NWMI
NORTHWEST MEDICAL ISOTOPES

ATTACHMENT 3

Northwest Medical Isotopes, LLC

Response to the U.S. Nuclear Regulatory Commission Request for Additional Information Regarding the Preliminary Safety Analysis Report and Environmental Review of the Northwest Medical Isotopes, LLC Construction Permit Application Docket No. 50-609

Dated: September 29, 2016

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NWMI-2016-RAI-004, Rev. 0
November 2016

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Response to the U.S. Nuclear Regulatory Commission Request for Additional Information Regarding the Preliminary Safety Analysis Report of the Northwest Medical Isotopes, LLC Construction Permit Application Docket No. 50-609

NWMI-2016-RAI-004, Rev. 0

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Title: **Response to the U.S. Nuclear Regulatory Commission Request for Additional Information Regarding the Preliminary Safety Analysis Report of the Northwest Medical Isotopes, LLC Construction Permit Application Docket No. 50-609**	
Approved by: **Carolyn Haass**	***Signature***: Carolyn C Haass

REVISION HISTORY

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CONTENTS

ATTACHMENTS

TERMS

Acronyms and Abbreviations

^{99}Mo	molybdenum-99
^{235}U	uranium-235
^{238}U	uranium-238
ADAMS	Agencywide Documents Access and Management System
ALARA	as low as reasonably achievable
ALI	annual limits on intake
ANECF	average neutron energy causing fission
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOA	area of applicability
ASCE	American Society of Civil Engineers
ASTM	American Society for Testing and Materials
BMS	building management system
C-I	Seismic Category I
C-II	Seismic Category II
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CF_2	carbon fluoride
CFR	Code of Federal Regulations
CP	Construction Permit
DAAP	diamylamylphosphonate
DAC	derived air concentration
DBE	design-basis event
EDE	effective dose equivalent
EOI	end of irradiation
ESF	engineering safety feature
FEMA	Federal Emergency Management Agency
FPC	facility process control
FSAR	final safety analysis report
H_2	hydrogen gas
HEGA	high-efficiency gas adsorption
HEPA	high-efficiency particulate air
HEU	high-enriched uranium
HMI	human-machine interface
HVAC	heating, ventilation, and air conditioning
I&C	instrument and control
IBC	International Building Code
IEU	intermediate-enriched uranium
IROFS	items relied on for safety
ISA	integrated safety analysis
ISG	Interim Staff Guidance
k_{eff}	k-effective
LEU	low-enriched uranium
MCE_R	maximum-considered earthquake
MCNP	Monte Carlo N-Particle
MHA	maximum hypothetical accident
MMI	Modified Mercalli Intensity
MoS	margin of subcriticality

MU	University of Missouri
MURR	University of Missouri Research Reactor
NCS	nuclear criticality safety
NESHAP	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NS	non-seismic
NSR	nonsafety-related
NWMI	Northwest Medical Isotopes, LLC
OBE	operating basis earthquake
OSL	optically stimulated luminescence
OSTR	Oregon State University TRIGA Reactor
OSU	Oregon State University
PGA	peak ground acceleration
PHA	process hazard analysis
PPE	personal protective equipment
PSAR	preliminary safety analysis report
QA	quality assurance
QAPP	quality assurance program plan
QL	quality level
RAI	request for additional information
RAM	radioactive material
RCA	radiologically controlled area
RPF	Radioisotope Production Facility
RSAC	Radiological Safety Analysis Computer
SAR	safety analysis report
SEP	standby electrical power
SNM	special nuclear material
SR	safety-related
SSC	structures, systems, and components
SSE	safe-shutdown earthquake
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TRIGA	Training, Research, Isotopes, General Atomics
U	uranium
U.S.	United States
UC	uranium carbide
UH_3	uranium trihydride
UO_2	uranium dioxide
$UO_2(NO_3)_2$	uranyl nitrate
UPS	uninterruptable power supply
USGS	United States Geological Survey
UZrH	uranium zirconium hydride
WCDE	weighted committed dose equivalent

Units

°C	degrees Celsius
°F	degrees Fahrenheit
μ	micron
cm	centimeter
cm^2	square centimeter
ft	feet
ft^2	square feet
g	gram
g	ground acceleration
gal	gallon
ha	hectare
hp	horsepower
hr	hour
in.	inch
in.^{2}	square inch
kg	kilogram
km	kilometer
km^2	square kilometer
kPa	kilopascal
kW	kilowatt
L	liter
lb	pound
m	meter
MeV	million electron volts
mg	milligram
mi	mile
mi^2	square mile
min	minute
mrem	millirem
mSv	millisievert
rem	roentgen equivalent in man
wk	week
yr	year

GENERAL INFORMATION

No.	Request for Additional Information
	General Information
RAI G-2	*Section 50.33(h) of 10 CFR Part 50 states that "If the applicant, other than an applicant for a combined license, proposes to construct or alter a production or utilization facility, the application shall state the earliest and latest dates for completion of the construction or alteration."* *In "General Information in Accordance with 10 CFR 50.33" of the introduction section of the CP application, NWMI states that it "expects to complete construction of the facility at earliest and latest by the second quarter 2016 and fourth quarter of 2017, respectively."* *NWMI's earliest and latest dates for construction completion are not consistent with the CP application review schedule the NRC staff provided to NWMI on March 28, 2016 (ADAMS Accession No. ML16056A122) or the timing of a Commission decision.* *Provide revised earliest and latest dates that NWMI expects to complete construction of the facility.*

The earliest and latest dates for construction completion of the Northwest Medical Isotopes, LLC (NWMI) radioisotope production facility (RPF) are third quarter 2018 and first quarter 2019, respectively. This construction completion date is contingent upon the U.S. Nuclear Regulatory Commission (NRC) approval no later than the end of third quarter 2017.

RAI G-3	*NUREG-1537, Part 2, Chapter 13, as augmented by the ISG, is applicable to reviewing a description of the accident analyses for the licensing of a radioisotope production facility and a non-power reactor facility. Whenever the term "reactor" appears, it is understood to mean a "non-power reactor facility," a "radioisotope production facility," or both as applicable.* *NUREG-1537, Part 2, Chapter 13, "Accident Analyses," Acceptance Criteria, states that:* *For a research reactor, the results of the accident analysis have generally been compared with 10 CFR Part 20 criteria (10 CFR 20.1 through 20.602 and appendices for research reactors licensed before January 1, 1994, and 10 CFR 20.1001 through 20.2402 and appendices for research reactors licensed on or after January 1, 1994)...for research reactors licensed on or after January 1, 1994, occupational exposure is discussed in 10 CFR 20.1201 and public exposure is discussed in 10 CFR 20.1301. In several instances, the staff has accepted very conservative accident analysis with results greater than the 10 CFR Part 20 dose limits discussed above.* *The ISG Augmenting NUREG-1537, Part 2, Chapter 13, "Accident Analyses," Section 13b.1.2, "Accident Initiating Events," states that among other considerations, the reviewer should confirm several items including: (a) instruments, controls, and automatic protective systems were assumed to be operating normally or to be operable before the initiating event; (b) maximum acceptable non-conservative instrument error may be assumed to exist at accident initiation; (c) credit was taken during the scenario for normally operating process systems; and (d) protective actions were initiated by either the operating staff, control systems, or engineered safety features.* *As stated in Chapter 13b, "Radioisotope Production Facility Accident Analyses," of the ISG Augmenting NUREG-1537, Part 1, the NRC staff has determined that the use of Integrated Safety Analysis methodologies as described in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and NUREG-1520 "Standard Review Plan for Fuel Cycle Facilities License Applications," as well as application of the radiological and chemical consequence and likelihood criteria contained in the performance requirements of 10 CFR 70.61, designation of items relied on for safety, and establishment of management measures, are acceptable ways of demonstrating adequate safety for the radioisotopes production facility. Applicants may propose alternate accident analysis methodologies, alternate radiological and chemical consequences and likelihood criteria, alternate safety features, and alternate methods of assuring availability and reliability of the safety features.* *NWMI PSAR, Section 13.2.1, "Maximum Hypothetical Accident," provides a description and analyses of the maximum hypothetical accident (MHA) at the NWMI facility. The PSAR states that the event is not credible and assumes that the offgas treatment system that releases all radioiodine and noble gas radioisotopes retained in that system from the radioisotopes production facility stack without mitigation.*

No.	Request for Additional Information
	Additionally, the PSAR states that the dose consequences from the event are within the intermediate consequence level as defined in 10 CFR 70.61, "Performance requirements." *PSAR Section 13.2.1 contains a mix of MHA and 10 CFR 70.61 methodologies. It is not apparent which accident analysis philosophy is being used for the RPF. The accident analysis in PSAR Section 13.2.1 makes overly conservative assumptions relative to the accident analysis methodology that was discussed in ISG Augmenting NUREG-1537, Part 1, Chapter 13b that was derived from NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications," for fuel facilities, or the methodology that has been applied for non-power reactors. For example, the MHA analyses assumes that safety systems, such as, the offgas dissolver train and mitigating offgas treatment are not functional (note that the items relied on for safety (IROFS) is called Primary Offgas Release System). Furthermore, the MHA dose acceptance criteria used by NWMI is significantly greater than the requirements of 10 CFR Part 20 which are generally applied to the MHA for non-power reactors.* *Demonstrate that the MHA in PSAR Section 13.2.1, including the items stated in the ISG Augmenting NUREG-1537, Part 2, Section 13b.1.1, such as appropriately functioning instruments, controls, automatic protective systems normally operating process systems and protective actions initiated by either the operating staff, control systems, or engineered safety features meets the generally accepted dose requirements of 10 CFR Part 20.* *Otherwise, provide an accident analysis that is either consistent with the requirements of 10 CFR 70.61 (e.g., application of items relied upon for safety to prevent or mitigate the event) or propose an alternate methodology.*

The maximum hypothetical accident (MHA) discussion was inappropriately included in Section 13.2.1 of the preliminary safety analysis report (PSAR) (NWMI-2013-021, *Construction Permit Application for Radioisotope Production Facility*). NWMI interpreted NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria*, Part 2, Chapter 13, as augmented by the Interim Staff Guidance (ISG) (NRC, 2012), as requiring an analysis consistent with both 10 CFR 70.61, "Performance Requirements," and with the MHA approach. Additionally, the PSAR combined the two in its approach to the MHA, producing an analysis inconsistent with the intended requirements of an MHA analysis.

The accident analyses in the final safety analysis report (FSAR), as part of the Operating License Application, will be consistent with the requirements of 10 CFR 70.61. As stated in Chapter 13b, "Radioisotope Production Facility Accident Analyses," of the ISG Augmenting NUREG-1537, Part 1:

> *NRC staff has determined that the use of Integrated Safety Analysis methodologies as described in 10 CFR Part 70 Subpart H and NUREG-1520, application of the radiological and chemical consequence and likelihood criteria contained in the performance requirements of 10 CFR 70.61, designation of items relied on for safety, and establishment of management measures, are acceptable ways of demonstrating adequate safety for the medical isotopes production facility.*

The accident analyses in the PSAR are based on (1) use of integrated safety analysis (ISA) methodologies, as described in 10 CFR 70 Subpart H and NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, (2) application of the radiological and chemical consequence and likelihood criteria contained in the performance requirements of 10 CFR 70.61, (3) designation of items relied on for safety (IROFS), and (4) establishment of management measures to demonstrate adequate safety.

The ISA includes a systematic analysis and discussion of credible accidents for determining the limiting events for several accident categories. The limiting event in each category is analyzed quantitatively to determine consequences. Radiological accident consequences, as mitigated by structures, systems, and components (SSC) and administrative safety measures, are evaluated against the performance requirements of 10 CFR 70.61. The safety measures are designated as IROFS.

No.	Request for Additional Information
	Based on the response to RAI G-3, PSAR Section 13.2, "Analysis of Accidents with Radiological and Criticality Safety Consequences" will be revised and the MHA discussion in PSAR Section 13.2.1 will be deleted.
RAI G-4	*Subparagraph 50.35(a)(1) of 10 CFR Part 50 requires that the applicant describe the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and identifies the major features or components incorporated therein for the protection of the health and safety of the public.* *NWMI PSAR, Section 3.5.1.3.1, "Safety-Related Structures, Systems, and Components," describes those structures, systems and components (SSC) that are relied on to remain functional during normal conditions and during and after design-basis events (DBE) will ensure that acute chemicals exposures to an individual produced from licensed material or hazardous chemicals will not lead to:* • *Irreversible or other serious, long lasting health effects to workers* • *Mild transient health effects to individuals located outside [radiological production facility] RPF [radiologically controlled area] RCA* *It is not clear in the PSAR if the SSCs will protect for high consequences which could lead to endangerment of the life of a worker and/or lead to irreversible or other long lasting health effects to individual outside the controlled area.* *Provide additional information explaining why the SSCs will be designed to protect against intermediate consequences, and not for high consequences.*
	SSCs will be designed to protect against both high and immediate consequences. PSAR Section 3.5.1.3.1 described both the high and the immediate consequence performance requirements from 10 CFR 70.61. To eliminate confusion and ensure completeness, these bullets were removed from PSAR Section 3.5.1.3 (see Attachment A), and the 10 CFR 70.61 performance requirements are referenced.

CHAPTER 2.0 – SITE CHARACTERISTICS

No.	Request for additional information
	Section 2.2 – Nearby Industrial, Transportation, and Military Facilities
RAI 2.2-1	*NUREG-1537, Part 1, Section 2.2.2, "Air Traffic," states that factors such as frequency and type of aircraft movement, flight pattern, local meteorology, and topography should be considered for sites located within 8 kilometers of an existing or projected commercial or military airport and that the analysis should demonstrate that there is a low potential that any aircraft could affect the [RPF] reactor or that the consequences from any aircraft-associated accident are already bounded or considered in the accident analysis.* *NWMI PSAR Section 2.2.2.1, "Airports," describes three helicopter ports (University Hospitals, Missouri University (MU), and Clinics heliports) within 8 km of the radiological production facility (RPF), and two small private airports (Cedar Creek Airport and Sugar Branch Airport) within 16 km of the RPF site.* *The 200d² calculation uses a shorter distance of 10.4 km than that stated in the first paragraph of 10.5 km. NWMI does not provide an analysis of possible effects to the RPF to evaluate any potential radiological risks to the facility staff, the public, and the environment. This information is needed to determine the potential radiological risks, if any, to the facility staff, the public, and the environment resulting from aircraft-associated accidents.*
RAI 2.2-1a	*Provide an analysis that demonstrates the risk to be low potential from hazards posed by activities associated with the referenced heliports and airports, or how the RPF design can accommodate any possible hazards associated with these heliports and airports and impact with the facility.*

There are three airports and three helicopter ports located within 16 kilometers (km) (10 miles [mi]) of the proposed RPF site. The three airports include:

- Columbia Regional Airport (public) located approximately 10.4 km (6.5 mi) south of the RPF site
- Cedar Creek Airport (private) located approximately 10.6 km (6.6 mi) northeast of the RPF site
- Sugar Branch Airport (private) located approximately 15.6 km (9.7 mi) northwest of the RPF site

These airports are identified in PSAR Chapter 2.0, Figure 2-30, of the Construction Permit Application.

The nearest airport to the RPF is the Columbia Regional Airport, which is used by commercial and privately owned aircraft. The airport is situated on approximately 0.532 ha (1,314 acres) and is owned and operated by the City of Columbia. This airport is the only public use airport located in Boone County, Missouri, for which records are kept. For the 12-month period ending October 31, 2013, the airport had 16,610 aircraft operations for an average of 26 flights per day, including:

- 81 percent general aviation
- 16 percent air taxi
- 2 percent military
- 1 percent air carrier

Cedar Creek airport is a private turf landing strip approximately 10.6 km (6.6 mi) northeast of the RPF site. The facility houses two private single engine aircraft. The specific number of flights to and from the facility is not available.

The Sugar Branch airport is a private, turf landing strip approximately 15.6 km (9.7 mi) northwest of the RPF site. The facility houses one single engine aircraft. The specific number of flights to and from the facility are not available.

Three helicopter ports are located within 16 km (10 mi) of the RPF site and support hospital operations, including:

- University of Missouri Hospitals and Clinics heliport located 6 km (3.7 mi) northwest
- University of Missouri (MU) heliport located 6 km (3.7 mi) northwest
- Boone Hospital Center heliport located 6.3 km (3.9 mi) northwest.

No operations data are available for these heliports.

Based on NUREG-1537, sites located between 8 km (5 mi) and 16 km (10 mi) from an existing or projected commercial or military airport with more than approximately 200 d^2 (where d is the distance in kilometers from the airport to the RPF site) commercial or military aircraft movements per year, the probability of aircraft accidents is considered less than an order of magnitude of 10^{-7} per year.

The number of operations at the Cedar Creek and Sugar Branch airports are not available. However, daily operations were assumed based on the aircraft housed, including two operations per day from Cedar Creek (730 operations/year) and one operation per day from Sugar Branch (365 operations/year). Based on the results presented in Table 1, all three airports are under the 200 d^2 limits.

Table 1. 200 D^2 Limits

Airport	Distance km (mi)	Flights per year	200 d^2 limits[a]
Columbia Regional Airport	10.4 (6.5 mi)	16,610	21,632
Cedar Creek	10.6 (6.6 mi)	730	22,472
Sugar Branch	15.6 (9.7 mi)	365	48,672

[a] d is the distance in kilometers from the airport to the RPF site (200 × distance squared).

Based on this requirement, none of these airports needs to be further evaluated. The guidance also requires that special consideration be given to facilities sited within the trajectory of a runway of any airport. The RPF site is not located within a trajectory of a runway of the airport.

Because the three heliports are closer than 8 km (5 mi) to the RPF site, the frequency of an aircraft crashing into the site needs to be evaluated. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Subsection 3.5.1.6, provides a methodology for determining the probability of an aircraft crash into a facility from airways. However, the approach requires knowledge of the number of flights per year along the airway. Because this information is not available for the flight paths near the RPF, DOE-STD-3014-2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities*, was used to determine the frequency of crashes. The following equation is used.

$$F_h = N_h \times P_h \times f_h(x,y) \times A_h$$

Where:

F_h	=	Crash impact frequency
N	=	Flight per year
P_h	=	Probability of a crash
$f_h(x,y)$	=	Probability, given a crash, that the crash occurs in a 1-mi^2 area surrounding the facility
A_h	=	Effective plant area.

The effective area for an aircraft was determined by two components: the aircraft crashing into the facility either by skidding or by flying directly into it. The effective area was calculated based on an aircraft skidding or flying into the facility in the direction that produces the largest area (i.e., crashing in a direction perpendicular to the largest diagonal of the building).

The following formula was used to calculating the skid and fly in areas of an aircraft crashing into the facility.

$$A_{eff} = A_f + A_s$$

Where:

$$A_f = (WS + R) \times H \times cot\phi + \frac{(2 \times L \times W \times WS)}{R} + L \times W$$

and:

$$A_s = (WS + R) \times S$$

Where:

A_f	=	Effective fly-in area
As	=	Effective skid area
WS	=	Aircraft wingspan
R	=	Length of the diagonal of the facility = $\sqrt{L^2 + W^2}$
H	=	Facility height, facility-specific
cotΦ	=	Mean of the cotangent of the aircraft impact angle
L	=	Length of facility, facility-specific
W	=	Width of facility, facility-specific
S	=	Aircraft skid distance (mean value).

DOE-STD-3014-2006 notes that in calculating an effective area, the analyst needs to be cognizant of the "critical areas" of the facility. The critical areas are locations in a facility that contain hazardous material and/or locations that, once impacted by a crash, can lead to cascading failures (e.g., a fire, collapse, and/or explosion that would impact the hazardous material). The critical areas of the RPF are considered to be the hot cell and waste management areas.

