy and associated parameters for the I&C system are provided below. Preliminary process sequences are provided in Chapter 4.0 to communicate the control strategy for normal operations, which sets the requirements for the process monitoring and control equipment, and the associated instrumentation.

All normal operating functions for low-dose liquid solidification will be controlled locally using FPC system HMIs in the low-dose waste room (Room W107). A local control room will be provided in this room for most waste handling operations. All normal operating functions for the high-dose liquid waste solidification, high-dose waste decay, spent resin dewatering, and solid waste handling hot cell operations will be controlled from the waste handling control room. Liquid waste collection and low-dose liquid waste evaporation operations will be controlled from the control room. Table 7-11 lists the anticipated control parameters, monitoring parameters, and primary control locations for each subsystem. In addition, the implementation of IROFS CS-14, CS-15, CS-20, CS-27, and RS-10 interlocks for this system are under development. Details of the control system (e.g., interlocks and permissive signals), nuclear and process instruments, control logic and elements, indication, alarm, and control features will be developed for the Operating License Application.

Table 7-11. Waste Handling System Control and Monitoring Parameters

Subsystem name	Control parameters (automatic/manual)	Monitoring parameters	Primary control location
High-dose liquid waste collection	• Valve position	• Density • Differential pressure • Flowrate • Flowrate totalizer • Level • Temperature • Pressure • Radiation • Valve position	Control room
High-dose liquid waste solidification	• Valve position	• Density • Differential Pressure • Flowrate • Flowrate totalizer • Level • Temperature • Pressure • Radiation • Valve Position	Low dose solidification room
Low-dose liquid waste collection	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Flowrate totalizer • Level • Temperature • Pressure • Valve position	Control room
Low-dose liquid waste evaporation	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Differential pressure • Flowrate • Level • Temperature • Pressure • Valve position	Control room
Low-dose liquid waste solidification	• Flowrate (A) • Pump actuation (A/M) • Pump motor speed (A) • Temperature (A) • Valve actuation (A/M)	• Density • Differential pressure • Flowrate • Flowrate totalizer • Level • Temperature • Pressure • Valve position	Low dose solidification room
Spent resin dewatering	• Valve actuation (A/M)	• Valve position	Low dose solidification room
Solid waste encapsulation	• Actuate grout mixer (M)	• Pressure	Low dose solidification room
High-dose waste decay	TBD	TBD	Low dose solidification room
High-dose waste handling	TBD	TBD	Low dose solidification room

TBD = to be determined.

Table 7-12 provides a preliminary listing of the waste handling system interlocks and permissive signals that have been identified. These devices will be further developed and detailed information will be provided in the Operating License Application.

Table 7-12. Waste Handling System Interlocks and Permissive Signals

Interlock or permissive input	Hard-wired or PLC	Safety interlock
High-dose waste collection tank (WH-TK-100) high-liquid level alarm	PLC	N/A
High-dose waste collection tank (WH-TK-100) low-liquid level alarm	PLC	N/A
High-dose waste collection tank (WH-TK-100) low-pressure alarm	PLC	N/A
High-dose waste concentrator (WH-Z-200) high-liquid level alarm	PLC	N/A
High-dose waste concentrator (WH-Z-200) low-liquid level alarm	PLC	N/A
High-dose waste concentrator (WH-Z-200) demister high-differential pressure alarm	PLC	N/A
High-dose waste concentrator (WH-Z-200) condenser high-differential pressure alarm	PLC	N/A
High-dose waste concentrator (WH-Z-200) condenser offgas high-temperature alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-240) high-liquid level alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-240) low-liquid level alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-240) low-pressure alarm	PLC	N/A
High-dose waste container offgas filter (WH-F-330) high-pressure differential alarm	PLC	N/A
Condensate collection tank (WH-TK-400) high-liquid level alarm	PLC	N/A
Condensate collection tank (WH-TK-400) low-liquid level alarm	PLC	N/A
Condensate collection tank (WH-TK-400) low-pressure alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-420) high-liquid level alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-420) low-liquid level alarm	PLC	N/A
Low-dose waste collection tank (WH-TK-420) low-pressure alarm	PLC	N/A
Low-dose waste evaporation tank 1 (WH-TK-500) high-liquid level alarm	PLC	N/A
Low-dose waste evaporation tank 1 (WH-TK-500) low-liquid level alarm	PLC	N/A
Low-dose waste evaporation tank 1 (WH-TK-500) low-pressure alarm	PLC	N/A
Low-dose waste evaporation tank 2 (WH-TK-530) high-liquid level alarm	PLC	N/A
Low-dose waste evaporation tank 2 (WH-TK-530) low-liquid level alarm	PLC	N/A
Low-dose waste evaporation tank 2 (WH-TK-530) low-pressure alarm	PLC	N/A
Low-dose waste container offgas filter (WH-F-630) high-pressure differential alarm	PLC	N/A

PLC = programmable logic controller. TBD = to be determined.