The critical areas dimensions are estimated at 30.5 × 24 meters (m) (100 × 80 feet [ft]), which provides a diagonal (R) of 39 m (128 ft). The facility height (H) of 22.9 m (75 ft) was used. DOE-STD-3014-2006 provides estimates for aircraft wingspan, mean of the cotangent of the aircraft impact angle, and skid distance for five different aircraft types. For helicopters, the cotΦ value is 0.58 and the skid length is typically assumed to be 0. The effective area is calculated in Table 2.

Table 2. Affective Area for Helicopter

Aircraft	Wing span[a] WS (ft)	cotΦ[a]	Skid distance[a] S (ft)	Effective plant area A_h (mi^2)
Helicopter	50	0.58	0	0.00079

[a] DOE-STD-3014-2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities*, U.S. Department of Energy, Washington, D.C., 1996 (R2006).

For a helicopter, fh(x,y) is estimated based on half the average length of a flight with the lateral variations in crash locations assumed to be one-quarter mile on the average from the centerline of the flight path or 2/L. The probability Ph (2.50E-05) is taken from DOE-STD-3014-2006, Appendix B Table B-1. The total number of flights from the three helipads are estimated at 1,825 per year. A conservation estimate is that 5 percent of these helicopters overfly the facility. In addition, a conservative estimate of total flight path is the distance to the closest helipad or 6 km (3.7 mi).

Based on these assumptions, the helicopter impact frequency is calculated as follows:

$$F_h = 91 \times 2.5E^{-05} \times \frac{2}{3.7} \times 7.9E^{-04}$$

$$F_h = 9.7E^{-07}$$

The calculated crash impact frequency from the heliport is less than the requirement of NUREG-0800 of being within an order of magnitude of 10^{-7} per year. Therefore, no further analysis is required. This information will be added to PSAR Section 2.2.2.

No.	Request for additional information
RAI 2.2-1b	*Justify using 10.4 km in the 200d² calculation when the distance to the Columbia Regional Airport to the RPF site is stated as 10.5 km.*

PSAR Section 2.2.2.1 had a typographical error. 10.4 km (6.5 mi) is the correct distance from the Columbia Regional Airport to the RPF site, based on Google Earth measurements, and 10.4 km (6.5 mi) is the distance used in the associated calculations. The stated distance of 10.5 km will be changed to 10.4 km (6.5 mi) in PSAR Section 2.2.2.1.

RAI 2.2-2	*NUREG-1537, Part 1, Section 2.2.3, "Analysis of Potential Accidents at Facilities," states that if a facility (i.e., nearby industrial, transportation, or military facility) "cannot affect the [RPF], the applicant should make a statement to that effect and give the basis for this statement."* *Furthermore, NUREG-1537, Part 2, Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," states, in part, that the review:* *...should confirm that any hazards to the [RPF] facility posed by normal operations and potential malfunctions and accidents at nearby manmade stationary facilities... have been described and analyzed to the extent necessary to evaluate the potential radiological risks to the facility staff, the public, and the environment.* *NWMI PSAR Section 2.2.3.1.3, "Flammable Vapor Clouds (Delayed Ignition)," Table 2-19, "Flammable Vapor Clouds and Vapor Cloud Explosion from External Sources," does not include the acceptable distance for diesel at MU South Farm.* *PSAR Section 2.2.3.1.2, "Nearby Facilities," Table 2-17 and Table 2-19 only considers a portion of the total amounts of gasoline and diesel storage at the Magellan Facility, and the propane at MU South Farm, to determine the acceptable distance of peak positive incident overpressure of 6.9 kPa (1 lb/in.²). NWMI indicates that an analysis accounting for the total amount of gasoline and diesel stored at the Magellan facility yields a minimum separation distances (i.e., safe standoff distances) greater than its current location to the RPF. Such scenario may pose a risk to the RPF since the location of those tanks are closer than the safe standoff distance. It presents the same potential risk for an analysis accounting for the total amount of propane stored at MU South Farm facility. The PSAR seems to only consider the acceptable distance to the largest tank of gasoline, diesel, or propane, at these nearby facilities which is significantly less than the total inventory. Therefore, additional information is needed for the NRC staff to determine that potential accidents at nearby facilities would not pose sufficient risk to the RPF to render the site unsuitable for construction and operation and to understand the potential radiological risks, if any, to the facility staff, the public, and the environment, resulting from accidents at nearby facilities.*
RAI 2.2-2a	*Clarify whether the assumptions in PSAR Section 2.2.3.1, "Determination of Design-Basis Events," bound any other possible explosion scenario, such as the explosion of the total inventory at nearby facilities that could potentially affect RPF operations or safe shutdown.*

The analysis performed does not bound an explosion of the total inventory of nearby facilities. The analysis uses the largest tank for two identified facilities to determine the effect on RPF operations or safe shutdown. It was determined to be highly unlikely for the total inventory from both facilities to be involved in the explosion scenario due to the following. At the MU South Farm, the closest facility, the total inventory of propane is in multiple disperse locations. For the Magellan Pipeline facility, an accidental explosion of multiple tanks at one time adding to the pressure wave is also highly unlikely.

RAI 2.2-2b	*Provide the acceptable distance for diesel at MU South Farm and describe potential effects in the RPF, if any.*

As identified in PSAR Section 2.2.3.1.2, Table 2-17, the acceptable distance or diesel at the MU South Farm is 0.94 km (0.58 mi). The diesel tanks are 1.6 km (1 mi) from the RPF. Based on the explosion analysis conducted in EDF-3124-0016, *Analysis of Potential Accidents at Nearby Facilities*, an explosion of the tanks would not affect the RFP.

No.	Request for additional information
	Section 2.3 – Meteorology
RAI 2.3-1	*NUREG-1537, Part 2, Section 2.1, "Geography and Demography," states, in part, that "as part of this review, the reviewer should check the exclusion area distances against distances used in analysis presented in Chapter 11 and 13 of the SAR [safety analysis report]."* *NWMI PSAR Section 2.3.2, "Site Meteorology," states, in part, that "conservative assumptions were used, in both the Radiological Safety Analysis Computer code to support 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center distance." However, the exclusion area is not specifically described in this Section 2.1 or in Chapter 11 and 13 of the PSAR.* *Confirm that the exclusion area boundary is what is described in Chapter 11 of the PSAR as the "controlled area." If not the same, describe the exclusion area boundary for the RPF.*

The boundary of the "controlled area" described in PSAR Chapters 11.0 and 13.0 is the same as the "exclusion area boundary." PSAR Chapters 2.0, 11.0, and 13.0 will be updated to use the same terminology when referring to the "exclusion area boundary."

RAI 2.3-2	*NUREG-1537, Part 1, Section 2.3.1, "General and Local Climate," states that historical seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes, waterspouts, thunderstorms, lightning, and hail, should be stated. The applicant should give the known and maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. The applicant should estimate the 100-year return wind speed.* *NWMI PSAR, Section 2.3.1.7, "Extreme Weather," states that the RPF location area exhibits significant atmospheric instability, heavy precipitation, and many intense thunderstorms and is located in a tornado prone area, but did not provide annual frequencies of severe weather phenomena. This information is needed to determine if a weather-related event of credible frequency or consequences at the site render it unsuitable for operation, as designed, and likely to cause damage to the RPF facility during its lifetime that could release uncontrolled radioactive material to the unrestricted area.*
RAI 2.3-2a	*Provide the seasonal and annual frequencies of tornadoes, thunderstorms, lighting, and hail.*

The seasonal and annual frequencies of tornadoes, thunderstorms, lighting, and hail are provided as follows in Table 3 through Table 9. This information will be added to PSAR Section 2.3.1.7.

Table 3. Seasonal Frequency of Historical Tornadoes in Boone County, Missouri (1954 to 2016)

Month	Magnitude (Fujita Scale)			
	F0	F1	F2	F3
January	1	-	-	-
February	1	-	-	-
March	-	2	-	-
April	1	2	5	-
May	1	1	2	-
June	1	1	-	-
July	2	1	-	-
August	-	-	-	-
September	-	2	-	-
October	2	1	-	-
November	-	-	-	3
December	1	1	1	-

Source: http://www.ncdc.noaa.gov/stormevents/

Table 4. Annual Frequency of Historical Tornadoes in Boone County, Missouri (1954 to 2016)

Year	Magnitude (Fujita Scale)				Total
	F0	F1	F2	F3	
1954	-	-	3	-	3
1956	-	-	1	-	1
1959	2	1	-	-	3
1965	1	-	-	-	1
1966	-	1	-	-	1
1972	-	1	-	-	1
1973	1	1	1	-	3
1980	-	-	1	-	1
1982	1	1	-	-	2
1984	-	3	-	-	3
1985	-	1	-	-	1
1987	1	-	-	-	1
1990	-	-	-	2	2
1992	1	1	-	-	2
1995	1	-	-	-	1
1998	-	-	-	1	1
1999	-	-	2	-	2
2000	1	1	-	-	2
2001	1	-	-	-	1

Source: http://www.ncdc.noaa.gov/stormevents/

Table 5. Boone County Seasonal Thunderstorm Wind Events (8/29/1955 to 5/11/2016)

Month	Wind Velocity (mph)								
	70-74	75-79	80-84	85-89	90-94	95-99	100-104	105-109	110-114
January	-	2	-	-	-	-	-	-	-
February	-	-	-	-	-	-	-	-	-
March	-	8	1	3	2	-	-	-	-
April	-	12	5	2	2	-	-	1	-
May	-	13	7	9	3	2	1	1	2
June	-	20	3	6	3	1	1	2	-
July	-	12	8	10	6	1	2	2	-
August	1	18	6	2	3		1	1	-
September	-	4	1	3	-	-	-	-	-
October	-	-	-	-	-	-	-	-	-
November	-	-	1	-	-	-	-	1	-
December	-	2	-	-	-	-	-	-	-

Source: http://www.ncdc.noaa.gov/stormevents/

Table 6. Boone County Annual Thunderstorm Wind Events (8/29/1955 to 5/11/2016)

Year	Events	Year	Events	Year	Events	Year	Events
1956	1	1971	1	1987	2	2002	6
1957	-	1972	-	1988	2	2003	-
1958	3	1973	1	1989	-	2004	8
1959	-	1974	-	1990	3	2005	7
1960	-	1975	1	1991	1	2006	11
1961	3	1977	1	1992	1	2007	8
1962	-	1978	1	1993	-	2008	6
1963	2	1979	-	1994	2	2009	6
1964	-	1980	-	1995	5	2010	6
1965	-	1981	7	1996	2	2011	15
1966	2	1982	16	1997	1	2012	1
1967	3	1983	1	1998	9	2013	-
1968	-	1984	3	1999	1	2014	5
1969	1	1985	-	2000	17	2015	4
1970	1	1986	3	2001	6	2016	2

Source: http://www.ncdc.noaa.gov/stormevents/

Table 7. Boone County Lightning Events (7/5/1998 to 6/30/2016)

Location	Date	Description
Columbia	7/5/1998	Lightning strike was blamed for a fire at a residence in southwest Columbia. Firefighters arrived to find flames shooting through a hole in the roof.
Columbia	5/22/2002	A fire started by lightning destroyed 50 percent of a home in Columbia.
Columbia	8/25/2004	Lighting strike melted power lines at Providence and Green Meadows roads. About 5,000 people were affected by the resulting power outage, including New Haven Elementary School.
Columbia	8/25/2004	Lightning strike started a house fire.
Columbia	6/6/2005	Lightning strike started a house fire.
Columbia	8/26/2006	Five radio stations were knocked off the air when lightning struck a Cumulus Broadcasting transmitter tower. Control boards in the studios, computers, and magnetic door locks in the building were also damaged by the strike.
Columbia	7/19/2007	Lightning strike started a fire at a photography studio.
Sapp	4/23/2008	Lightning strike started a house fire.
Columbia	5/30/2008	Lightning strike started a house fire.
COU Memorial Airport	6/13/2008	Lightning strike started a house fire.
Browns	6/17/2009	Lightning strike killed woman in an open field at Rocky Fork Lakes Conservation Area.
Harg	7/3/2011	Lightning strike started a house fire.
Columbia	7/23/2011	Lightning struck cell phone being used by woman in Cosmo Park.

Source: http://www.ncdc.noaa.gov/stormevents/

Table 8. Boone County Seasonal Hail Events 4/23/1958 - 5/11/2016

Location	Diameter (in.)											Total
	0.75	0.88	1.00	1.25	1.50	1.75	2.00	2.50	2.75	3.00	4.00	
January	2	1	3	-	-	-	-	-	-	-	-	9
February	1	-	-	-	-	-	-	-	-	-	-	1
February	-	-	1	-	-	-	-	-	-	-	-	1
March	18	4	20	2	3	11	1	1	-	1	-	61
April	21	6	18	4	3	15	2	-	3	-	-	72
May	33	21	21	2	3	22	1	-	1	-	1	105
June	15	8	9	3	1	12	1	-	-	-	-	49
July	5	1	3	-	-	2	-	-	-	-	-	11
August	1	1	2	-	1	1	-	-	-	-	-	6
September	8	2	4	-	1	3	-	-	1	-	-	19
October	-	-	-	-	-	1	-	-	-	-	-	1
November	1	2	5	-	-	3	-	-	2	-	-	13
December	-	2	2	-	-	1	-	-	-	-	-	5

Source: http://www.ncdc.noaa.gov/stormevents/

Table 9. Boone County Annual Hail Events 4/23/1958 - 5/11/2016

Year	Events	Year	Events	Year	Events	Year	Events
1958	1	1972	-	1987	1	2002	13
1959	1	1973	3	1988	5	2003	13
1960	1	1974	6	1989	1	2004	-
1959	1	1975	1	1990	4	2005	36
1961	1	1976	2	1991	5	2006	49
1962	2	1977	1	1992	7	2007	5
1963	1	1978	-	1993	4	2008	19
1964	-	1979	-	1994	3	2009	11
1965	1	1980	1	1995	10	2010	7
1966	2	1981	4	1996	5	2011	21
1967	1	1982	15	1997	1	2012	8
1968	3	1983	1	1998	3	2013	8
1969	1	1984	15	1999	7	2014	9
1970	2	1985	2	2000	13	2015	3
1971	-	1986	2	2001	10	2016	2

Source: http://www.ncdc.noaa.gov/stormevents/

No.	Request for additional information
RAI 2.3-2b	*Provide the maximum annual frequency of occurrence and time duration of freezing rain and discuss the potential effects of these meteorological events on the RPF.*

Winter weather events since 1996 in Boone County, Missouri are provided in Table 10. These events include snowstorms, ice storms, and extreme cold events. The RPF is being designed to ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, to withstand expected meteorological events. This information will be factored in the design requirements of PSAR Section 3.2.5, "Rain, Snow, and Ice Loading" for the RPF. Table 10 will be added to PSAR Section 3.2.5.

Table 10. Boone County Winter Weather Events (1/1/1996 to 6/30/2016) (2 pages)

Date	Storm type	Duration (days)	Description
01/02/96	Winter storm	1	6-9 inches of snow in region
01/03/96	Winter storm	2	
11/25/96	Ice storm	1	Numerous traffic accidents
01/08/97	Winter storm	2	5-7 inches of snow, strong winds, very cold temperatures
01/15/97	Winter storm	2	Freezing rain and sleet with ¼ to ½ in. of ice accumulation followed by 3 to 8 in. of snow in the region
01/27/97	Winter storm	1	Freezing rain with ½ to 1 in. of ice accumulation
04/10/97	Winter storm	1	2 to 6 in. of snow in the region
12/08/97	Winter storm	1	2 to 4 in. of snow in region
01/12/98	Winter storm	1	Freezing drizzle resulting in thin glaze of ice on roads
03/08/98	Winter storm	2	4 to 6 in. of snow in region
12/21/98	Winter storm	2	Light freezing drizzle, sleet, and snow left a thin coating of ice on roads
01/01/99	Winter storm	2	6 to 10 in. of snow across region with about an inch of freezing rain and sleet; very cold temperatures
01/27/00	Winter storm	3	4 to 5 in. across region
03/11/00	Winter storm	1	4 to 7 in. of snow
12/13/00	Heavy snow	1	6 to 12 in. across region
12/16/00	Extreme cold/wind chill	2	Wind chills from -20°F to -40°F
01/29/02	Ice storm	2	1¼ to ½ in. of ice accumulation; power outages
03/02/02	Winter storm	1	½ in. of sleet followed by 4 to 6 in. of snow; winds of 20 to 30 mi/hr
03/25/02	Winter storm	2	Sleet followed by snow; 3- to 4-in. accumulation of the mix
12/04/02	Winter storm	1	2 to 5 in. of snow across region
12/24/02	Winter storm	1	4 to 8 in. of snow across region
01/01/03	Winter storm	2	Sleet accumulation up to 1 in. followed by 6 to 8 in. of snow across the region
02/23/03	Winter storm	2	3 to 6 in. of snow across the region
12/09/03	Winter storm	2	3 to 5 in. of snow across the region

Table 10. Boone County Winter Weather Events (1/1/1996 to 6/30/2016) (2 pages)

Date	Storm type	Duration (days)	Description
12/13/03	Winter storm	1	3 to 6 in. of snow across the region
01/25/04	Winter storm	1	Freezing rain followed by 1 to 2 in. of sleet and then 1 to 2 in. of snow
11/24/04	Winter storm	1	4 to 6 in. of snow across region
12/08/05	Winter storm	1	2 in. of snow
11/29/06	Winter storm	3	Over a foot of snow in some areas
01/12/07	Ice storm	3	Up to 1.5 in. of sleet and ¼ to ½ in. of ice accumulation in region
12/8/2007	Ice storm	4	Up to a ½ in. of ice accumulated along with up to 1 in. of sleet
1/31/2011	Winter storm	2	Up to 20 in. of snow fell along with winds gusting over 40 mi/hr.
12/21/2013	Ice storm	1	Average ice accumulation on trees and other overhead surfaces was from 0.25 to 0.30 in; about ½ inch of sleet also fell in some locations
1/5/2014	Winter storm	1	6 to 9 in. of snow across with strong northerly winds produced snow drifts of 2 to 5 ft
2/4/2014	Winter storm	1	6 to 13 in. of snow across the region

Source: http://www.ncdc.noaa.gov/stormevents/

No.	Request for additional information
	Section 2.5 – Geology, Seismology, and Geotechnical Engineering
Applies to RAIs 2.5-1 through 2.5-5	*NUREG-1537, Part 1, Section 2.5, "Geology, Seismology and Geotechnical Engineering," states that the applicant should detail the seismic and geologic characteristics of the site and region surrounding the site. The degree of detail and extent of considerations should be commensurate with the potential consequences.* *NUREG-1537, Part 1, Section 2.5.2, "Site Geology, states that the applicant should discuss in detail the structural geology at the facility site and should pay particular attention to specific structural units of significance to the site such as folds, faults, synclines, anticlines, domes and basins.* *In Sections 2.5.3, "Seismicity," 2.5.4, "Maximum Earthquake Potential," 2.5.5, "Vibratory Ground Motion," 2.5.6, "Surface Faulting," and 2.5.7, "Liquefaction Potential," the NUREG states, in part, that:* *The applicant should list all historically recorded earthquakes... of Modified Mercalli intensity of greater than IV or magnitude (Richter) greater than 3.0... [in the list]...the applicant should evaluate the largest earthquake that could occur... and isoseismal maps for the earthquakes should be presented" ... the applicant should assess the ground motion at the site from the maximum potential earthquakes... and the applicant should establish the vibratory ground motion design spectrum [and] the applicant should discuss soil structure and prepare an appropriate state of the-art analysis for liquefaction at the site.* *NUREG-1537, Part 2, Section 2.5 further states that the review should confirm that the information presented has been obtained from sources of adequate credibility and is consistent with other available data such as the final safety analysis report (FSAR) of a nearby nuclear power plant."*

No.	Request for additional information
RAI 2.5-1	*NWMI PSAR Section 2.5.1.3, "Local Topography and Soils of Boone County," states that several areas of the county contain well developed cave and sinkhole formations. PSAR Section 2.5.2.3, "Mississippian Age Osagean Series Burlington Formation," references a report by Terracon Consultants, Inc. (Terracon) that states that "no caves or sinkholes are known to exist, or are published to exist within approximately 1 mile of [this project site].... However, several areas of known karst activity are present ..."*
RAI 2.5-1a	*Confirm by reference or investigation that no new sinkhole formations have developed at the project site since the Terracon preliminary report was issued in 2011.*

No sinkholes have occurred at the RPF site since the Terracon preliminary report was issued in 2011 (Terracon, 2011a/b). The most recent study (Boone County, 2015) shows that the project site is northeast of the nearest areas considered to have the potential for sinkholes. The most recent sinkhole occurred in May 2014 and was located on East Gans Creek Road, approximately 1.17 km (0.73 mi) to the southwest of the RPF site.

RAI 2.5-1b	*Clarify and identify measures to be taken to preclude potentially detrimental effects of sinkhole formations on the foundations in the future.*

A site-specific geotechnical investigation of the RPF will be conducted site to ensure that the area does not have the potential for sinkholes. If the investigation does identify the potential for sinkholes, the design would incorporate one of the following alternatives: (1) excavate site both vertically and horizontally to remove that potential and backfill with structural fill, or (2) install piers to bedrock to support the substructure if a sinkhole was to occur. If one of these alternatives needs to be implemented, it will be determined after the geotechnical investigation is complete, incorporated in the final RPF design, and presented in the FSAR as part of the Operating License Application.

RAI 2.5-2	*NWMI PSAR Section 2.5.2.1, "Quaternary Age Holocene Series (Qal)," states, in part, that "highly plastic clays that exhibit volume change with variations in moisture are commonly encountered near the ground surface (Terracon 2011)." This statement is repeated three times in this PSAR section. Additional information is needed to preclude adverse effects of this phenomenon on structural foundations.* *Clarify any measures to be taken to preclude adverse effects of this phenomenon on the structural foundations.*

A site-specific geotechnical investigation of the RPF site will be conducted to identify the site-specific soil characteristics. If highly plastic clays are identified at the site, the design will include excavation of the clays and then backfill with structural fill. The structural details will be developed in the final RPF design and presented in the FSAR as part of the Operating License Application.

RAI 2.5-3	*NWMI PSAR Section 2.5.3, "Onsite Soil Types," states that:* *Soils with moisture levels above their measured plastic limits may be prone to rutting and can develop unstable sub-grade conditions during general construction operations (Terracon 2011). Moderate to high plasticity clays were observed at the site. Such soils are commonly referred to as "expansive" or "swelling soils" ... Footings, floor slabs, and pavements supported on expansive soil often shift upward or downward causing possible distortions, cracking or structural damage.* *Additional information is needed on measures to prevent potential structural damage of the foundation from occurring as a result of these clays.* *Clarify any measures to be taken to preclude these potential adverse effects on the structural foundations from occurring.*

A site-specific geotechnical investigation of the RPF site will be conducted to identify the site-specific soil characteristics. If highly plastic clays are identified at the site, the design will include excavation of the clays and then backfill with structural fill. The structural details will be developed in the final RPF design and presented in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 2.5-4	*NWMI PSAR Section 2.5.4, "Seismicity," presents a listing of recorded earthquakes with a magnitude equal to or larger than 3.0 in Table 2-28, "Recorded Missouri Earthquake History," as required by NUREG-1537, Part 1, Section 2.5.3, "Seismicity." The last listed earthquake, with magnitude 4.6, occurred in 2002.* *Clarify that there were no other recorded earthquakes of magnitude 3.0 or larger between 2002 and 2016, and if there were, update the table.*

PSAR Chapter 2.0, Table 2-28, will be revised (as shown below) to incorporate earthquakes since 2002 with a magnitude over 3.0.

Table 2-28. Recorded Missouri Earthquake History (3 pages)

Date	Location	Magnitude	Recorded damage
12/16/1811 (1811–1812 series)	New Madrid Region, Missouri	7.7	Generated great waves on the Mississippi River causing major flooding, high river back cave-ins. Topographic changes affected an area of 78,000 to 130,000 km^2 (30,116 to 50,193 mi^2). Later geologic evidence indicated that the epicenter was likely in northeast Arkansas. The main shocks were felt over an area covering at least 5,180,000 km^2 (2,000,000 mi^2). Chimneys were knocked down in Cincinnati, Ohio, and bricks were reported to have fallen from chimneys in Georgia and South Carolina. The first shock was felt distinctively in Washington, D.C., 1,127 km (700 mi) away.
12/23/1812 (1811–1812 series)	New Madrid, Missouri	7.5	Second major shock more violent than the first.
2/7/1812 (1811–1812 series)	New Madrid, Missouri	7.7	Three main shocks reaching MMI of XII, the maximum on scale. Aftershocks continued to be felt for several years after the initial tremor. Historical accounts and later evidence indicate that the epicenter was close to the town of New Madrid, Missouri. This quake produced the largest liquefactions fields in the world.
1/4/1843	New Madrid, Missouri	Not listed	Cracked chimneys and walls in Memphis, Tennessee, and reportedly collapsed one building. The earth sank in some places near the town of New Madrid, Missouri, and an unverified report indicated that two hunters were drowned during the formation of a lake. The total felt area included at least 1,036,000 km^2 (400,000 mi^2).
4/24/1867	Eastern Kansas	Not listed	Reports indicated that an earthquake occurred in eastern Kansas and was felt as far eastward as Chicago, Illinois. It may have been noticeable in Columbia.
8/31/1886	Charleston, South Carolina	Not listed	An MMI of II earthquake recorded in St. Louis, Missouri, and was felt as far westward as Columbia. There were no reports of structural damage.

Table 2-28. Recorded Missouri Earthquake History (3 pages)

Date	Location	Magnitude	Recorded damage
10/31/1895	Charleston, Missouri	6.6	Largest earthquake to occur in the central Mississippi River valley since the 1811–1812 series. Structural damage and liquefaction phenomena were reported along a line from Bertrand, Missouri, in the west to Cairo, Illinois, to the east. Sand blows were observed in an area southwest of Charleston, Puxico, and Taylor, Missouri; Alton, and Cario, Illinois; Princeton, Indiana; and Paducah, Kentucky. The earthquake caused extensive damage (including downed chimneys, cracked walls, shattered windows, and broken plaster) to schools, churches, and private residences. Every building in the commercial area of Charleston was damaged. Cairo, Illinois, and Memphis, Tennessee, suffered significant damage. Near Charleston, 1.6 ha (4 acres) of ground sank and a lake formed. The shock was felt over all or portions of 24 states and in Canada. Ground shaking was recorded along the Ohio River Valley.
1903	New Madrid, Missouri	5.1	No information given.
4/9/1917	St. Genevieve/ St. Mary's Area, Missouri	Not listed	A sharp disturbance at St. Genevieve and St. Mary's, Missouri. According to the Daily Missourian, No. 187, dated April 9, 1917, the earthquake was not felt in Columbia. However, on the following day several people reported feeling the shock and attributed it to an explosion. No damage was reported in Columbia. Reportedly felt over a 518,000 km^2 (200,000 mi^2) area from Kansas to Ohio and Wisconsin to Mississippi.
5/1/1920	Missouri or Illinois	Not listed	This earthquake reportedly shook buildings across St. Louis. Two shocks were felt in Mt. Vernon, Illinois, and three were felt in Centralia, Illinois. The epicenter of this earthquake is unknown and is thought to have originated east of Columbia in Illinois. In the Evening Missourian, No. 207, dated May 1, 1920, the U.S. Weather Bureau reported that the shock was not felt in Columbia. However, in a later investigation a few people reported feeling a slight tremor.
8/19/1934	Rodney, Missouri	Listed as strong	At nearby Charleston, windows were broken and chimneys collapsed or were damaged. Similar effects were observed in Cairo, Mounds, and Mounds City, Illinois, and at Wickliffe, Kentucky. The area of destructive intensity included more than 596 km^2 (230 mi^2)
11/23/1939	Western Illinois	Not listed	An earthquake occurred near Red Bud, Illinois, and a reported MMI of II was recorded in Columbia, Missouri. The approximately distance from the epicenter to Columbia was 213 km (132 mi).
3/3/1963	Near Menorkanut, Missouri	Not listed	MMI of III was recorded in Columbia. The approximately distance from the epicenter to Columbia was 317 km (197 mi).
10/21/1965	Eastern Missouri	Not listed	MMI of V in Columbia. The approximate distance from the epicenter to Columbia was 163 km) (101 mi).

Table 2-28. Recorded Missouri Earthquake History (3 pages)

Date	Location	Magnitude	Recorded damage
11/9/1968	Wabash Valley Seismic Zone, southern Illinois	5.4	Strongest magnitude in central U.S. since the 1895 earthquake. Moderate damage to chimneys and walls at Hermann, St. Charles, St. Louis, and Sikeston, Missouri. Shaking was felt. Areas include all or portions of 23 states from Minnesota to Georgia and from Pennsylvania to Kansas, and in multi-story buildings in Boston, Massachusetts and southernmost Ontario, Canada.
1987	Wabash Valley Seismic Zone, near Olney, Richland County, SE Illinois	5.0	Chimneys and bricks fell, underground pipes were damaged, and sidewalks and streets cracked in at least four cities in Illinois, Indiana, and Kentucky. Shaking was felt in 17 states, from Pennsylvania to Kansas and from Alabama to Minnesota and southernmost Ontario, Canada.
2002	Wabash Valley Seismic Zone, Posey County, SW Indiana	4.6	Moderate earthquake caused chimney damage and cracked windows in and near Evansville, Indiana. Shaking was reported in seven states, including Missouri.
8/16/2003	20 km WNW of Alton, Missouri	3.7	Minor quake, no damage reported
5/18/2005	Missouri	3.3	Minor quake, no damage reported
7/31/2005	Missouri	3.3	Minor quake, no damage reported
6/7/2011	18 km NNW of Potosi, Missouri	3.9	Minor quake, no damage reported
9/22/2011	22 km NNE of Doniphan, Missouri	3.6	Minor quake, no damage reported
1/16/2015	15 km N of Doniphan, Missouri	3.5	Minor quake, no damage reported
10/16/2015	14 km NNW of Doniphan, Missouri	3.2	Minor quake, no damage reported
7/5/2016	6 km SW of Caruthersville, Missouri	3.0	Minor quake, no damage reported

Sources:

USGS, 2013c, "Three Centuries of Earthquakes Poster," pubs.usgs.gov/imap/i-2812/i-2812.jpg, U.S. Geological Survey, Reston, Virginia, accessed July 23, 2013.

USGS, 2002, "Earthquakes in the Central United States 1699 -2002," pubs.usgs.gov/imap/i-2812/i-2812.jpg, U.S. Geological Survey, Reston, Virginia, June 18, 2002.

MU, 2006, *Missouri University Research Reactor (MURR) Safety Analysis Report*, MU Project# 000763, University of Missouri, Columbia, Missouri, August 18, 2006.

USGS, 2016, "Search Earthquake Catalog," http://earthquake.usgs.gov/earthquakes/search/, U.S. Geological Survey, Reston, Virginia, accessed October 7, 2016.

MMI = Modified Mercalli Intensity.

No.	Request for additional information
RAI 2.5-5	*NWMI PSAR, Section 2.5.6, "Vibratory Ground Motion," states that the seismic design parameters for the proposed project are discussed in terms of the 2012 International Building Code (IBC) and associated standards. Later discussions in this section refer to the 2009 IBC and American Society of Civil Engineers (ASCE) 7-05, without any explanation. Additional information is needed to resolve this apparent discrepancy between IBC 2012 and IBC 2009.* *Provide justification for this change from IBC 2012 to IBC 2009.*

The correct reference is IBC 2012. The 2009 IBC reference callouts in PSAR Chapter 2.0 will be changed to 2012.

RAI 2.5-6	*NUREG 1537, Part 2, Section 2.5 "Geology, Seismology and Geotechnical Engineering" states that "the information on potential seismic effect should be in a form suitable for developing design basis in Chapter 3 for the SSCs, and ... this information presented should be obtained from sources of adequate credibility, and is consistent with other available data, such as data from the USGS or in the FSAR of a nearby nuclear power plant."* *NWMI PSAR Section 2.5.6, "Vibratory Ground Motion," states that "for MU facilities the 2012 IBC has been levied as the required building code. Therefore, the seismic design parameters for the proposed project are discussed in terms of the 2012 IBC and associated standards."* *However, the University of Missouri Research Reactor (MURR) and Callaway Nuclear Plant, which is in the proximity of the RPF site, adapted the seismic response spectra provided by NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and adjusted to reflect the ground acceleration response of 0.2 g.* *NWMI PSAR Section 2.5.5, "Maximum Earthquake Potential," states that Boone County would be severely impacted by a 7.6 magnitude earthquake with the epicenter on or near the New Madrid Seismic Zone, with an estimated intensity of VII at the site, as shown on Table 2-29, "Projected Earthquake Hazards for Boone County." Information is needed to justify that the RPF design and its seismic input parameters are adequate to prevent and mitigate any radiological releases below the 10 CFR Part 20 limits in the event of a postulated earthquake.*
RAI 2.5-6a	*Provide an estimated maximum ground acceleration at the site, corresponding to this intensity VII earthquake.*

The estimated maximum ground acceleration at the RPF site will meet Regulatory Guide 1.60, *Design Response Spectra for Seismic Design of Nuclear Power Plants*, free-field response spectrum anchored to a peak ground acceleration (PGA) of 0.20 g.

RAI 2.5-6b	*Justify that the RPF and its seismic parameters are designed to prevent and mitigate any radiological releases below the 10 CFR Part 20 limits in the event of an earthquake.*

PSAR Sections 3.4 and 3.5 provide design criteria and the analysis methodology for seismic events, including a safe shutdown earthquake (SSE). The seismic design of the RPF and associated IROFS will ensure the functionality and/or integrity of SSCs required to prevent radiological release below the performance requirements of 10 CFR 70.61. Additional information on the seismic requirements and evaluations of the RPF and associated IROFS will be provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 2.5-7	*NUREG-1537, Part 1, Section 2.5.5, "Vibratory Ground Motion," states that the applicant should assess the ground motion at the site from the maximum potential earthquakes associated with each tectonic province and should consider any site amplification effects. Using the results, the applicant should establish the vibratory ground motion design spectrum.* *The vibratory ground motion design spectrum was not provided in the NWMI PSAR, Section 2.5.6, "Vibratory Ground Motion." Information on the vibratory ground motion is needed to assess the adequacy of the ground motion at the RPF site.* *Provide the vibratory ground motion design spectrum in PSAR Section 2.5.6.*

As stated in the response to RAI 2.5-6a, the estimated maximum ground acceleration at the RPF site will meet Regulatory Guide 1.60 free-field response spectrum anchored to PGA of 0.20 g.

No.	Request for additional information
RAI 2.5-8	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.* *NWMI PSAR Section 2.5.6, states that the Boone County site is a soil Site Class D site. Later in Section 3.4.1.1, "Design Response Spectra" states that the Phase 1 Assessment (Terracon, 2011 a/b) the site is referred to as Class C.* *Clarify if the RPF site is a soil Site Class C or D, and correct any discrepancies.*

The seismic soil classification for the RPF site is Class C. Reference to the Boone County site as being soil Class D in PSAR Chapter 2.0, Section 2.5.6 will be changed to Class C.

No.	Request for additional information
RAI 2.5-9	*NUREG 1537, Part 1, Section 2.5.7, "Liquefaction Potential," pertains to the evaluation of soil structure, and states, in part, that:* *If the foundation materials at the site adjacent to and under safety-related structures are saturated soils or soils that have a potential for becoming saturated, the applicant should prepare an appropriate state-of-the-art analysis of the potential for liquefaction at the site. The applicant should also determine the method of analysis on the basis of actual site conditions, the properties of the facilities, and the earthquake and seismic design requirements for the protection of the public.* *NUREG-1537, Part 2, Section 2.5, "Geology, Seismology and Geotechnical Engineering," instructs the review to confirm that the information on the geologic features and the potential seismic activity at the site have been provided in sufficient detail and in a form to be integrated acceptably into design bases for structures, systems and operating characteristics of the facility.* *NWMI PSAR Section 2.5.8, "Liquefaction Potential," provides information based on preliminary investigations of the RPF site by Terracon, and concludes that the available data is insufficient and contradictory and the liquefaction potential cannot be conclusively determined. It also states that additional geotechnical analysis will be conducted at the RPF site to determine the liquefaction potential of the soils on site. Information is needed to understand if this is part of ongoing research and development (10 CFR 50.34(8)) or will be provided in the FSAR.* *Provide the timeframe for the performance of these additional state of the-art geotechnical investigations and analyses.*

The geotechnical investigations and analyses of the RPF site will be completed in the first quarter of 2017.

CHAPTER 3.0 – DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

No.	Request for additional information
	Section 3.2 – Meteorological Damage
(Applies to RAIs 3.2-1 through 3.2-4)	*NUREG-1537, Part 1, Section 3.2, "Meteorological Damage," states that:* *The design criteria should provide reasonable assurance that potential meteorological damage would not significantly affect designed structures, systems and components (i.e., they would continue to perform necessary operational and safety functions).* *NUREG-1537, Part 2, Section 3.2, "Meteorological Damage," instructs the review to:* *Examine the description of the site meteorology to ensure that all structures, systems and components that could suffer meteorological damage are considered in this section of the SAR... The reviewer should compare design specifications for structures, systems and components with the functional requirements and capability to retain function throughout the predicted meteorological conditions and... the design to protect against meteorological damage provides reasonable assurance that the facility... will perform the safety functions discussed in this SAR... and to protect the health and safety of the public from radioactive materials and radioactive exposure.*
RAI 3.2-1	*NWMI PSAR, Section 3.2.3.1.3, "Live Loads," and Table 3-13, "Floor Live Loads," listed that the uniform and concentrated loads for the hot cell roof and cover block laydown are to be determined.* *These live loads may have effect on the global response of the structure and on the design and sizing of the affected structural members.* *If the actual magnitude of these loads is not available at the preliminary design stages, confirm that either conservative estimated loads are used to account for their effects, or the design will be revisited when the loads are available at a later stage, before the designs are finalized.*

During the structural analysis, unknown loads will have a conservative value assumed and marked with "(HOLD)." As the design matures, the actual values will be inserted in the analysis and the HOLDs removed. Final design media cannot be issued if there are HOLDs identified. The facility live loads will be established during the completion of the final facility design and provided in the FSAR as part of the Operating License Application.