7.3.6.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application.

7.3.7 Criticality Accident Alarm System

The RPF will use a CAAS to monitor for a criticality and provide emergency notifications for evacuation.

7.3.7.1 Design Criteria

Design criteria for the CAAS I&C systems are described in Section 7.2.

7.3.7.2 Design Basis and Safety Requirements

The design basis and safety requirements for the CAAS I&C systems are described in Section 7.2.

7.3.7.3 System Description

The CAAS will be provided as a vendor package with an integrated control system. The CAAS control HMI will be located in the control room and will provide local alarms at the detector locations and at the CAAS HMI. The FPC system will provide alarm and status monitoring in the control room. The facility-wide notification system configuration will be provided in the Operating License Application. The surveillance requirements for the CAAS system are described in Chapter 6.0.

7.3.7.4 System Performance Analysis and Conclusion

The system performance analysis for each process system will be provided in the Operating License Application. The overall I&C system performance analysis is discussed in Section 7.2.

The CAAS will provide for continuous monitoring, indication, and recording of neutron or gamma radiation levels in areas where personnel may be present and wherever an accidental criticality event could result from operational processes. The CAAS will be capable of detecting a criticality accident that produces an absorbed dose in soft tissue of 20 radiation absorbed dose (rad) of combined neutron or gamma radiation at an unshielded distance of 2 meters (m) from the reacting material within 1 minute (min), except for events occurring in areas not normally accessed by personnel and where shielding provides protection against radiation generated from an accidental criticality. Two detectors will cover each area needing CAAS coverage.

The control unit electronics will actuate local and remote alarms. The locations of the detectors will be provided in the Operating License Application.

The CAAS detectors will provide local annunciation and remote annunciation in the control room to alarm when the radiation levels exceed established setpoints. Alarming CAAS monitors will communicate the location of the criticality accident alarm to the FPC system. Diagrams of the CAAS and associated systems will be provided in the Operating License Application.

The uninterruptible power supply (UPS) will provide emergency power to the CAAS during a loss of off-site power. The CAAS will meet the criteria of 10 CFR 20.1501, "General," and use the guidance provided by ANSI/ANS 8.3, *Criticality Accident Alarm System*, and Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*. As a safety-related system, the CAAS will be designed to remain operational during design basis accidents, which are described in Chapter 13.0.

7.4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS

7.4.1 System Description

The ESFs are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to workers, the public, and environment within acceptable values. Chapter 6.0 provides a description of the ESFs, including the accidents mitigated and SSCs used to provide the ESFs.

The ESF systems will operate independently from the FPC systems as hard-wired controls. However, the ESFs will integrate into the FPC systems and provide a common point of HMI, monitoring, and alarming at the control room and local HMI workstations.

Table 7-13 lists the ESFs that will require actuation by the I&C system. Monitoring systems that are credited in the safety analysis are also included in the table.

Table 7-13. Engineered Safety Feature Actuation or Monitoring Systems (2 pages)

Engineered safety feature	IROFS	Accident(s) mitigated	I&C SSCs providing engineered safety feature
Primary offgas relief system	RS-09	Dissolver offgas failure during dissolution operation	Pressure relief device, pressure relief tank
Active radiation monitoring and isolation of low-dose waste transfer	RS-10	Transfer of high-dose process liquid outside the hot cell shielding boundary	Radiation monitoring and isolation system for low-dose liquid transfers
Cask local ventilation during closure lid removal and docking preparations	RS-13	Target cladding leakage during shipment	Local capture ventilation system over closure lid during lid removal
Cask docking port enabler	RS-15	Cask not engaged in the cask docking port prior to opening the docking port door	Sensor system controlling cask docking port door operation
Process vessel emergency purge system	FS-03	Hydrogen deflagration or detonation	Backup bottled nitrogen gas supply
Active discharge monitoring and isolation	CS-14	Accidental criticality	To be provided in the Operating License Application
Independent active discharge monitoring and isolation	CS-15	Accidental criticality	To be provided in the Operating License Application
Evaporator or concentrator condensate monitoring	CS-20	Prevent nuclear criticality from high-volume transfer to non-geometrically favorable vessels in solutions with normally low fissile component concentrations	Conductivity analyzer and control valve