RAI 3.2-2	*NWMI PSAR Section 3.1.3, "U.S. Nuclear Regulatory Commission," states that:* *Table 3-3 lists the NRC design inputs for the RPF identified in NWMI DRD-2013-030. The RPF system design descriptions identify the specific requirements for that system produced by each applicable reference." PSAR Table 3-3, "Relevant U.S. Nuclear Regulatory Commission Guidance," includes Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, 2007" (Revision 1).* *PSAR Section 3.1.7, "Codes and Standards," states that:* *Table 3 7 lists design inputs for the RPF identified in NWMI-DRD-2013-030. The RPF system design descriptions identify the specific requirements for that system produced by each applicable reference.* *PSAR Table 3.7, "Design Codes and Standards," includes ANSI/ANS-2.3, "Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics as [at] Nuclear Facility Sites, 2011."* *PSAR Section 3.2.4.2, "Tornado Loading," states that:* *To date, the NRC has not endorsed the 2011 revision of American National Standards Institute (ANSI)/American Nuclear Society (ANS) 2.3, Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites, for use in design of power reactors. However, considering that the RPF is a production facility as opposed to a power reactor, and the wind field characteristics given in ANSI/ANS 2.3 are more consistent with the target performance goals, the tornado load requirements for the RPF design are based on NUREG-1520 and ANSI/ANS 2.3.*

No.	Request for additional information
	However, the Forward to ANSI/ANS 2.3-2011 states, that: *This standard is a revision to ANSI/ANS 2.3-1983, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites." The revision of the 1983 standard began in May of 2005. In this revision, the scope of the standard was expanded to include hurricane wind characteristics. A change to the Fujita damage scale as a function of wind velocities, adopted in 2007 by the National Weather Service, resulted in the wind speeds associated with the Fujita damage scale being replaced by the Enhanced Fujita Scale as shown in Table 1. Also included in the scope expansion is the applicability of this standard to all nuclear facility sites, not just nuclear power plant sites.* *ANSI/ANS 2.3-2011 applies the same criteria to nuclear facilities whether a production facility or a power reactor.* *PSAR Section 3.2.4.2, "Tornado Loading," describes the design basis for tornado winds, atmospheric pressure drop, and tornado generated missile impact effects.* *The NRC staff identified that the information provided in Tables 3-16, "Tornado Wind Field Characteristics," and 3-17, "Tornado Wind Driven Missile Criteria," differs from the data provided in Regulatory Guide 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1, for the tornado wind velocities, atmospheric pressure drop, tornado-generated missile velocities, and the altitude of automobile missile strikes.* *The cited ANSI/ANS 2.3-2011, "Estimating Tornado, Hurricane and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites," has not yet been endorsed by the NRC, and the referenced NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications," states that the tornado occurrence probability be quantified by the applicant. NUREG-1520, Part 3 Appendix D states, "...depending on the geographic location of the facility the effects of a tornado with an annual exceedance probability of 10-5 or greater may need to be considered."* *Since the PSAR identified Regulatory Guide 1.76, Revision 1, as a design input to the RPF and because ANSI/ANS 2.3-2011 has not been accepted by the NRC, justify the use of tornado effect parameters which are less conservative than those provided in Regulatory Guide 1.76, Revision 1.*

The PSAR, including Section 3.2.4.2 and all appropriate supporting documentation, will be modified to state that Regulatory Guide 1.76, *Design Basis Tornado and Tornado Missiles for Nuclear Power Plants*, will be the basis for tornado wind loads and wind-generated missiles.

RAI 3.2-3	*NWMI PSAR Section 3.2.3.1.1, "Dead Loads," states, in part, that the "dead loads consist of the weight of all materials of construction ... [and] also consist of the weight of fixed equipment, including the weight of cranes." It is not apparent that the weight and loading from other sub-systems were included in the dead load.* *This information is needed to confirm that adequate uniform loading is included in the dead load to account for the piping, cable trays/cable, conduits, heating, ventilation and air conditioning (HVAC) duct and appurtenances, tubing, etc. weights in the global structural analysis and the design and sizing of the structural elements.* *Confirm that adequate uniform loading is included in the dead load to account for the piping and other sub-systems (i.e., cable trays, conduits, HVAC ductwork and components, tubing) weight and loadings in the design and sizing of the structural elements.*

The density of all interconnections (e.g., heating, ventilation, and air conditioning [HVAC] ductwork, conduits, cable trays, and piping) between equipment will be conservatively estimated and included in the final design for dead load for fixtures attached to ceilings or anchored to floors in the RPF. This information will be provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 3.3 – Water Damage
RAI 3.3-1	*NUREG-1537, Part 1, Section 3.3, "Water Damage," states, in part, that:* *... the applicant should specifically describe the proposed site and facility designs to protect against water damage of the structures, systems and components assumed to function in the SAR. This should include ... and (3) the impact on equipment, such as fans, motors, and valves resulting from degradation of the electromechanical function due to water.* *NUREG-1537, Part 1, Section 3.3, "Water Damage" states, in part, that:* *... (2) the impact on systems resulting from instrumentation and control electrical and or mechanical malfunction due to water, and (3) the impact on equipment, such as fans, motors and valves resulting from degradation of electromechanical function due to water.* *NUREG-1537, Part 2, Section 3.3, "Water Damage," states that the areas of review include the design and design bases for all structures, systems, and components that could be affected by predicted hydrological conditions at site, and states that:* *The design criteria and designs should provide reasonable assurance that structures, systems and components would continue to perform required safety function under water damage conditions. For the design the applicant should use local building codes, as applicable to help ensure that water damage to structures, systems and components at the facility site would not cause or allow uncontrolled release of radioactive material.* *NWMI PSAR Sections 3.3.1, "Flood Protection," and 3.3.1.1, "Flood Protection Measures for Structures, Systems, and Components," refer to PSAR Section 2.5.3, "Onsite Soil Types," for additional information on flood protection measures. Section 2.5.3 of the PSAR does not contain this information.* *Provide the correct reference in the PSAR for additional detail for flood protection measures.*

PSAR Sections 3.3.1 and 3.3.1.1 will be modified to point to PSAR Section 2.4.3 (instead of PSAR Section 2.5.3) for flood information.

RAI 3.3-2	*NUREG-1537, Part 1, Section 3.3, states, in part, that:* *...(2) the impact on systems resulting from instrumentation and control electrical or mechanical malfunction due to water, and (3) the impact on equipment, such as fans, motors, and valves, resulting from degradation of the electromechanical function due to water.* *NUREG-1537, Part 2, Section 3.3, states, in part, that:* *the design criteria and designs should provide reasonable assurance that structures, systems, and components would continue to perform required safety functions under water damage conditions for the design the applicant should use local building codes, as applicable, to help ensure that water damage to structures, systems, and components at the facility site would not cause unsafe reactor operation, would not prevent safe reactor shutdown, and would not cause or allow uncontrolled release of radioactive material.* *NWMI PSAR Section 3.3 discusses water damage and Sections 3.3.1.3 and 3.3.1.4.1 deal with flooding due to malfunction of the Fire Protection System, but the NRC staff could not find a discussion of the effects of discharge of the Fire Protection System on structures, systems and components.* *This information is needed to determine the adequacy of the measures to be taken for the protection of sensitive safety-related components under the effects of inadvertent discharge of the Fire Protection System sprinklers or rupture of the non-seismic fire protection piping as a result of the postulated seismic event.*

No.	Request for additional information
RAI 3.3-2 (cont)	*Provide information discussing measures to be taken for the protection of sensitive safety-related equipment under the effects of inadvertent discharge of the Fire Protection System water.*

Design of fire suppression systems using water (e.g., automatic sprinklers, hose stations) includes elements such as the grading and channeling of floors, raising of equipment mounts above floors, shelving and floor drains, and other passive means. These features will ensure sufficient capacity for gravity-driven collection and drainage of the maximum water discharge rate and duration to avoid localized flooding and resulting water damage to equipment within the area. In addition, particularly sensitive systems and components, whether electrical, optical, mechanical and/or chemical, are typically protected within enclosures designed for the anticipated adverse environmental conditions resulting from these types of water discharges. If critical for safety, these water-sensitive systems and components will be installed within the appropriate severe environment-rated enclosures in accordance with the relevant industry standard(s) (e.g., NEMA enclosure standards).

Selection of specific fire suppression systems for facility locations will be guided by the recommendations offered in relevant industry standards (e.g., NFPA 801, *Standard for Fire Protection for Facilities Handling Radioactive Materials*) and will depend on the level of fire hazards at those locations, as determined from the final facility and process systems designs. These final detailed designs will include any facility design elements and sensitive equipment protection measures deemed necessary for addressing the maximum inadvertent rate and duration of water discharges from the fire protection systems. The final comprehensive facility design, along with commitments to design codes, standards, and other referenced documents (including any exceptions or exemptions to the identified requirements), will be identified and provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 3.4 – Seismic Damage
RAI 3.4-1	*NUREG-1537, Part 1, Section 3.4, "Seismic Damage," states, in part, that the applicant should specify and describe structures, systems and components that are required to maintain the necessary safety function if a seismic event should occur. The reactor facility seismic design should provide reasonable assurance that the reactor could be shut down and maintained in a safe condition. To verify that seismic design functions are met, the applicant should give the bases for technical specifications necessary to ensure operability, testing, and inspection of associated systems, including instrumentation and control portions, as applicable.* *NUREG-1537, Part 2, Section 3.4, "Seismic Damage," states that the review should include the designs and design bases of structures, systems, and components that are required to maintain function in case of a seismic event at the facility site. The finding required is that the facility design should provide reasonable assurance that the RPF can be shut down and maintained in a safe condition.* *NUREG-1537, Part 2, Section 2.5, "Geology, Seismology and Geotechnical Engineering," states, in part, that the information has been obtained from sources of adequate credibility and is consistent with other available data, such as data from the USGS or in the FSAR of a nearby nuclear power plant.* *NWMI PSAR Section 3.4.1.1, "Design Response Spectra Safe-Shutdown Earthquake," states that:* *The safe-shutdown earthquake ... for the RFP facility is specified as risk-targeted maximum considered earthquake (MCE_R), as determined in accordance with ASCE 7 and the Federal Emergency Management Agency (FEMA) P-753, NEHRP Recommended Seismic Provisions for New Buildings and Other Structures. The MCE_R for this site is governed by the by the probabilistic maximum-considered earthquake ground shaking, which has an annual frequency of exceedance of $4x10^{-4}$.* *MURR and Callaway Nuclear Plant which is in the proximity of the RPF site, adapted the seismic response spectra provided by NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and adjusted to reflect the ground acceleration response of 0.2 g.*

No.	Request for additional information
RAI 3.4-1 (cont)	*Provide justification for not using Regulatory Guide 1.60 spectra adjusted to reflect maximum ground acceleration of 0.2 g for the safe-shutdown earthquake.*

As stated in the response to RAI 2.5-6a, the SSE design basis for the RPF will be the Regulatory Guide 1.60 spectrum anchored to a PGA of 0.20 g. Seismic loads for the design of IROFS will be derived from in-structure floor response spectra generated by the finite element model of the building structure, as outlined in the response to RAI 3.4-7.

(Applies to RAIs 3.4-2 through 3.4-9)	*NUREG-1537, Part 2, Section 2.5, "Geology, Seismology and Geotechnical Engineering," states, in part, that "the information on potential seismic effects should be in a form suitable for developing design bases in Chapter 3 for structures, systems and components."*
RAI 3.4-2	*NWMI PSAR Section 2.5.6, "Vibratory Ground Motion," states that "The Boone County site is a soil site Class D. When modified for a Class D site... the site coefficients Fa and Fv, Sms, and Sml values of 0.341 and 0.223 respectively (Fa=1.6 and Fv=2.4) are obtained."* *PSAR Section 3.4.1.1, "Design Response Spectra," in Table 3-22, "Safe Shutdown Earthquake Criteria," contains different parameters based on soil site Class C.*
RAI 3.4-2a	*Resolve the inconsistency and update Table 3-22 for site Class D properties, and demonstrate that a correct value was used in the evaluations.*

The NRC has recommended using Regulatory Guide 1.60 for radioisotopes production facilities (e.g., 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities"). NWMI will use a spectrum anchored to 0.20 g PGA for the RPF design basis. Regulatory Guide 1.60 is not indexed to any specific soil type, with its frequency content sufficiently broad to cover all soil types. Therefore, soil type for the RPF will not be a parameter used to determine the RPF's design response spectra. The composition of soil in which the RPF is embedded will be included in the soil-structure-interaction analysis as part of the building response analysis. This information will be provided in the FSAR as part of Operating License Application.

Table 3-22 will be deleted from PSAR Section 2.5.6. In addition, Regulatory Guide 1.60 will be added to PSAR Section 3.4.1.1 for the determination of the RPF design response spectra.

The seismic soil classification for the RPF site is Class C. Thus, the reference to the Boone County site as being soil Class D in PSAR Section 2.5.6 will be changed to Class C.

PSAR Section 2.5.6 will be modified to reflect the above information.

RAI 3.4-2b	*Provide justification for the statement that, "This safe-shutdown earthquake spectrum is 5% damped which, per ASCE 43 is appropriate for the design of the structural systems to be used for the Radioisotope Production Facility (RPF)."*

Response spectra corresponding to the recommended damping values of Regulatory Guide 1.61, *Damping Values for Seismic Design of Nuclear Power Plants*, will be used to derive seismic loads. Damping varies depending on the type of SSC. Structural damping will follow the recommendations of Regulatory Guide 1.61, which range from about 3 to 7 percent. Plotting response spectra at 5 percent damping for purposes of illustration is a convention within the nuclear industry, but for analysis loads, damping will vary depending on the earthquake level (operating basis earthquake [OBE] or SSE) and the type of SSC.

No.	Request for additional information
RAI 3.4-3	*NWMI PSAR Section 3.4.1.1, "Design Response Spectra Safe Shutdown Earthquake," Figure 3.1, "Safe-Shutdown Earthquake Response Spectrum," presents the suggested spectrum at five-percent damping.* *Provide a discussion that justifies that the identified response spectrum after the adjustments made for Class D site (as discussed in RAI 3.4-2) is adequately representing the ground response accelerations for an Intensity VII earthquake, established in PSAR Section 2.5.5, Table 2-29, "Projected Earthquake Hazards for Boone County," for the RPF site.*

The RPF building structure and IROFS analysis will be based on the Regulatory Guide 1.60 ground motion spectrum indexed to a PGA of 0.20 g. This PGA matches that of the University of Missouri Research Reactor (MURR) and the Calloway Nuclear Generating Station, which both are within 80.5 km (50 mi) of the RPF, as suggested by the NRC staff during the November 10, 2016 Public Meeting. The analysis procedure develops ground motion acceleration time histories that match or exceed the Regulatory Guide 1.60 spectrum as input to the building finite element model. Structural damping will follow the recommendations of Regulatory Guide 1.61, which range from about 3 to 7 percent.

The reference to a Class D site is a term used in IBC 2012, and will not be the basis for the RPF seismic design. The composition of soil in which the RPF is embedded will be included in the soil-structure-interaction analysis as part of the building response analysis.

Table 3-22 will be deleted from PSAR Section 2.5.6. In addition, Regulatory Guide 1.60 will be added to PSAR Section 3.4.1.1 for the determination of the RPF design response spectra.

The seismic soil classification for the RPF site is Class C. Thus, the reference to the Boone County site as being soil Class D in PSAR Section 2.5.6 will be changed to Class C.

PSAR Section 2.5.6 will be modified to reflect the above information.

RAI 3.4-4	*NWMI PSAR Section 3.4.1.2.1, "Equivalent-Static Seismic Analysis," under "Direction of Seismic Loading," states that "the seismic forces are applied independently in two orthogonal horizontal directions and the effects combined using the 100/30 rule. Vertical seismic effects will be accounted for by the use of seismic load combinations set forth in ASCE 7, Sections 12.4.2.3 and 12.4.3.2."*
RAI 3.4-4a	*Provide a discussion that justifies why the simultaneous application of three directional earthquake effects are not considered.*

Design of IROFS will consider seismic loads in all three directions using a combination of square-root-of-the-sum-of-squared or 10/40/40 methodologies. The 10/40/40 methodology will be used in the development of the final RPF design and in the FSAR as part of the Operating License Application.

RAI 3.4-4b	*Provide a discussion that justifies why the effects are not combined per Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," using either the square-root-of-sum-of-the squares, or 100-40-40 methodology.*

Design of IROFS will consider seismic loads in all three directions using a combination of square-root-of-the-sum-of-squared or 10/40/40 methodologies. The 10/40/40 methodology will be used in the development of the final RPF design and in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 3.4-5	*NWMI PSAR Section 3.4.1.1, "Design Response Spectra – Soil-Structure Interaction and Dynamic Soil Pressures," states that a fixed base model is expected to give conservative results in comparison to accounting for the effects of soil-structure interaction.* *However, for a relatively short RPF structure with large plan dimensions, a fixed base model may not accurately represent the dynamic characteristics and the structural response, where the rocking mode may be predominant and significant.* *Provide additional justification for the use of fixed base mathematical model for this relatively short facility with a relatively large footprint, where the rocking mode may be predominant in facility response.*

The analysis of the RPF building structure to the SSE will include the effects of a soil-structure interaction rather than the assumption of a fixed base.

RAI 3.4-6	*NWMI PSAR Section 3.4.2, "Seismic Qualification of Subsystems and Equipment," describes various qualification methods in PSAR Sections 3.4.2.1, "Qualification by Analysis," and 3.4.2.2, "Qualification by Testing."* *Many components whose continued operability/functionality during and after the postulated seismic events cannot be demonstrated by analytical means, require actual shake-table (or other dynamic) testing to be seismically qualified.* *Clarify if NWMI's purchase specifications will specify the qualification method to be used, or will it be left to the vendor's discretion.*

NWMI will define specific acceptable qualification methods in the procurement packages to demonstrate seismic qualifications. Seismic qualification of IROFS will include three options of: (1) calculations and verification that the main structural components of the SSC can withstand the seismic loads derived from the in-structure floor response spectra at the damping value derived from Regulatory Guide 1.61, (2) reference to available shake table testing that demonstrates the seismic capacity of the SSC or of multiple similar items, and (3) demonstration of the seismic capacity through the performance of the type of SSC in actual earthquakes.

RAI 3.4-7	*NWMI PSAR Section 3.4.2.1.2, "Dynamic Analysis," describes various methods for seismic qualification of subsystems and equipment and cites ASCE 4, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary, 2000" and the NRC Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components.* *ASCE 4 discusses Time History Method and Direct-Spectra-to-Spectra method for the development of in-structure response spectra, while Regulatory Guide 1.122 only covers Time History method.* *Clarify which method of in-structure response development is used for the RPF, and if it is the Time History Method which option of ASCE 4 will be used.*

In-structure floor response spectra will be developed through a finite element model of the RPF building using an artificial time history that matches or envelops the Regulatory Guide 1.60 spectrum at PGA = 0.20 g.

RAI 3.4-8	*NWMI PSAR Section 3.4.2.2, "Qualification by Testing," states that subsystems and equipment not relied on for nuclear safety but designed as a component of a seismic system per IBC 2012, Chapter 17, will be required to be seismically qualified in accordance with ICC-ES-AC156, "Acceptance Criteria for Seismic Certification by Shake-Table Testing of Nonstructural Components.* *However, it is not stated what measures are to be taken to preclude gross failure of the Seismic Category II or Non-Seismic SSCs during postulated seismic events from damaging the safety-related SSCs (generally referred to as Seismic II/I considerations), or detrimental "banging" effects on safety-related SSCs due to the excessive displacements caused by the postulated earthquake.*

No.	Request for additional information
RAI 3.4-8a	*Confirm that the Seismic II/I systems and components whose gross failure may impact the functionality of safety systems and components, are designed to remain stable during and after the postulated earthquake effects.*

The capacity of the standard support design for overhead fixtures mounted above RPF IROFS will be checked to ensure that the supports can withstand the seismic loads derived from the floor spectra (e.g., remain stable during and after postulated earthquake effects) of the attachment floor slab. This information will be provided in the FSAR as part of the Operating License Application.

RAI 3.4-8b	*Confirm that a rattle space is established to ensure that the Seismic II/I systems and components cannot damage the safety-related items due to their potentially excessive deformations during the seismic event.*

The RPF seismic design will include a check to ensure that pounding or sway impact will not occur between adjacent fixtures (e.g., rattle space). Estimates of the maximum displacement of any fixture can be derived from the appropriate floor response spectrum and an estimate of the fixture's lowest response frequency. This information will be provided in the FSAR as part of the Operating License Application.

RAI 3.4-9	*NWMI PSAR Section 3.4.3, "Seismic Instrumentation," provides a description of the seismic instrumentation provided for the RPF.* *However, it does not specify if the seismic instrumentation is considered to be safety-related Seismic Category I, and is purchased as Seismically Qualified system to be able to fulfill the purpose.* *Clarify whether the seismic instrumentation will be a safety-related Seismic Category I installation.*

Seismic instrumentation for the RPF site is not an IROFS; it provides no safety function and is therefore not "safety-related." Although the seismic recorders have no safety function, they must be designed to withstand any credible level of shaking to ensure that the ground motion would be recorded in the highly unlikely event of an earthquake. This capability requires verification of adequate capacity from the manufacturer (e.g., prior shake table tests of their product line), provision of adequate anchorage (e.g., manufacturer-provided anchor specifications to ensure accurate recordings), and a check for seismic interaction hazards such as water spray or falling fixtures. With these design features, the instrumentation would be treated as if it were safety-related QL-2. Additional information on seismic instruction will be provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 3.5 – Systems and Components
(Applies to RAIs 3.5-2 and 3.5-3)	*The definition for IROFS is provided in 10 CFR 70.4. The definition states:* *Items relied on for safety mean structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in § 70.61 or to mitigate their potential consequences. This does not limit the licensee from identifying additional structures, systems, equipment, components, or activities of personnel (i.e., beyond those in the minimum set necessary for compliance with the performance requirements) as items relied on for safety.*

No.	Request for additional information
RAI 3.5-2	*NWMI PSAR, Section 3.5.1.3.1, states that a safety-related SSCs must meet at least one of the following criteria. If they do not, they are considered non-safety-related (NSR). The safety-related criteria are:* • *Ensure and safeguard the integrity of the primary system (RPF) boundary* • *Provide the capability to prevent or mitigate the consequences of accidents that could result in exposures to workers, the public, or environment that exceed 10 CFR 20 guidelines* • *Ensure the potential for an inadvertent criticality accident is not credible* • *Ensure that acute chemical exposures to an individual produced from licensed materials or hazardous chemicals will not lead to (1) Irreversible or other serious, long-lasting health effects to workers (2) Mild transient health effects to individuals located outside the RPF RCA* • *Prevent intake of 30 milligrams (mg) or more of uranium in soluble form by any individual located outside the RPF RCA* *The PSAR proposes a five-part definition of safety-related SSCs, modifies the 10 CFR 50.2 definition of safety-related SSCs to include performance requirements for SSCs important to safety in the RPF. While four of the five parts of this definition are performance-based, the third part of this definition (i.e., "Ensure the potential for an inadvertent criticality accident is not credible") is not performance-based. The regulations in 10 CFR Part 70 are not a requirement for a production facility, however the NRC staff finds that their use as an accident consequence and likelihood criteria may be found acceptable.* *However, 10 CFR 70.61 provides performance requirements for subcriticality. 10 CFR 70.61(d) states:* *In addition to complying with paragraphs (b) and (c) of this section, the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety. Preventive controls and measures must be the primary means of protection against nuclear criticality accidents.* *Provide a performance-based criterion for an inadvertent criticality accident utilizing the criteria of 10 CFR 70.61(d) or explain why the 10 CFR 70.61 performance criteria is not used.*
NWMI is using the 10 CFR 70.61 performance requirement for subcriticality. PSAR Section 3.5.1.3 was revised, the bullets removed, and 10 CFR 70.61 performance requirements/criteria are referenced.	
RAI 3.5-3	*NWMI PSAR Section 3.5.1.3.3, "Quality Group Classifications for Structures, Systems, and Components," discusses the classification of SSCs. In that discussion, the applicant states that there are three QA Levels.* • *QA Level 1 (QL-1) is applied to safety-related SSCs and "implements the full measure of NWMI's QAPP [Quality Assurance Program Plan]."* • *QA Level 2 (QL-2) "includes the quality activities performed by NWMI, generally on a continuing basis, that are applied to ensure that the items that are not QA Level 1 are available and reliable to perform their safety functions (when needed). These quality activities include configuration management, maintenance, training and qualifications, procedures, assessments, incident investigations, records management, and other QA elements."* • *QA Level 3 (QL-3) is applied to NSR SSCs and is "controlled in accordance with standard commercial practices."* *While QL-1 and QL-3 clearly state what SSCs are assigned to these levels and the level of quality assurance applied, this is unclear for QL-2.* *From the information provided, it can be inferred that QL-2 does not implement the full measure of NWMI's QAPP and that it is not controlled in accordance with standard commercial practices. Additional information is needed to understand application of QAPP to QL-2.*

No.	Request for additional information
RAI 3.5-3a	*Define and provide the basis for the difference between QL-1 and QL-2.*

NWMI has revised its Quality Assurance (QA) Plan to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 was modified to reflect the changes in the quality level definitions (Attachment A). The basis for the difference in QL-1 and QL-2 is a graded approach to quality, by which the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance. The graded approach permits the implementing organization to focus resources on those activities that are deemed, by qualitative analysis, to reduce the associated risks and hazards. The activities and tasks are performed in accordance with approved implementing procedures.

The graded approach to quality is a process by which the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance. A graded approach permits the implementing organization to focus resources on those activities that are deemed, by qualitative analysis, to reduce the associated risks and hazards.

Activities and tasks are performed in accordance with the quality level definitions.

- Quality Level 1 will be applied to IROFS (SSCs and activities). IROFS are QL-1 items in which failure or malfunction could directly result in a condition that adversely affects workers, the public, and/or environment, as described in 10 CFR 70.61.
- Quality Level 2 will be applied to safety SSCs that are non-QL-1 SSCs. Some of the required characteristics may be examined less rigorously than for QL-1 items.
- Quality Level 3 items include those items that are not classified as QL-1 or QL-2. QL-3 items are controlled in accordance with standard commercial practices.

RAI 3.5-3b	*If two SSCs (i.e., pipe, valve, tank, heat exchanger, etc.) must meet the same performance characteristics, but one SSC is governed by QL-1 and the other by QL-2, describe how they will they be physically different.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. For two SSCs with the same performance requirements, NWMI would not expect the components to be materially different. However, the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance; therefore, a QL-2 SSC may have different verification requirements.

RAI 3.5-3c	*PSAR Table 3-25, "System Safety and Seismic Classification and Associated Quality Level Group," lists systems that are NSR, Seismic Category II, and QL-2. This infers that only NSR Seismic Category II SSCs are QL-2. If this is the case, state this explicitly, and if this is not the case provide examples of SSCs that are considered QL-2 that are not Seismic Category II.*

PSAR Chapter 3.0, Table 3-25, has been revised and is provided in Attachment A. The table was modified to match the changes in the NWMI QA Plan. A room continuous air monitor in the operating gallery is an example of a QL-2 item that is not Seismic Category II.

RAI 3.5-3d	*For QL-1 SSCs, clarify whether their associated configuration management, maintenance, training and qualifications, procedures, assessments, incident investigations, records management, etc. are considered QL-1 or QL-2. If considered QL-2, provide justification why they are not controlled under the full measure of NWMI's QAPP.*

QL-1 SSCs are controlled to the full measure of NWMI's QA Plan. The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 was modified to reflect the changes in the quality level definitions (Attachment A).

No.	Request for additional information
RAI 3.5-4	*NUREG-1537, Part 2, Section 3.5, "Systems and Components," Acceptance Criteria, states, in part, that the design criteria should include response to transient and potential accident conditions analyzed in the safety analysis report.* *NWMI PSAR Section 3.5.1.3.4, "Seismic Classification for Structures, Systems, and Components," states that Seismic Category I (C-I) applies to safety-related SSCs, that Seismic Category II (C-II) applies to those NSR SSCs whose structural failure during a safe shutdown earthquake could degrade the function of SR SSCs or impact the main control room, and that SSCs not classified C-I or C-II are classified non-seismic (NS). Since it is not clear what QL-2 encompasses other than the statement that they are not controlled under the "full measure of NWMI's QAPP," clarification is needed.*
RAI 3.5-4a	*PSAR Table 3-25, "System Safety and Seismic Classification and Associated Quality Level Group" infers that NSR C-II SSCs are automatically QL-2. Provide the rationale for that QA classification versus the QL-1 classification.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Chapter 3.0, Table 3-25, was updated to reflect the changes in the quality level definitions (see response to RAI 3.5-3c).

RAI 3.5-4b	*Clarify if any NSR C-II SSCs are considered QL-1 or QL-3.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 was revised to change the definition of nonsafety-related SSCs, and PSAR Chapter 3.0, Table 3-25, was modified (see response to RAI 3.5-3c). However, nonsafety-related SSCs should not be QL-1 even if the SSCs are Seismic Category II (C-II) (see Attachment A).

RAI 3.5-4c	*Explain the differences in C-I acceptance criteria under QL-1 and the C II acceptance criteria under QL-2.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 was revised to change the definitions and Table 3-25 was modified (see response to RAI 3.5-3c). The differences in acceptance criteria for QL-1 and QL-2 is a graded approach to quality, by which the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance. The main difference in the acceptance criteria for the Seismic Category I (C-I) and C-II classification is that C-I has to function after an SSE, while C-II has to maintain its integrity (see Attachment A).

RAI 3.5-4d	*Since C-II SSCs need only maintain structural integrity, explain if the structural integrity acceptance criteria for C-I SSCs and C-II SSC will be the same or different. If different, provide the basis for the difference.*

While the structural integrity performance requirement should be the same, the acceptance criteria for C-I and C-II SSCs will be determined by the quality level of the SSC and may have different verification requirements.

(Applies to RAIs 3.5-5 and 3.5-6)	*In the ISG Augmenting NUREG-1573, Parts 1 and 2, the NRC staff stated, in part, that "addressing the baseline design criteria and defense in depth practices in 10 CFR 70.64 is an acceptable way of demonstrating adequate safety of structures, systems, and components in the design of a radioisotope production facility."*

No.	Request for additional information
RAI 3.5-5	*NWMI PSAR Section 3.5.2.2, "Classification of Systems and Components Important to Safety," discusses SSCs that are considered important to safety. The criteria for important to safety is the five criteria for safety-related SSCs presented in Section 3.5.1.3.1, "Safety-Related Structures, Systems, and Components," and an additional criterion that states:* • *Prevent degradation of function and/or performance of any safety-related SSC.* *Thus, important to safety encompasses safety-related SSCs and other SSCs that could affect the function/performance of safety-related SSCs. However, NWMI does not provide a classification for important-to-safety SSCs, just the classifications of safety-related and NSR. PSAR Table 3-25, "System Safety and Seismic Classification and Associated Quality Level Group" associated with PSAR Section 3.5.2.2 delineates systems that are NSR, but also C-II and QL-2. This implies in Section 3.5.2.2 that NSR C-II SSCs are considered important to safety because of their C-II classification. In addition, the term, "important to safety," is not used in Section 3.5.1.3.4, "Seismic Classification for Structures, Systems, and Components," where the C-I, C-II and NS classifications are discussed.*
RAI 3.5-5a	*Clarify if NSR C-II SSCs are important to safety.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 has been revised to change the definition of nonsafety-related SSCs, and Table 3-25 was modified (see response to RAI 3.5-3c). QL-2 SSCs are safety-related items that are not QL-1 (see Attachment A).

RAI 3.5-5b	*Provide examples of NSR NS SSCs that are important to safety.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 has been revised to change the definition of nonsafety-related SSCs, and Table 3-25 was modified (see response to RAI 3.5-3c). QL-2 SSCs are safety-related items that are not QL-1 (see Attachment A).

RAI 3.5-5c	*Clarify if all NSR important-to-safety SSCs are QL-2.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 has been revised to change the definition of nonsafety-related SSCs, and Table 3-25 has been modified (see response to RAI 3.5-3c). QL-2 SSCs are safety-related items that are not QL-1 (see Attachment A).

RAI 3.5-5d	*Provide the basis for not designating NSR SSCs important to safety QL-1.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. PSAR Section 3.5.1.3 has been revised to change the definition of nonsafety-related SSCs, and Table 3-25 has been modified (see response to RAI 3.5-3c). QL-2 SSCs are safety-related items that are not QL-1 (see Attachment A).

RAI 3.5-6	*NUREG-1537, Part 1, Chapter 9, "Auxiliary Systems," states that the applicant should include the design bases for each auxiliary system.* *In cross-referencing the design bases in NWMI PSAR Chapter 9.0, "Auxiliary Systems," with those in PSAR Chapter 3.0, "Design of Structures, Systems, and Components," an apparent discrepancy was identified.* *PSAR Section 3.5.2, "Radioisotope Production Facility," states that systems and components within the RPF are presented in [PSAR] Section 3.4.1; whereas PSAR Section 3.4.1, "Seismic Input," discusses seismic design considerations. Since the higher tier Section PSAR 3.5 presents RPF systems and components, this statement is potentially incorrect.* *Provide additional information to explain and/or correct this apparent discrepancy in PSAR Section 3.5.2 that states systems and components within the RPF are presented in PSAR Section 3.4.1.*

The cross-reference to PSAR Section 3.4.1 was a typographical error. The first sentence in PSAR Section 3.5.2, which pointed to PSAR Section 3.4.1 will be changed to PSAR Section 3.5.1 (see Attachment A).

No.	Request for additional information
RAI 3.5-7	*Subparagraph 50.35(a)(1) of 10 CFR Part 50 requires that the applicant describe the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and identifies major features or components incorporated therein for the protection of the health and safety of the public.* *The ISG Augmenting NUREG-1537, Part 2, Section 3.5b, "Radioisotope Production Facility," states that the application should provide the same type of information prescribed in Section 3.5a on the design, construction and operating characteristics of all safety-related systems and components in the radioisotope production facility.* *NWMI PSAR Chapter 1, Table 1-12, "Summary of Confinement Engineered Safety Features," identifies several IROFS and/or SSCs associated with the offgas treatment and building ventilation system which play an important role in protecting both the worker and public health and safety. These include the process vessel emergency purge system (IROFS FS-03), the exhaust stack height (IROFS FS-05), and the primary offgas relief system (IROFS RS-09).* *The design basis earthquake for the RPF is identified in Section 3.4.1.1 of the PSAR (i.e., earthquake with a 2,500-year return period). The PSAR does not state if this seismic design criteria is also being applied to important components within the structure. The staff needs to understand if these components which NWMI has identified as being important for public health and safety are designed to withstand the same design basis earthquake as the facility structure.* *Describe the seismic design criteria for portions of the ventilation and offgas systems that NWMI has identified as IROFS and is relying on to protect workers and the public.*

The same seismic design criteria for the RPF structure to withstand a design basis earthquake also applies to the ventilation and offgas systems components that are accredited IROFS. These components will be designed to perform their safety function following a design basis event (DBE), including an earthquake.

No.	Request for additional information
RAI 3.5-8	*Subparagraph 50.35(a)(1) of 10 CFR Part 50 requires that the applicant describe the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.* *The ISG Augmenting NUREG-1537, Part 2, Section 3.5b, states that the application should provide the same type of information prescribed in Section 3.5a on the design, construction and operating characteristics of all safety-related systems and components in the radioisotope production facility.* *NWMI PSAR Section 3.1.7, "Codes and Standards," identifies codes and standards used as guidance for the design of the facility. Included in the list is the American Society for Testing and Materials (ASTM) 1533 Standard Guide for General Design Considerations for Hot Cell Equipment (2008). ASTM 1533 (both the 2008 version and the more recent 2015 version) states in Section 7.13, "The method of hot cell equipment repair should be considered during the design phase." The application does not discuss design features intended to facilitate maintenance in the remote cells, particularly the tank hot cell.* *Describe how NWMI has considered maintenance operations, particularly in the hot cells, when it performed its analysis that identified major features that are incorporated into the design for protection of the health and safety of workers and the public.*

Each of the hot cells will have manipulators that will be used to perform maintenance within the hot cells. Equipment within the hot cells will also be positioned on skids for ease of removal and replacement if necessary. If maintenance cannot be performed by the in-cell manipulators, the cover blocks can be removed and the required equipment replaced. For the tank hot cell, a portable manipulator can be moved to different locations with the tank hot cell to perform maintenance. The design philosophy that will be incorporated in the FSAR as part of the Operating License Application will use remote handling for as much maintenance as possible within the hot cells. In additional, the ventilation and changes in building configuration will be designed to maintain zones and barriers consistent with defense-in-depth, redundancy, and independence to protect workers and the public.

No.	Request for additional information
RAI 3.5-9	*The ISG Augmenting NUREG-1537, Part 1, Section 6b.3, "Nuclear Criticality Safety in the Radiological Production Facility," states, in part, that the license application must propose equipment, facilities, and procedures to ensure that the design provides for criticality control, including adherence to the double-contingency principle. Furthermore, Section 6b.3 of the ISG Augmenting NUREG-1537, Part 1 states, in part, that:* *"The applicant should describe a program that ensures compliance with the double-contingency principle, where practicable. Processes in which there are no credible accident scenarios that lead to criticality meet the double-contingency principle by definition. This principle, as given in American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," states that at least two changes in process conditions must occur before criticality is possible. If there are no process changes leading to criticality, then the principle is satisfied."* *The ISG Augmenting NUREG-1537, Part 2, Section 6b.3 states that the applicant should describe a program that ensures compliance with the double-contingency principle. Additional details on double-contingency are provide in NUREG-1520, which contains current industry practices on controlled parameters and acceptance criteria. NUREG-1520, Section 3.4.3.2, "Integrated Safety Analysis Summary and Documentation," states, in part, that if a process is designed, and an IROFS procedure specified, to ensure critical mass control by double-batching proof, the margin from a single batch to the subcritical limit should be specified. Traditionally, the single batch is 45% of the subcritical limit. NUREG-1520, Section 5.4.3.1.7.3, "Evaluation and Implementation of Controlled Parameters", Subsection 4c, states that when overbatching of special nuclear material (SNM) is credible, the largest mass resulting from a single failure is shown to be subcritical and overbatching beyond double-batching should be considered unless it requires multiple independent failures or is precluded by equipment capacity, availability of material, or other considerations.* *NWMI PSAR Chapters 1, 3, 4, and 6 discuss the results of preliminary criticality safety evaluations and the passive design features, active engineering features, and administrative controls necessary for adherence to the double-contingency principle. PSAR Chapters 6 and 13 identify double-batching as a bounding accident. In response to RAI 3.5-1 (ADAMS Accession No. ML16123A119), NWMI states that SSCs will be used to prevent an inadvertent criticality accident and in response to RAI 6.3-5, NWMI commits to maintaining a rigorous nuclear criticality safety program using the double-contingency principle. The PSAR does not specifically address the margin from a single process batch to the subcritical limit nor does it state if overbatching of SNM is credible and whether overbatching beyond double-batching has been considered in the NWMI RPF design.* *Additional information is needed for the NRC staff to understand the technical aspects of the criticality safety evaluation and confirm that the acceptance criteria of the ISG Augmenting NUREG-1537 have been met. Specifically, the NRC staff needs to confirm the margin from a single batch to the subcritical limit and whether the margin meets the traditional value of 45%. The NRC staff also needs to understand whether overbatching is credible for the NWMI RPF or is precluded by equipment capacity, availability of material, or other consideration. It is not clear whether the information provided in the PSAR is complete and comprehensive regarding the quantities of fissionable material contained in the process equipment during normal operations.*
RAI 3.5-9a	*Provide additional detail on the margin from a single process batch to the subcritical limit in all production process equipment and provide an evaluation of overbatching or a reference to such an evaluation.*

NWMI developed several calculations (e.g., NWMI-2015-CRITCALC-006, *Hot Cell Tank Pit*, and NWMI-2015-CRITCALC-002, *Irradiated Target Low-Enriched Uranium Material Dissolution*), that looked at overbatching. The hot cell pit criticality calculation, NWMI-2015-CRITCALC-006, looked at conditions in the large hot cell that were well above double-batching conditions.