Table 7-13. Engineered Safety Feature Actuation or Monitoring Systems (2 pages)

Engineered safety feature	IROFS	Accident(s) mitigated	I&C SSCs providing engineered safety feature
Closed heating or cooling loop with monitoring and alarm	CS-27	Accidental criticality	Closed-loop, high-volume heat transfer fluid systems to prevent nuclear criticality or transfer of high-dose material across shielding boundary in the event of a leak into the heat transfer fluid with normally low fissile component concentrations
Dissolver offgas vacuum receiver or vacuum pump	TBD	Potential limiting control for operations; motive force for dissolver offgas	Dissolver offgas vacuum receiver tanks, dissolver offgas vacuum pumps

I&C = instrumentation and control.
IROFS = items relied on for safety.
SSC = structures, systems, and components.
TBD = to be determined.

7.4.2 Annunciation and Display

The actuation of an ESF will be displayed on the FPC system HMI and locally at the affected system with an audible alarm. The alarm annunciator display panel and the alarm or event display will show the triggering event. Once actuated, the ESFs will require manual input from the operator to reset the ESF. Clearing the triggering event will be required.

7.4.3 System Performance Analysis

Section 7.2.4 provides additional details on the analysis of system performance. Potential variables, conditions, or other items that will be probable subjects of technical specifications associated with the FPC system are provided in Chapter 14.0.

7.5 CONTROL CONSOLE AND DISPLAY INSTRUMENTS

7.5.1 Design Criteria

Design criteria for the control room I&C systems are described in Section 7.2.

7.5.2 Design Basis and Safety Requirements

The design basis and safety requirements for the control room I&C systems are described in Section 7.2.

7.5.3 System Description

The control room will provide the majority of interfaces for the facility and process control systems, with overall process controls, monitoring, alarms, and acknowledgement. The control room will consist of a properly sized and shaped control console with two or three operator interface stations or HMIs (one being a dedicated engineering interface), a master PLC or distributed controller, and all related and necessary cabinetry and subcomponents (e.g., input/output boards, gateways, Ethernet switches, power supplies, and UPS). This control system will be supported by a data highway of sensing instrument signals in the facility process areas that will be gathered onto the highway throughout the facility by an Ethernet communication-based interface backbone and brought into the control room and onto the console displays.

Dedicated controllers and human-machine monitoring interfaces or stations for other equipment systems will also be in the control room. This equipment includes the facility crane, closed-circuit television system, CAAS, and radiation monitoring system. A control panel for all facility on-site and off-site (if required) communications (e.g., telephone, intercom) will likely also be located there. The control room door into the facility will be equipped with controlled access.

The BMS will be primarily controlled and monitored from the control room. Utility systems with vendor packages and integrated controls will provide surveillance monitoring to the control room.

The FPC system will operate with a synchronized hot standby redundant system structure for all hot cell processes. Each hot cell process will be an independent subsystem having a local HMI with monitoring and control functions from the control room. Workstations for each system within the control room will be hot standby redundant. The redundant stations will run software on identical PLC systems. The PLC systems will monitor each other. On loss of synchronizing signal from one system, the other system will continue with control and monitoring.

Process systems that will be primarily controlled in the control room include uranium recovery and recycle, target dissolution, and liquid waste handling. The target receipt system will be controlled with local HMIs in the irradiated target basket receipt bay or target cask preparation airlock. Mo production process hot cell systems, including target disassembly and Mo recovery and purification, will be controlled with local HMIs in the hot cell operating gallery. The hot cell processes will have monitoring and redundant control functions from the control room.

The FPC subsystem for target fabrication processes will be controlled with local HMIs in the target fabrication area, with surveillance monitoring in the control room.

Local HMIs will be provided in Room W107, which houses equipment for low-dose waste solidification. Low-dose liquid waste will be piped in from the holding tanks in the utility area above Room W107, and drums of solidified waste will be transported out by pallet jack. This local HMI will be the primary control location for the high-dose liquid waste solidification, high-dose waste decay, spent resin dewatering, and solid waste handling hot cell operations.

7.5.4 System Performance Analysis and Conclusion

The system performance analysis for each process system will be provided in the Operating License Application. The overall I&C system performance analysis and conclusion is provided in Section 7.2.

7.6 RADIATION MONITORING SYSTEMS

The radiation monitoring systems will include CAMs, continuous monitoring at the exhaust stacks, process control instruments, and personnel monitoring and dosimetry. Process control instruments used to analyze for uranium concentrations are described in each respective process system in Section 7.3.

The objective of the radiation monitoring system is to provide the RPF control room personnel with a continuous record and indication of radiation levels at selected locations where radioactive materials may be present, stored, handled, or inadvertently introduced. The system is also designed to ensure that there is accurate and reliable information concerning radiation safety as related to personnel safety. The design considerations for the radiation monitoring system include the following:

- Provision of information to RPF operators so that in the event of an accident resulting in a release of radioactive material, decisions on deployment of personnel can be properly made.
- Indication and recording in the control room of the gamma and airborne radiation levels in selected areas as a function of time, and, if necessary, alarming to indicate any abnormal radiation condition. These indicators aid in maintaining plant contamination levels as low as reasonably achievable (ALARA) and in minimizing personnel exposure to radiation.
- Provision of local alarms and/or indicators positioned at key points throughout the RPF where a substantial increase in radiation levels might be of immediate importance to personnel frequenting or working in the area.

Radiation Monitoring Locations

RAMs will be located in areas where personnel may be present and where radiation levels could become significant based on the following considerations:

- Occupancy status of the area, including time requirements of personnel in the area, the proximity to primary and secondary radioactive sources, and shielding
- Potential for increase in the background radioactivity level
- Desirability of surveillance of infrequently visited areas

CAMs will be located in work areas where there is a potential for airborne radioactivity. The CAMs will have the capability to detect derived air concentrations within a specified time.

7.6.1 Design Criteria

Design criteria for the radiation monitoring I&C systems are described in Section 7.2.

7.6.2 Design Basis and Safety Requirements

The design basis and safety requirements for the radiation monitoring I&C systems are described in Section 7.2. The ESFs for this system are listed in Chapter 6.0.

7.6.3 System Description

The radiation safety monitoring system will include CAMs, continuous monitors at the exhaust stacks, and personnel monitoring and dosimetry.

Three basic types of personnel monitoring equipment will be used at the facility: count rate meters (friskers), hand/foot monitors, and portal monitors. All personnel whose duties require entry to restricted areas will wear individual external dosimetry devices (e.g., passive dosimeters such as thermoluminescent dosimeters that are sensitive to beta, gamma, and neutron radiation) from a National Voluntary Laboratory Accreditation (NAVLAP)-certified vendor. Personnel monitoring and dosimetry is described in Chapter 11.0, "Radiation Program and Waste Management."

7.6.3.1 Air Monitoring

Continuous air monitors – CAM units will consist of a particulate measuring channel with a filter to capture particulate. Air will be drawn through the system by a pump assembly. The sample will be withdrawn from inside the appropriate area, room, or cell through an isokinetic nozzle with the sampling volume flow at a known fixed rate, so that the accumulation of radioactive particles can be interpreted as a quantitative sample. After passing through the nozzle, the sample will be drawn through tubing and through a fixed or moving filter tape before being discharged to the atmosphere. The samplers also have a purging system for flushing the volume cell surrounding the gas sample chamber with clean air for purposes of calibration and the removal of crust activity. Replaceable liners will be changed out periodically when contamination becomes excessive. Flow regulating will ensure that flow through the filters remains constant.

Each instrument channel will include a detector, preamplifier, count rate meter, and power supply. The detector may be a scintillation counter or similar device having a gamma sensitive crystal, and a photo multiplier whose output pulses are counted by the rate meter. Each readout module will be equipped with a light that illuminates when the radiation level exceeds preset limits. The setpoint will be adjustable over the entire detection range. Pressing a button will cause the meter to indicate the alarm setpoint. Visible alarms will be accompanied by a simultaneous local audible alarm with an alarm light in the control room. A normally energized light will deenergize when there is a detector signal failure, circuit failure, power failure, or failure due to a disconnected cable. Power for the monitors that initiates a safety signal will be provided from the UPS. Loss of power and signal failure will be monitored for each detector.