No.	Request for additional information
RAI 3.5-9b	*Provide a listing of all areas and process equipment that are expected to contain fissionable material.*

The list of all areas and process equipment that are expected to contain fissionable material is provided in PSAR Chapter 4.0, Sections 4.3 and 4.4 (e.g., Table 4-51, "Uranium Recovery and Recycle In-Process Special Nuclear Material Inventory").

RAI 3.5-9c	*Provide the maximum quantities (kgs) or concentrations (g/l) of fissionable material possible in all areas or process equipment during normal operations.*

The maximum quantities or concentrations of fissionable material for all process areas or process equipment during normal operations are provided in PSAR Chapter 4.0, Sections 4.3 and 4.4 (e.g., Table 4−51).

RAI 3.5-9d	*Provide the bases for the above maximum quantities (numbers and types of targets or SNM in storage or being processed).*

The bases for the above maximum quantities, including numbers and types of targets or special nuclear material (SNM) in storage or being processed, are provided in PSAR Chapter 4.0, Sections 4.3 and 4.4 (e.g., Table 4-51).

RAI 3.5-9e	*Provide the quantities (kgs) or concentrations (g/L) of fissionable material used in criticality analyses for all areas or process equipment.*

The quantities or concentrations of fissionable material used in the criticality analyses for all areas or process equipment are provided in each individual criticality calculation or criticality safety evaluation. The single process batch to subcritical limit will be presented in the FSAR as part of the Operating License Application.

RAI 3.5-9f	*Provide a list of references to documents containing criticality analyses for above areas or process equipment.*

The criticality safety analyses conducted for the RPF are documented in the following:

- Criticality Calculations
  - NWMI-2015-CRITCALC-001, *Single Parameter Subcritical Limits for 20 wt% Uranium-235 – 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-004, *Single Parameter Subcritical Limits for 20 wt% Uranium-235 – Low-Enriched Uranium Target Material*
  - NWMI-2015-CRITCALC-005, *Target Fabrication Tanks, Wet Processes, and Storage*
  - NWMI-2015-CRITCALC-006 *Hot Cell Tank Pit*
- Criticality Safety Evaluations
  - NWMI-2015-CSE-001, *Irradiated Target Handling and Disassembly*
  - NWMI-2015-CSE-002, *Irradiated Low-Enriched Uranium Target Dissolution*
  - NWMI-2015-CSE-003, *Molybdenum-99 Recovery*
  - NWMI-2015-CSE-008, *Hot Cell Uranium Purification (Recovery and Recycle)*
  - NWMI-2015-CSE-009, *Liquid Waste Processing*
  - NWMI-2015-CSE-010, *Solid Waste Collection, Encapsulation, and Staging*
  - NWMI-2015-CSE-011, *Offgas and Ventilation*
  - NWMI-2015-CSE-012, *Target Transport Cask and Drum Handling*
  - NWMI-2015-CSE-013, *Analytical Laboratory*

CHAPTER 5.0 – COOLANT SYSTEMS

No.	Request for additional information
	Section 5.1 – Summary Description
RAI 5.1-1	*The ISG Augmenting NUREG-1537, Part 2, Section 5b, "Radioisotope Production Facility Cooling Systems," states, in part, that "the reviewer should ascertain that the application has provided an adequate analysis to ensure that there is no need for auxiliary cooling during the course of any part of the radioisotope production process." The ISG Augmenting NUREG-1537, Part 1, Section 5b, "Radioisotope Production Facility Cooling Systems," states, in part, that "the license application should consider the decay time allowed after the end of irradiation in the reactor before the separation process progresses."* *NWMI PSAR Section 5.1.1, "Irradiated Target Basis," states a minimum decay time for receipt of targets based on the location of the RPF relative to the reactor sites. NWMI, in a public meeting presentation on March 16, 2016 (ADAMS Accession No. ML16153A409), stated that the estimated travel time and distance from the MURR irradiation site and the RPF is 30 minutes and 6 miles. The estimated travel time of 30 minutes is considerably less than the minimum decay time for receipt specified in Section 5.1.1. These targets form the basis of the heat generation rate calculations for evaluating the need for auxiliary cooling.* *Additional information is needed for NRC staff to understand the basis of the decay time assumed for MURR targets and that the decay time is conservative for evaluating the need for auxiliary cooling.* *Provide additional detail on the minimum decay time allowed after the end of irradiation of MURR targets including how the handling and transportation times have been determined and demonstrate why the minimum decay time for target receipt from MURR is conservative for evaluating the need for auxiliary cooling.*

Several material-handling steps must occur after the end of irradiation within the reactor before a cask containing irradiated targets can be transported to the RPF. Examples include transfer of targets into the cask, removal of the loaded cask from the reactor pool, assembly of the cask lid, removal of water from the cask, drying the cask, performing the cask leak-check procedure, and cask decontamination and verification. At-reactor handling procedures are projected to require significantly longer than eight hours for an individual cask. Independent of the actual cask handling time required, the clock time for end of irradiation of a target batch becomes a datapoint recorded on transfer papers, and a cask will not be unloaded until the minimum decay time after end of irradiation used in safety evaluations has elapsed.

RAI 5.1-2	*The ISG Augmenting NUREG-1537, Part 2, Section 5b, "Radioisotope Production Facility Cooling Systems," states, in part, that the reviewer should ascertain that the application has provided an adequate analysis to ensure that there is no need for auxiliary cooling during the course of any part of the radioisotope production process.* *NWMI PSAR Section 5.1.1, "Irradiated Target Basis," states, in part, that "The MURR operation is based on irradiating eight targets per week..." and "Therefore, heat load from receipt of MURR targets has been used as an upper bound for irradiated target receipts at the RPF." PSAR Section 4.1.2.1, "Process Design Basis," states that, "The RPF is designed to have a nominal operational processing capability of one batch per week of up to 12 targets from University of Missouri Research Reactor (MURR)." The nominal operational processing capability of 12 targets per week from MURR as stated in PSAR Section 4.1.2.1 is greater than the 8 targets per week from MURR as stated in PSAR Section 5.1.1 for evaluating the need for auxiliary cooling.* *Additional information is needed for NRC staff to understand the upper bound for irradiated target receipts at the RPF as used in the analysis to determine the need for auxiliary cooling.*

No.	Request for additional information
RAI 5.1-2 (cont)	*Resolve the inconsistency between the number of MURR targets (i.e., 8 per week) used for assessing the thermal characteristics of the processing equipment and the number of MURR targets (i.e., 12 per week) used in the RPF facility design basis and demonstrate that 8 targets per week is a conservative design basis for the cooling system evaluation considering that the RPF is designed to have a nominal operational processing capability of up to 12 MURR targets per week.*

The target load per week described in PSAR Section 5.1.1 will be changed to 12 MURR targets per week in the FSAR as part of the Operating License Application. The modification will include update of NWMI-2015-CALC-022, *Maximum Vessel Heat Load, Temperature, and Pressure Estimates*, with a more detailed analysis and revision of PSAR Section 5.1.1, Figure 5-2. The inconsistency identified is not expected to modify the thermal analysis in subsequent sections of PSAR Chapter 5.0. The thermal load is characterized by radial heat transfer in a vessel and the uranium concentration of solutions held within vessels throughout the RFP. Increasing the number of targets processed during a given week increases the total liquid volume contained in geometrically favorable vessels (or liquid level height), but does not change the uranium concentration or radial thermal flux.

CHAPTER 6.0 – ENGINEERED SAFETY FEATURES

No.	Request for additional information
	Section 6.3 – Nuclear Criticality Safety in the Radioisotope Production Facility
RAI 6.3-8	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that the applicant should include a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following: (a) a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, (b) determine the bias and uncertainty in the bias, and (c) determine the upper subcritical limit.* *Table 3 and Section 5.1.1 of the Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0) shows the majority of benchmark experiments to be in the range of 0.005 to 0.1 MeV, which is stated as being in the thermal range. The minimum value of the average neutron energy causing fission (ANECF) for any benchmark is given as 0.0043 MeV, which is considered significantly above the thermal range.*
RAI 6.3-8a	*Provide the Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0).*

NWMI-2014-RPT-006, *MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections* (Rev. 0) (public version) is provided in Attachment B.

RAI 6.3-8b	*Clarify if the units on the ANECF values given in MeV are correct in the validation report.*

The units in the average neutron energy causing fission (ANECF) values given in the validation report (NWMI-2014-RPT-006) are in million electron volts (MeV).

RAI 6.3-9	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that the applicant should include a summary description of a documented, reviewed, and approved validation report (by NCS [nuclear criticality safety] function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following: (a) a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, (b) determine the bias and uncertainty in the bias, and (c) determine the upper subcritical limit.* *The Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0) states that the low-enriched uranium (LEU) and intermediate-enriched uranium (IEU) benchmarks only cover the range of 8 – 1611 in H/X, but the Area of Applicability (Table 13) states that the range is covered from 0 – 1400. (The high-enriched uranium [HEU] cases extend down to H/X of 3, but they are not used in determining the upper subcritical limit and applicability has not been demonstrated.) Since the neutron spectrum is very sensitive to changes in H/X in the low H/X limit, additional information is needed for including the range from 0 – 8 for enrichments occurring at the facility.* *Justify including the range of H/X from 0 – 8 in Table 13. If HEU cases are used in making this justification, demonstrate their applicability to your operations.*

The validation report was generated prior to any calculations being performed for the NWMI RPF process. The intention was to provide as broad a base of coverage within each area of applicability (AoA) parameter range as possible. The H/X range was extended below a value of 8 based on a data trending analysis performed in the validation report (NWMI-2014-RPT-006). Subsequent to publishing the validation report, analyses have been performed for all NWMI processes showed that the extrapolation is no longer necessary. Therefore, the AoA for H/X will be changed to include values from 8 to 1,400.

No.	Request for additional information
RAI 6.3-10	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that the applicant should include a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following: (a) a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, (b) determine the bias and uncertainty in the bias, and (c) determine the upper subcritical limit.* *The Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0) includes an evaluation of bias in the criticality evaluation. More information is needed to determine the adequacy of the evaluation.*
RAI 6.3-10a	*Address the larger negative bias for moderators including hydrogen, reflectors including depleted uranium and labeled "miscellaneous," and forms including metal and labeled "miscellaneous" (Figures 5, 6, and 7).*

Section 5.1 of the validation report (NWMI-2014-RPT-006), including Figures 5, 6, and 7, evaluates trends in important validation parameters. The calculation methodology should have a method bias that has neither dependence on a characteristic nor is a smooth function of a parameter. If a trend in a parameter exists, the bias will vary as a function of that trend over the parameter range. If no trend in the parameter exists, then the bias will be constant over the parameter range.

Figure 5 groups individual experiments into sets that correspond to common moderators that include water, graphite, carbon fluoride (CF_2), hydrogen bound in uranium trihydride (UH_3), and no moderator. When the calculation results for these experiment sets are graphed, some of the experimental results lie below a k_{eff} of 1.0. Figure 5 does not represent a bias calculation; it is an evaluation to determine if a trend exists in the moderator parameter that would suggest the method bias (calculated in Section 5.3 of the validation report) has a dependence on moderation. In the Section 5.1.5 discussion of conclusions regarding the trending evaluation depicted in Figure 5, rather than stating the evaluation demonstrates no significant bias with the various moderators, the statement should read, "the evaluation demonstrates no significant trend with respect to moderation that would influence the method bias."

Similarly, for Figure 6, the intent is to determine if a trend exists in the reflector parameter that would suggest the method bias (calculated in Section 5.3) has a dependence on reflection. The Section 5.1.6 discussion will be modified to "the evaluation demonstrates no significant trend with respect to reflection that would influence the method bias."

For Figure 7, the intent is to determine if a trend exists with respect to chemical form that would suggest the method bias has a dependence on chemical form. Section 5.1.7 will be modified to "the evaluation demonstrates no significant trend with respect to chemical form that would influence the method bias." The method bias is developed in Section 5.3, and all of the experiment sets included in Figures 5 through 7 are evaluated there for the method bias calculation.

RAI 6.3-10b	*Justify their inclusion in the area of applicability (Table 13) without additional margin, and clarify what is meant by hydrogen moderator and by miscellaneous reflectors and forms.*

The hydrogen referred to in Table 13 and Figure 5 of the validation report (NWMI-2014-RPT-006) is included in a high-enriched uranium (HEU) experiment set with the chemical form UH_3. The miscellaneous reflectors referred to in Table 13 and Figure 6 are contained in a subset of experiments using graphite, copper, and aluminum reflectors. The miscellaneous chemical forms referred to in Table 13 and Figure 7 are a collection of 12 experiments using UH_3, uranium carbide (UC), and uranium zirconium hydride (UZrH).

As described in the response to RAI 6.3-11, the validation AoA will be changed to include only [Proprietary Information] as chemical forms.

No.	Request for additional information

The moderating materials will be changed to include no moderator and water. The reflecting materials will be changed to include no reflector, water, concrete, polyethylene, and paraffin, and the absorber materials will be changed to include aluminum, steel, stainless steel, polyethylene, and paraffin.

RAI 6.3-11	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that the applicant should include a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following: (a) a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, (b) determine the bias and uncertainty in the bias, and (c) determine the upper subcritical limit.* *The Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0) includes a discussion of materials in the area of applicability.* *The validation evaluated the bias for certain forms, moderators, and reflectors, but not absorbers. Some of these materials are only present in a small number of cases (e.g., U-ZrH and UO_2SO_4 forms; graphite, concrete, U-238, and BeO reflectors; Cu absorbers), and may not be present across the neutron energy spectrum. Therefore additional information is needed to determine whether they should be included in the area of applicability (AOA).*
RAI 6.3-11a	*Justify inclusion of the various chemical forms and moderating, reflecting, and absorbing materials in the AOA (Table 13).*

The validation report (NWMI-2014-RPT-006) was generated prior to any calculations being performed for the NWMI RPF process. The intention was to provide as broad a base of coverage within each AoA parameter range as possible. The chemical forms and moderating, reflecting, and absorbing materials present in the critical experiments were placed in the AoA. Subsequent to publishing the validation report, analyses have been performed for all RPF processes and it can now be concluded that some of the chemical forms and the moderating, reflecting, and absorbing materials listed in the AoA are not necessary to support the NWMI calculations. Therefore, the AoA will be changed to include only [Proprietary Information] as chemical forms. The moderating materials will be changed to include no moderator and water. The reflecting materials will be changed to include no reflector, water, concrete, polyethylene, and paraffin, and the absorber materials will be changed to include aluminum, steel, stainless steel, polyethylene, and paraffin.

RAI 6.3-11b	*For compounds, elements, and nuclides for which there are few benchmarks available or they do not adequately cover the range in H/X or ANECF, justify their inclusion without additional margin.*

Each calculation document includes an evaluation of the validation AoA. For systems that have compounds, elements, or nuclides that fall outside the validation AoA, an increased margin of subcriticality (MoS) may be warranted, depending on the specific problem being analyzed. The analyst will document any extrapolation beyond the validation AoA in the calculation and justify whether an increase to the MoS is or is not required.

RAI 6.3-12	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that the applicant should include a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis. The summary description of a reference manual or validation report should include the following: (a) a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, (b) determine the bias and uncertainty in the bias, and (c) determine the upper subcritical limit.* *The Validation Report (NWMI-2014-RPT-006, "MCNP 6.1 Validations with Continuous Energy ENDF/B-VII.1 Cross-Sections," Rev. 0) includes a discussion of physical parameters in the AOA. Additional information is needed to determine if the range of parameters to be modeled to support the facility design are within the AOA.*

No.	Request for additional information
RAI 6.3-12a	*Describe the physical parameters associated with the anticipated facility design, and compare that to the area of applicability (Table 13).*

The validation report (NWMI-2014-RPT-006) was generated prior to any criticality safety calculations being performed for the NWMI RPF process. The intent was to provide as broad a base of coverage within each AoA parameter range as possible. The actual physical parameters associated with the facility and process designs were not established at the time the validation report was originally issued (as Atkins-NS-NMI-14-01, 2014). Subsequent to its issue, criticality safety calculations have been performed. Each calculation documentation includes an evaluation of the validation AoA. For systems that are outside the validation AoA, an increased MoS may be warranted, depending on the specific problem being analyzed. The analyst will document any extrapolation beyond the validation AoA in the calculation and justify whether an increase to the MoS is or is not required.

RAI 6.3-12b	*Justify the minimum margin of subcriticality of 0.05.*

[Proprietary Information]

No.	Request for additional information
[Proprietary Information]	

No.	Request for additional information
	[Proprietary Information]

No.	Request for additional information
[Proprietary Information]	
RAI 6.3-13	*[Proprietary Information]*
RAI 6.3-13a	*[Proprietary Information]*
[Proprietary Information]	
RAI 6.3-13b	*[Proprietary Information]*
[Proprietary Information]	

No.	Request for additional information
RAI 6.3-13c	*Explain why the analysis was limited to a 1 cm slab of solution on the hot cell floor.*

Per NWMI-2015-CRITCALC-006, the modeled maximum volume of any of the uranium containing vessels is [Proprietary Information]. To reach a 1 cm (0.39-in.) depth on the floor of the uranium purification hot cell floor, [Proprietary Information] would have to be spilled at the same time. This would be equivalent to a simultaneous spill of almost [Proprietary Information] in the hot cell. A spill of this magnitude should bound any spill expected to occur during operation of the facility. In addition, all of the vessels in the interaction calculation were assumed to be full of fluid, even though a large portion of the actual fluid would be on the floor to achieve the 1 cm (0.39-in.) height, thus maximizing the possible interaction effects of the scenario.

RAI 6.3-14	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that criticality accident evaluations should include accident analyses involving licensed materials and an interpretation of the sequence of events. It is presumed that all criticality accident analyses would assume high consequences; therefore, the applicant should include every credible event that could result in an uncontrolled criticality event.* *As part of the evaluation of NWMI application the staff reviewed NWMI-2015-CSE-008, "NWMI Preliminary Criticality Safety Evaluation: Hot Cell Uranium Purification," Rev. A. This document provided an evaluation of criticality safety scenarios in the hot cell purification unit. Additional information is needed to determine if this analysis ensures that the process is subcritical under normal and abnormal conditions and adequately satisfies the double contingency principle.*
RAI 6.3-14a	*Justify that the measures preventing solution backflow to the fresh resin supply system in Scenario C7 are sufficiently reliable to ensure criticality is "highly unlikely," specifically the double block-and-bleed valve and the paddle blank.*

Per NWMI-2015-SAFETY-004, various operator or maintenance personnel errors have an on-demand failure probability of 10^{-2} to 10^{-4}. Each of the administrative actions relied on as controls are also backed up by a control requiring that a manager or supervisor verify correct valve alignment/presence/absence of the paddle blank in accordance with the activity that is being undertaken. With this required verification, either the primary or secondary contingency is unlikely to occur. These actions are determined to be sufficiently independent such that for the scenario to occur, two independent, unlikely failures would be required, thus satisfying the double-contingency principle and rendering the occurrence of the scenario to be highly unlikely.

RAI 6.3-14b	*Whereas both of these controls are considered administrative, in that they rely on proper valve alignment or equipment installation, justify their use in lieu of more traditional passive backflow prevention (e.g., overflow drains, siphon breaks), in accordance with NWMI preferred design hierarchy.*

The addition of fresh resin is not a routine occurrence, and is expected to occur on an annual or semi-annual schedule. Fresh resin will typically be added to a water-filled column with no SNM present. Once the new resin is added, the valves will be closed and the blank inserted. The operating/design conditions of this equipment (column is pressurized) does not lend itself to passive features. The analyzed controls are considered to be adequate.

The current design of this equipment does not include passive backflow design features, as the analyzed controls are considered to be adequate. Consideration will be given to providing passive backflow controls for this scenario and will be provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 6.3-15	*Section 6.b.3 of the ISG Augmenting NUREG 1537, Part 2, states that criticality accident evaluations should include accident analyses involving licensed materials and an interpretation of the sequence of events. It is presumed that all criticality accident analyses would assume high consequences; therefore, the applicant should include every credible event that could result in an uncontrolled criticality event.* *As part of the evaluation of NWMI application the staff reviewed NWMI-2015-CSE-008, "NWMI Preliminary Criticality Safety Evaluation: Hot Cell Uranium Purification," Revision A. This document provided an evaluation of criticality safety scenarios in the hot cell purification unit. Additional information is needed to determine if this analysis ensures that the process is subcritical under normal and abnormal conditions and adequately satisfies the double contingency principle.*
RAI 6.3-15a	*Explain the nature and operation of tank venting and the overloop seal system described in Scenario C8, including whether they are passive or active, and justify that they are sufficiently reliable to ensure criticality is "highly unlikely."*

Scenario C8 in NWMI-2015-CSE-008 evaluates the backflow of material into the incoming air or gas lines to a vessel holding fissile material. Both the tank venting and overloop seal systems described in Scenario C8 are passive systems. An overloop seal system provides an overloop in the supply line for each gas addition stream that is of sufficient height such that even at the maximum possible pressure in the attached fissile-containing vessel, the fissile fluid cannot reach a sufficient height to flow through the loop and back into the nongeometric favorable parts of the gas system. Tanks are vented directly to the process vessel ventilation system. The vent header in the hot cell is geometrically favorable and has drain lines to favorable geometry tanks or sumps. Since both systems are passive, these controls are considered unlikely to fail, and failure of both simultaneously is considered to be highly unlikely.

RAI 6.3-15b	*Clarify whether the overloop seal system is passive, as stated in Section 4.1.4.4 of NWMI-2015-CSE-008, or active, as stated in NWMI PSAR, Section 13.2.4.8.6.*

The overloop seal system is a passive system. An overloop seal system provides an overloop in the supply line for each non-fissile utility that is of sufficient height such that even at the maximum possible pressure in the attached fissile containing vessel, the fissile fluid cannot reach a sufficient height to flow through the loop and back into the non-fissile utility vessel. No operator actions or active equipment operations are required for this system to perform its intended function.

RAI 6.3-16	*Section 6.b.3 of the ISG Augmenting NUREG-1537, Part 2, states that criticality accident evaluations should include accident analyses involving licensed materials and an interpretation of the sequence of events. It is presumed that all criticality accident analyses would assume high consequences; therefore, the applicant should include every credible event that could result in an uncontrolled criticality event.* *As part of the evaluation of NWMI application the staff reviewed NWMI-2015-CSE-008, "NWMI Preliminary Criticality Safety Evaluation: Hot Cell Uranium Purification," Rev. A. This document provided an evaluation of criticality safety scenarios in the hot cell purification unit. Section 4.2.4 states that Scenario C6 is prevented by the use of favorable-geometry intermediate day tanks isolated from unfavorable chemical reagent or water supply systems with air breaks. However, Section 6.1 describes CSE-08-PDF12 as requiring either a day tank or an air break. Insufficient information is provided on how the day tank or air break will be used to prevent criticality.* *Clarify whether both day tanks and air breaks will be used on these unfavorable geometry supply systems, or only one or the other.*

As stated in Section 4.2.4 of NWMI-2015-CSE-008, criticality in each of these systems will be prevented by incorporation of safe-geometry intermediate day tanks in the liquid systems that are physically isolated from any larger-geometry tanks with an air break, such that backflow of uranium to an unsafe geometry is physically impossible. The current wording of the control CSE-08-PDF12 does not reflect the actual design and will be revised to clarify that the control consists of a safe-geometry intermediate day tank that is physically isolated from any larger geometry tank with an air break.

CHAPTER 7.0 – INSTRUMENTATION AND CONTROL SYSTEMS

No.	Request for additional information
	Section 7.1 – Summary Description
RAI 7.1-1	*NUREG-1537, Part 1, Section 7.1, "Summary Description," states, in part, that the applicant should summarize the technical aspects, safety, philosophy, and objectives of the instrument and control (I&C) design and discuss such factors as redundancy, diversity, and isolation of functions. The ISG Augmenting NUREG-1537, Parts 1 and 2, Section 7b, "Radioisotope Production Facility Instrumentation and Control Systems," Section 7b.1, "Summary Description," states, in part, that the applicant should provide a summary description of the I&C systems including the design bases; the safety, considerations, and objectives; and the operational characteristics of the production facility that determine or limit the I&C design. NUREG-1537, Part 2, Section 7.1, "Summary Description," states, in part, that the acceptance of the summary description should be based on its completeness in addressing the factors listed in Part 1.* *NWMI PSAR Section 7.1, "Summary Description," discusses the I&C design in terms of RPF processes and systems including SNM preparation and handling processes, radioisotope extraction and purification processes, process utility and support systems, criticality accident alarm system, radiation monitoring system, facility ventilation system, and mechanical utility systems. PSAR Section 7.1 states that the facility process control (FPC) and the building management system (BMS) provide monitoring and control functions. PSAR Section 7.1 identifies how and where the processes or systems are monitored and controlled. However, it does not identify any specific I&C technical aspects, philosophy, or objectives of the instrumentation. The discussion does not address redundancy, diversity, or isolation of functions except for the engineering safety features which are stated to be independent, hard-wired analog controls. The discussion does not state the design basis or operational characteristics of the I&C.* *Additional information is needed for the NRC staff to understand the design basis, technical aspects, safety, philosophy, and objectives of the I&C design and confirm that the I&C design meets the acceptance criteria of NUREG-1537, Part 2, Section 7.1.* *Provide additional detail regarding the design bases, technical aspects, safety, philosophy, and objectives for all I&C components that monitor and control RPF processes or systems.*

The instrument and control (I&C) systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, preliminary design of the RPF I&C systems (e.g., details regarding the design bases, technical aspects, safety, philosophy, and objective for all I&C components that monitor and control RPF processes or systems) was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the final safety analysis report (FSAR) as part of the Operating License Application. Note that concepts like redundancy, independence, and diversity of systems are specifically identified as necessary in PSAR Sections 7.2 through 7.6.

For the RPF Construction Permit Application, the preliminary design of the RPF I&C systems is considered functional and at a conceptual level. Our intent at this stage was to describe the design methodology and provide reasonable assurance that the final design will conform to the design bases with an adequate margin for safety.

No.	Request for additional information
RAI 7.1-2	*NUREG-1537, Part 1, Section 7.1, "Summary Description," states, in part, that the general description of each category of I&C subsystems should include the types of parameters monitored, the number of channels designed to monitor each parameter, and the actuating logic. NUREG-1537, Part 2, Section 7.1, "Summary Description," states, in part, that the acceptance of the summary description should be based on its completeness in addressing the factors listed in Part 1.* *NWMI PSAR Section 7.1, "Summary Description," discusses the I&C design in terms of RPF processes and systems. Section 7.1 states that the FPC and the BMS provide monitoring and control functions, but it does not discuss I&C subsystems that are part of the overall FPC system and BMS. PSAR Section 7.1 identifies how and where the processes or systems are monitored and controlled without identifying the types of parameters monitored, the number of channels monitoring each parameter or the actuation logic.* *Additional information is needed for the NRC staff to understand the types and numbers of parameters monitored and the actuation logic and confirm that the I&C design meets the acceptance criteria of NUREG-1537, Part 2, Section 7.1.* *Provide additional detail for each category of I&C subsystems that includes the types of parameters monitored, the number of channels designed to monitor each parameter, and the actuating logic.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C subsystems (including types of parameters monitored, number of channels designed to monitor each parameter, and actuation logic) was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application. Section 7.2 does not address specific aspects of the I&C system, although Tables 7-4 through 7-12 list the location and types of parameters anticipated to be monitored.

For the RPF Construction Permit Application, the preliminary design of the RPF I&C systems is considered functional and at a conceptual level. Our intent at this stage was to describe the design methodology and provide reasonable assurance that the final design will conform to the design bases with an adequate margin for safety.

RAI 7.1-3	*NUREG-1537, Part 1, Section 7.1, "Summary Description," states, in part, that the general description of each category of I&C subsystems should include a summary of the human-machine interface principles used in the location of instrumentation and controls. NUREG-1537, Part 2, Section 7.1, "Summary Description," states, in part, that the acceptance of the summary description should be based on its completeness in addressing the factors listed in Part 1.* *NWMI PSAR Section 7.1, "Summary Description," discusses the I&C design in terms of RPF processes and systems. PSAR Section 7.1 states that the target fabrication process, target receipt and disassembly process, target dissolution process, molybdenum recover and purification process, and low-dose liquid waste handling will be controlled by operators at local human-machine interfaces (HMI). PSAR Section 7.1 also identifies that operators at local HMIs will control the plant air system, gas supply system, process chilled water chillers, process steam boilers, demineralized water system, chemical supply system, and standby electric power system. PSAR Section 7.1 uses several different terms (i.e., operator interface displays, operator interface terminals, and HMIs) when referring to operator-controlled equipment. PSAR Section 7.1 does not include a summary of the HMI principles used in the location of I&C and does not define the different terms used to describe operator-controlled equipment.* *Additional information is needed for the staff to understand the various HMIs being discussed and to confirm that the I&C design meets the acceptance criteria of NUREG-1537, Part 2, Section 7.1.*

No.	Request for additional information
RAI 7.1-3 (cont)	*Provide additional information for each category of I&C subsystems that includes a summary of the HMI principles used in the location of I&C and define the difference in functionality and design between operator interface displays, operator interface terminals, and HMIs.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C subsystems, including specific details on human-machine interface (HMI), was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application. To be consistent in the PSAR, terms like "operator interface displays" and "operator interface terminals" will be replaced with the single term, HMI (e.g., pages 7-i, 7-iv, 7-4, 7-15, 7-17, 7-18, 7-20, and 7-21).

For the RPF Construction Permit Application, the preliminary design of the RPF I&C systems is considered functional and at a conceptual level. Our intent at this stage was to describe the design methodology and provide reasonable assurance that the final design will conform to the design bases with an adequate margin for safety.

No.	Request for additional information
	Section 7.2 – Design of Instrumentation and Control Systems
RAI 7.2-1	*NUREG-1537, Part 1, Section 7.2.3, "System Description," states, in part, that:* *The system description in the SAR should include equipment and major components as well as block, logic, and schematic diagrams ... the applicant should submit hardware and software descriptions and software flow diagrams for digital computer systems [and that] the applicant should describe how the system operational and support requirements will be met, how the operator interface requirements will be met, [and] should address the methodology and acceptance criteria used to establish and calibrate the trip or actuation setpoints, or interlock functions.* *NWMI PSAR Section 7.2.3, "System Description," describes the RPF I&C system basic components as including the FPC system, engineering safety feature (ESF) actuation systems, control console and display instruments, and the BMS. PSAR Sections 7.2.3.1 through 7.2.3.5 do not include a description of the ESF actuation systems or the BMS. Instead, PSAR Sections 7.2.3.1 through 7.2.3.5 provide high-level functional descriptions of the FPC system, control room/operator interface, fire protection system, facility communication systems, and analytical laboratory system. PSAR Section 7.2.3 also does not contain block, logic, or schematic diagrams of these subsystems nor does it address the methodology and acceptance criteria used to establish and calibrate the trip or actuation setpoints, or interlock functions.* *Additional information is necessary for the NRC staff to understand the relationship among all of the major I&C components.*

No.	Request for additional information
RAI 7.2-1a	*Provide additional information describing all of the equipment and major RPF I&C components including block, logic, and schematic diagrams to show the relationship between the various I&C subsystems; software flow diagrams for the digital computer systems; a description of how the system operational and support requirements and operator interface requirements are met.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing all of the equipment and major RPF I&C components (e.g., block, logic and schematic diagrams, software flow diagram, and description of how system operational and support requirements and operator interface requirements are met) was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

For the RPF Construction Permit Application, the preliminary design of the RPF I&C systems is considered functional and at a conceptual level. Our intent at this stage was to describe the design methodology and provide reasonable assurance that the final design will conform to the design bases with an adequate margin for safety.

RAI 7.2-1b	*Provide a description of the methodology and acceptance criteria used to establish the trip or actuation setpoints or interlock functions.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing the detailed methodology and acceptance criteria used to establish trip or actuation setpoints or interlock functions was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

As discussed briefly in PSAR Sections 7.2.4.1 and 7.2.4.2, trip or actuation setpoints for systems in Section 7.2 will be established to indicate a warning when a given parameter is approaching a setpoint and alarm/trip when it has reached a setpoint, both at the HMI and the control station, as appropriate. Alarm/trip setpoints will be established at levels that are protective of systems relied on for safety, as described in the PSAR (and follow-on FSAR), particularly IROFS. This means that alarm/trip setpoints will be established to provide reasonable assurance that these systems will be consistent with the design requirements and limitations established by the bounding analysis found in the PSAR and follow-on FSAR.

For the RPF Construction Permit Application, the preliminary design of the RPF I&C systems is considered functional and at a conceptual level. Our intent at this stage was to describe the design methodology and provide reasonable assurance that the final design will conform to the design bases with an adequate margin for safety.

No.	Request for additional information
RAI 7.2-2	*NUREG-1537, Part 1, Section 7.2.4, "System Performance Analysis," states, in part, that the applicant should conduct a performance analysis of the proposed system to ensure the design criteria and design bases are met and licensing requirements for the performance of the system are specified. The analysis should describe the operation of the I&C system and present the analysis of how the system design meets the design criteria and design bases.* *NWMI PSAR Section 7.2.4, "System Performance Analysis," states that IROFS will be <u>managed</u> by the FPC system. PSAR Section 7.2.4.2.2 states that the FPC system will initiate and control ESF activation and isolation. PSAR Section 7.2.4.2.6 states that the FPC system will have the ability to perform a manual activation of the ESF. However, PSAR Section 7.1 states that ESF systems will operate independently from the FPC system or BMS; will use hard-wired analog controls/interlocks to protect workers, the public, and environment; and ESF parameters and alarm functions will be integrated into and monitored by the FPC system or BMS. The different descriptions of the FPC system functionality in different sections of the PSAR appear to be inconsistent.* *Additional information is necessary for the NRC staff to fully understand the operation of the integrated RPF I&C system.* *Provide additional information in the I&C system description and performance analysis of the FPC system as it relates to ESF managing, monitoring, and actuation. Resolve the apparent inconsistency between the description in PSAR Section 7.1 that states "ESF systems are only monitored by the FPC," and the descriptions in PSAR Sections 7.2.4.2.2 and 7.2.4.2.6 that state "ESF system activation and isolation is initiated and controlled by the FPC and the FPC system has the ability to manually activate ESF."*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing the detailed methodology and operation of the integrated facility process control (FPC) system as it relates to engineered safety features (ESF) managing, monitoring, and actuation was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

PSAR Section 7.1 states, "Engineered safety feature (ESF) systems will operate independently from the FPC system or BMS." This sentence will be amended in future versions of the PSAR to say, "Engineered safety feature (ESF) systems will operate upon actuation of an alarm setpoint reached for a specific monitoring instrument/device. For redundancy, this will be in addition to the FPC system or BMS ability actuate ESF as needed." By amending this sentence, the descriptions in PSAR Sections 7.2.4.2.2 and 7.2.4.2.6 will be consistent with Section 7.1.

RAI 7.2-3	*NUREG-1537, Part 1, Section 7.2.5, "Conclusion," states that the applicant should summarize why the system design is sufficient and suitable for performing the functions stated in the design basis.* *NWMI PSAR Section 7.2.4.3, "Conclusion," states, in part, that the RPF I&C systems will meet the stated design criteria and design basis requirements outlined in NUREG-1537. PSAR Table 7-2, "Instrumentation and Control Criteria Crosswalk with Design Basis Applicability and Function Means," presents a crosswalk of the I&C subsystems and attempts to cross-reference the different I&C subsystems to specific design criteria. The table lacks the cross-referencing information necessary to locate the specific PSAR Chapter 7 section that provides a detailed description of how the design basis is satisfied. The omission of the cross-referencing information prevents evaluating that the system design is sufficient and suitable for performing the functions stated in the design basis.* *Additional information is necessary to fully understand the operation of the integrated RPF I&C systems.*

No.	Request for additional information
RAI 7.2-3 (cont)	*Provide additional information in Table 7-2 that provides a cross-reference to the specific section of PSAR Chapter 7 that discusses how the system is suitable for performing the functions stated for each design basis applicability item.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing the detailed methodology and operation of the integrated I&C systems was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

When the final RPF design is complete, PSAR Chapter 7.0, Table 7-2, will be expanded to provide a cross-reference to the specific section of each I&C section and how the system is suitable for performing the functions stated for each design basis applicability item.

No.	Request for additional information
	Section 7.3 – Process Control Systems
RAI 7.3-1	*The ISG Augmenting NUREG-1537, Part 2, Section 7b.3, "Process Control Systems," Acceptance Criteria, states, in part, that the system should be designed with sufficient control of reactivity for all required production and SNM fuel reconditioning process operations.* *NWMI PSAR Section 7.3, "Process Control Systems," states, in part, that the RPF process control will be administered by the FPC system and, 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. 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 control. The operator will initialize the transfer of fluids. The applicant must control quantities of fissionable materials and assure the quality of both software and operating procedures to preclude the possibility of criticality accidents.* *Additional information is necessary to understand how the key parameters are monitored to ensure adequate criticality control.* *Provide additional information regarding the adequacy of the facility's instrumentation to detect deviations from nominal concentrations and quantities, should they occur or provide a status of the development of the software and/or procedures, stating why this is not necessary for a CP.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing how the key parameters are monitored to ensure adequate criticality control (e.g., instruments to detect deviations from nominal concentrations and quantities, status of software development procedures) was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 7.4 – Engineered Safety Features Actuation Systems
RAI 7.4-1	*The ISG Augmenting NUREG-1537, Part 2, Section 7b.4, "Engineered Safety Features Actuation Systems," states, in part, that this section of the PSAR should describe the actuation systems for any ESFs discussed in PSAR Chapters 6 or 13. NUREG-1537, Part 1, Section 7.5, "Engineered Safety Features Actuation Systems," states, in part, that the "applicant should describe the ESF actuation system in sufficient detail to describe the functions required of the ESF and the operation of the system."* *NWMI PSAR Section 7.4, "Engineered Safety Features Actuation Systems," Section 7.4.1, "System Description," states that the PSAR Table 7-13, "Engineered Safety Feature Actuation or Monitoring Systems," lists the ESFs that will require actuation by the I&C system. PSAR Section 7.4 does not describe the actuation systems for any ESFs.* *Additional information is necessary to understand the functionality required of the ESF and operation of the system to meet the acceptance criteria of NUREG-1537, Part 2, and the ISG Augmenting NUREG-1537, Part 2.* *Provide additional information in sufficient detail describing the required functions and operation of the ESF actuation system.*

The I&C systems preliminary design was developed to ensure the sufficiency of the principal design criteria, design bases, and information relative to materials of construction, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. In addition, the preliminary design of the RPF I&C systems describing the functionality and operation required of the ESFs was *not* developed to constitute approval of the safety of any design feature or specification. Such approval is anticipated to be made following the evaluation of the final design of the RPF I&C system, and described in the FSAR as part of the Operating License Application.