CAMs will be provided with a check source. This check source will simulate a radiation field and will be used as a convenient operational and gross calibration check of the detectors and readout equipment. CAM calibration will include, where practical, exposures to the specific isotopes that the particular system monitors in the field. Instrument calibrations will be performed at prescribed frequencies. An electronic test signal and/or radioactive check source drift indication may also require CAM recalibration.

Radiation area monitors – The RAM detector unit will be housed in an environmentally suitable container that is mounted in a duct, on a wall, or other suitable surface. The sensitivity of each detector will be sufficient to have the alarm setpoint an order of magnitude higher than the detection threshold.

The detectors are designed to be operational over a wide range of temperatures. The design of the detectors will meet expected normal and abnormal environmental design conditions, as appropriate. Saturation will not be expected to adversely affect operation of the detector within its calibrated range.

Sensors will be mounted as close as practical to the most probable radiation sources with no objects, persons, pillars, and piping serving as shielding. The sensors will also be mounted so as to minimize inaccuracies due to any directionality of the detector.

Audible and visual alarm devices – When the radiation exceeds predetermined levels, alarms will actuate in the control room and at selected detector locations.

The alarms will consist of the following capabilities:

- "Alert light" will illuminate when the radiation level exceeds preset limits with an adjustable setpoint
- "High alarm red light" will illuminate when radiation levels exceed a predetermined alarm setpoint
- "Failure alarm" will sound when either the power or the channel's electronics fail

The visual alarms will be accompanied by a simultaneous audible alarm annunciator at the selected detector locations and in the control room. The annunciator windows for the monitors will be located in the control room. The alarm can be manually reset when the alarm conditions are corrected. The local alarm horns and warning lights will remain on until the radiation level is below the present level.

Additional CAM requirements and locations are described in Chapter 11.0.

7.6.3.2 Stack Release Monitoring

The exhaust stacks will be provided with continuous monitors for noble gases, particulate, and iodine. The stack monitoring system design basis is to continuously monitor the radioactive stack releases. Additional information will be provided in the Operating License Application. Airborne exposure pathway monitoring is described in Chapter 11.0.

7.6.4 System Performance Analysis and Conclusion

The system performance analysis and conclusion for each process system will be provided in the Operating License Application. The overall I&C system performance analysis is provided in Section 7.2.

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Public
Chapter 8

NWMI
NORTHWEST MEDICAL ISOTOPES

Chapter 8.0 – Electrical Power Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0
June 2015

Prepared for:
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Chapter 8.0 – Electrical Power Systems

Construction Permit Application for Radioisotope Production Facility

NWMI-2013-021, Rev. 0

Date Published:
June 29, 2015

***Document Number*:** **NWMI-2013-021**	***Revision Number*:** **0**
***Title*: Chapter 8.0 – Electrical Power Systems Construction Permit Application for Radioisotope Production Facility**	
***Approved by*: Carolyn Haass**	***Signature*:** Carolyn C Haass

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REVISION HISTORY

Rev	Date	Reason for Revision	Revised By
0	6/29/2015	Initial Application	Not required

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CONTENTS

FIGURES

TABLES

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

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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 could 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. The table presents the NEP system peak loads, identifies which systems have UPSs, and lists 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 power requirement[a]		Uninterruptable power	Standby electrical power requirement[a]	
	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 recyle system	10	13	No	10	13
Waste handling system	25	34	No	5	7
Radiation monitoring and CAAS systems	5	7	Yes[b]	5	7
Standby electrical power system	N/A		No	N/A	N/A
General facility electrical power	173	232	Yes[b]	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[b]	0[c]	0[c]
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
Facility chilled water system	1,300	1,743	No	0	0

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

Demand	Normal electrical power requirement[a]		Uninterruptable power	Standby electrical power requirement[a]	
	kW	hp		kW	hp
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] Peak power loads.

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

[c] 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 is to 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 90 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 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 12 hr 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 required for the RPF is 1,140 kW (1,528 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 90 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 90 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 90 min. The diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will 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 90 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will 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 90 min, while the diesel generator will maintain power until the normal power system is operating within acceptable limits. Additional information will 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.

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