The fourth column of PSAR Chapter 7.0, Table 7-13, provides some information on the anticipated technical means by which an ESF would be actuated; this mechanism is not described further in PSAR Section 7.4 because the design has not been finalized. The intent was only to provide reasonable assurance that we recognize that specifics of the ESF actuation mechanism will need to be addressed as part of the FSAR in the Operating License Application.

CHAPTER 8.0 – ELECTRICAL POWER SYSTEMS

No.	Request for additional information
	Section 8.2 – Emergency Electric Power Systems
RAI 8.2-1	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information maintained by the applicant be complete and accurate in all material respects.* *NUREG-1537, Part 1, Section 8.2, "Emergency Electric Power Systems," states that the design bases should provide voltage power requirements for emergency electric power systems.* *NUREG-1537, Part 2, Section 8.2, Acceptance Criteria, states, in part:* *The functional characteristics of the emergency power system should be commensurate with the design bases, which are derived from analyses presented in other chapters of the SAR. ... The source of electrical power (generator, batteries, etc.) should be capable of supplying power for the duration required by the SAR analysis.* *NWMI PSAR Section 8.1.2, "Design for Safe Shutdown," states that the uninterruptable power supplies (UPS) will be designed to operate up to 90 minutes, except the UPS for the fire protection system will operate for up to 24 hours. However, PSAR Section 3.5.2.7.9, "Standby Electrical Power," states that the design basis value for the UPSs is to "... maintain power availability for a minimum of 120 min [minutes] post-accident."* *Provide additional information to resolve or explain this apparent discrepancy in power availability times.*

PSAR Section 8.1.2 and 8.2 values were changed to 120 minutes to reflect the design basis in PSAR Section 3.5.2.7.9.

No.	Request for additional information
RAI 8.2-2	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information maintained by the applicant be complete and accurate in all material respects.* *NUREG-1537, Part 1, Section 8.2, "Emergency Electric Power Systems," states that the design bases should provide voltage power requirements for emergency electric power systems.* *NUREG-1537, Part 2, Section 8.2, Acceptance Criteria, states, in part:* *The functional characteristics of the emergency power system should be commensurate with the design bases, which are derived from analyses presented in other chapters of the SAR. ... The source of electrical power (generator, batteries, etc.) should be capable of supplying power for the duration required by the SAR analysis.* *NWMI PSAR Section 8.2 states that a 1000-kilowatt (kW) (1341 horsepower [hp]) diesel generator will provide standby electrical power (SEP). However, PSAR Section 8.2.2, "Ranges of Emergency Electrical Power Required," states, in part, that "The total peak SEP required for the RPF is 1,140 kW (1,528 hp)." Further, PSAR Table 8-1, "Summary of Radioisotope Production Facility and Ancillary Facilities Electrical Loads," lists the RPF electrical loads and shows that the total SEP required is 1,173.6 kW (1,585 hp). PSAR, Chapter 19.0, "Environmental Review," Table 19-60, "Emissions for Standby Emergency Diesel Generator" cites 2,600 kW as the basis for diesel generator emissions.* *Provide additional information to resolve or explain these apparent discrepancies in SEP requirements.*

The column headings in Table 8-1 of PSAR Chapter 8.0 were changed from "...power requirement" to "...peak power load" to be consistent with the description preceding the table. PSAR Section 8.2.2 will be modified to reflect the peak power of 1,178.6 kW (1,585 hp), as determined from Table 8-1. PSAR Chapter 19.0 used a larger estimate to ensure that emissions were bounded.

CHAPTER 9.0 – AUXILIARY SYSTEMS

No.	Request for additional information
	Section 9.1 – Heating, Ventilation and Air Conditioning (HVAC) Systems
(Applies to RAIs 9.1-1 through 9.1-3)	*NUREG-1537, Part 1, Chapter 9, "Auxiliary Systems," states that the applicant should include the design bases for each auxiliary system.* *NUREG-1537, Part 1, Chapter 3, "Design of Structures, Systems and Components," states, in part, that: "The bases for the design criteria for some of the systems discussed in this chapter may be developed in other chapters and should be appropriately cross referenced."* *NUREG-1537, Part 2, Section 9.1, "Heating, Ventilation, and Air Conditioning Systems," Review Procedures, states, in part, that: "The design bases should be compared with requirements from other chapters of the SAR ..."*
RAI 9.1-1	*NWMI PSAR Section 9.1.1, "Design Basis," states:* *The ventilation system is designed to provide confinement of hazardous chemical fumes and airborne radiological materials and conditioning of the RPF environment for facility personnel and equipment. The design basis of the ventilation system includes:* • *Confining hazardous chemical fumes* • *Preventing release and dispersal of airborne radioactive materials to protect health and minimize danger to life or property* • *Maintaining dose uptake through ingestion to levels ALARA [as low as reasonably achievable]* • *Providing ventilation air and conditioning the RPF environment for workers or occupants* • *Providing makeup air and condition the RPF environment for process and electrical equipment* *Whereas, PSAR Section 3.5.2.7.12, "Facility Ventilation System," which appears to refer to what PSAR Section 9.1.1 calls the "Ventilation System," states that the "facility ventilation system" design basis "functions" are as follows:* • *Provide confinement of hazardous chemical fumes and airborne radiological materials and conditioning of RPF environment for facility personnel and equipment* • *Prevent release and dispersal of airborne radioactive materials (e.g., maintain pressure gradients to ensure proper flow of air from least potentially contaminated areas to most potentially contaminated areas) to protect health and minimize danger to life or property* • *Maintain dose uptake through ingestion to levels as low as reasonably achievable (ALARA)* • *Provide makeup air and condition the RPF environment for process and electrical equipment* • *Process exhaust flow from the process vessel ventilation system* • *Provide confinement of airborne radioactive materials by providing for the rapid, automatic closure of isolation dampers within confinement zones for various accident conditions* • *Provide conditioned air to ensure suitable environmental conditions for personnel and equipment in RPF* *Thus, PSAR Section 9.1.1 appears to be inconsistent with PSAR Section 3.5.2.7.12 in that PSAR Section 3.5.2.7.12 refers to the "facility ventilation system," which is called the "ventilation system" in PSAR Section 9.1.1. PSAR Section 3.5.2.7.12 also lists two design basis functions for the facility ventilation system that are not listed in PSAR Section 9.1.1 for the "ventilation system," specifically:* • *Process exhaust flow from the process vessel ventilation system* • *Provide confinement of airborne radioactive materials by providing for the rapid, automatic closure of isolation dampers within confinement zones for various accident conditions* *Additionally, PSAR Section 3.5.2.7.12 lists five design basis "values" that are not included among the design bases in PSAR Section 9.1.1; specifically:* • *Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs* • *Provide an integrated leak rate for confinement boundaries that meets the requirements of accident analyses that complies with 10 CFR 20 dose limits* • *30-year design life*

No.	Request for additional information

- *Maintain occupied space at 24 degrees Celsius (°C) (75 degrees Fahrenheit [°F]) (summer) and 22°C (72°F) (winter), with active ventilation to support workers and equipment*
- *Maintain air quality that complies with 10 CFR 20 dose limits for normal operations and shutdown*

It is not clear whether what is called the "facility ventilation system" in PSAR Section 3.5.2.7.12 is exactly the same as what is called the "ventilation system" in PSAR Section 9.1.1, or if the terms are not interchangeable, whether one of the systems, or which parts thereof, are parts of the other system.

Additional information is needed to clarify the use of inconsistent terminology for systems and subsystems. Inconsistent design basis functions/values make it difficult to identify system boundaries and to determine the adequacy of the design basis.

Provide additional information to resolve and explain these apparent discrepancies in the description of the ventilation system. Explain how NWMI will correct the apparent discrepancies, as necessary, in the PSAR, and explain how NWMI will ensure the PSAR uses consistent terminology in these and other sections.

PSAR Section 9.1 will be modified to clarify terminology and to correct the apparent discrepancies. The bulleted items in PSAR Section 9.1 will be deleted, and the design basis description in Section 9.1.1 will be modified to cross-reference to PSAR Section 3.5.2.7.12 and to PSAR Chapter 6.0. References to "ventilation system" in PSAR Section 9.1 will be amended to read "facility ventilation system."

RAI 9.1-2 *NWMI PSAR Section 9.1.1 states, in part:*

The offgas treatment system will provide primary system functions to protect on-site and off-site personnel from radiological and other industrial-related hazards by:

- *Collecting the air in-leakage sweep from each of the numerous vessels and other components in the main processes*
- *Segregating the numerous vessel vents into subsystems*
- *Directing the vessel vent subsystem that may contain iodine to abatement equipment to reduce any iodine levels in the offgas stream*
- *Collecting a special high-volume humid vessel vent stream from the waste handling system*
- *Merging the collected vent subsystems into the main facility ventilation system for further treatment or filtration to remove entrained radionuclides*
- *Removing radionuclide particulate to comply with facility emission limits*
- *Providing airflow and negative pressure to the fission product offgas system and to ventilate RPF vessels*

Additional information on the design basis is provided in Chapter 3.0, "Design of Structures, Systems, and Components."

However, PSAR Section 3.5.2.7.11, "Process Vessel Ventilation System," appears to refer to what PSAR Section 9.1.1 calls the "offgas treatment system," and states that the "process vessel ventilation system" design basis "functions" are as follows:

- *Provide primary system functions to protect on-site and off-site personnel from radiological and other industrial related hazards*
- *Collect air in-leakage sweep from each of the numerous vessels and other components in main RPF processes and maintain hydrogen concentration process tanks and piping below lower flammability limit*
- *Minimize reliance on administrative or complex active engineering controls to provide a confinement system as simple and fail-safe as reasonably possible*

PSAR Section 3.5.2.7.11 then lists design basis "values" as follows:

- *Prevent inadvertent criticality and maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs*
- *30-year design life*
- *Contain and store noble gases generated in the RPF for at least 60 days*

No.	Request for additional information
	Thus, PSAR Section 9.1.1 appears to be inconsistent with PSAR Section 3.5.2.7.11 in that PSAR Section 3.5.2.7.11: • *Refers to the "process vessel ventilation system," which is apparently called the "offgas treatment system" in PSAR Section 9.1.1* • *Lists three design basis functions for the process vessel ventilation system that are not listed in PSAR Section 9.1.1 for the "offgas treatment system"* • *Omits six of the seven design bases given in PSAR Section 9.1.1 for the offgas ventilation system (the exception being the design basis/function of "Collect air in-leakage sweep from each of the numerous vessels and other components in main RPF processes")* • *Lists three design basis "values" that are not included among the design bases given in PSAR Section 9.1.1* *Additional information is needed to clarify whether the "process vessel ventilation system" in PSAR Section 3.5.2.7.11 is the same as the "offgas treatment system" in PSAR Section 9.1.1, or if the terms are not interchangeable, whether one of the systems, or which parts thereof, are parts of the other system. The use of inconsistent terminology for systems and subsystems and inconsistent design basis functions/values make it difficult to identify system boundaries and to determine the adequacy of the design basis.* *Provide additional information to resolve or explain these apparent discrepancies in the descriptions of the ventilation system to offgas system. Explain how NWMI will correct the apparent discrepancies, as necessary, in the PSAR.*
PSAR Section 9.1 will be modified to clarify terminology and to correct the apparent discrepancies. The bulleted items in PSAR Section 9.1 will be deleted. The design basis description in PSAR Section 9.1.1 will be modified to cross-reference to PSAR Section 3.5.2.7.11 and to PSAR Chapter 6.0. References to "offgas treatment system" in PSAR Section 9.1 will be amended to read "process vessel ventilation system."	
RAI 9.1-3	*NWMI PSAR Section 9.1.1, states, in part, that:* *The design basis of the ventilation system includes the following:* • *Providing ventilation air and conditioning the RPF environment for workers or occupants* • *Providing makeup air and condition the RPF environment for process and electrical equipment* *PSAR Section 3.5.2.7.12, "Facility Ventilation System," appears to refer to what PSAR Section 9.1.1 calls the "Ventilation System," and states that the facility ventilation system "design basis functions" include the following:* • *Provide makeup air and condition the RPF environment for process and electrical equipment* • *Provide conditioned air to ensure suitable environmental conditions for personnel and equipment in RPF* *PSAR Section 3.5.2.7.23, "Supply Air System," states that the design basis functions of the supply air system include the following:* • *Providing conditioned (e.g., filtered) air for facility workers and equipment and supply makeup air for exhaust air systems* • *Drawing outside air into the RPF air supply system through air handling units* • *Providing makeup air and condition the RPF environment for process and electrical equipment* • *Providing conditioned air to ensure suitable environmental conditions for personnel and equipment* *PSAR Section 3.5.2.7.23 further lists supply air system "design basis values" as follows:* • *30-year design life with the exception of common replaceable parts (e.g., pumps)* • *Maintain occupied space at 24°C (75°F) (summer) and 22°C (72°F) (winter), with active ventilation to support workers and equipment* • *Maintain air quality that complies with 10 CFR 20 dose limits for normal operations and shutdown*

No.	Request for additional information
RAI 9.1-3 (cont)	*PSAR Section 3.5.2.7.23 introduces another system designation (i.e., "supply air system"), which has design basis elements (functions and values) that appear to correspond to two of the design basis elements (functions/values) of the ventilation/facility ventilation system, further described in PSAR Section 9.7.2.2, "Supply Air System."* *Thus, it appears, that the "supply air system" in PSAR Section 3.5.2.7.23 is a subsystem of the "facility ventilation system" in PSAR Section 3.5.2.7.12, and the "ventilation system" in PSAR Section 9.1.1. Additional information is needed to clarify relationship between the supply air system, facility ventilation system, and ventilation system. The use of inconsistent terminology for systems and subsystems and inconsistent design basis functions/values make it difficult to identify system boundaries and determine the adequacy of the design basis.* *Provide additional information to clarify the relationship among the supply air, facility ventilation, and ventilation systems. Explain how NWMI will correct the apparent discrepancies, as necessary, in the PSAR.*

PSAR Section 9.1 will be modified to clarify terminology and to correct the apparent discrepancies. The bulleted items in PSAR Section 9.1 will be deleted, and the design basis description in PSAR Section 9.1.1 will be modified to cross-reference to PSAR Section 3.5.2.5.12 and to PSAR Chapter 6.0. References to "ventilation system" in PSAR Section 9.1 will be amended to read "facility ventilation system."

The supply air is a subsystem of the facility ventilation system. PSAR Section 3.5.2.7.23, "Supply Air System," will be eliminated and appropriate design basis values moved to PSAR Section 3.5.2.7.12, "Facility Ventilation."

RAI 9.1-4	*NUREG-1537, Part 1, Chapter 9 states that the applicant should include the design bases for each auxiliary system.* *PSAR Section 9.1.2, "System Description," states, in part: "abatement technologies, primarily high-efficiency particulate air (HEPA) filtration and activated carbon, will be used to ensure that air exhausted to the atmosphere meets 40 CFR 61, 'National Emission Standards for Hazardous Air Pollutants' (NESHAP), and applicable State law."* *The meeting of 40 CFR Part 61 and applicable state law constitutes a regulatory/legal requirement that should form part of the system design basis such that the system design will meet the performance requirements. However, the design requirement that exhaust air meet the stated regulatory/legal requirements it is not listed among the design basis elements in PSAR Section 9.1.1 and is not listed among the design basis functions or values in PSAR Section 3.5.2.7.12, as are certain other applicable regulatory requirements.* *Provide additional information to clarify whether this regulatory/legal requirement is a design basis element, function or value. If so, include it in the lists in PSAR Sections 9.1.1 and/or 3.5.2.7.12 as appropriate.*

The bulleted items in PSAR Section 9.1 will be deleted, and the design basis description in PSAR Section 9.1.1 will be modified to cross-reference to PSAR Section 3.5.2.7.12 and Chapter 6.0. Reference to 40 CFR 61, "National Emission Standards for Hazardous Air Pollutants" (NESHAP), is already included in PSAR Section 3.1.2 as a design input and will not be repeated in PSAR Section 3.5.2.7.12.

No.	Request for additional information
RAI 9.1-5	*NUREG-1537, Part 1, Chapter 9 states that the applicant should include the design bases for each auxiliary system.* *NUREG-1537, Part 1, Section 9.1 states that all used spaces in a facility may require HVAC systems to provide acceptable environments for personnel and equipment.* *NUREG-1537, Part 2, Section 9.1 states that areas of review should include features of the HVAC system that affect habitability and the working environment in the reactor facility for personnel and equipment.* *NWMI PSAR Section 9.1.2.3.1, "Zone I Exhaust System," states that the space temperature control will not be provided for Zone I spaces unless thermal loads are expected to cause temperatures to exceed equipment operating ranges without additional cooling. However, it is not explained what circumstances would result in temperatures exceeding operating ranges. Therefore, additional information is needed to determine whether HVAC systems can properly provide an acceptable environment.* *Provide additional information to explain when (i.e., design phase, construction, facility/systems testing, or operation) and how (e.g., as an operational sensing, calculation and control function) NWMI will determine whether HVAC space temperature control in Zone I will need to be provided.*

The need for HVAC space temperature control in Zone I will be evaluated and determined during the final design phase by performing a heat balance on the Zone I ventilation system. The maximum heat load on the ventilation system is anticipated to be dominated by heat losses from equipment in the Zone I ventilated areas (rather than decay heat) when operating at the maximum uranium throughput. Temperature control will also be evaluated for a loss of ventilation scenario. Results of the evaluation (including space temperature control systems that may be identified by the heat balance) will be described in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 9.3 – Fire Protection Systems and Programs
(Applies to RAIs 9.3-1 through 9.3-4)	*NUREG-1537, Part 2, Section 9.3, "Fire Protection Systems and Programs," Acceptance Criteria, states, in part, that "[m]ethods to detect, control, and extinguish fires should be stated in the plan."*
RAI 9.3-1	*NWMI PSAR Section 9.3.3.1.1, "Hot Cell, Waste Handling, and Shipping Areas," discusses the hot cell fire area. However, no information on fire suppression inside the hot cell enclosure was provided. Additional information is needed on the hot cell fire protection to determine if methods to detect, control, and extinguish fires are adequate.* *Provide information on the type of fire suppression system that will be used in the hot cell enclosure, including a discussion of any life safety concerns that could arise from use of the suppression system.*

Hot cell fire suppression systems have been commercially available for years and include product designs compliant with the National Fire Protection Association (NFPA) and other relevant industry standards. Selection of the specific hot cell enclosure fire suppression system will be finalized during the final RFP design, along with commitments to design codes, standards, and other referenced documents, including any exceptions or exemptions to the identified requirements. These final designs and commitments will be identified and provided in the FSAR as part of the Operating License Application.

The life safety concerns that could arise from the operation of hot cell fire suppression systems relate to the possible mobilization and spread of radiological contamination from the hot cell containment into other systems and facility areas, thereby posing risks of personnel exposure. The design and construction of hot cell fire suppression systems include features to handle the water and/or other types of fire suppression fluids during and after the system discharge(s) in the event of a fire detected within the hot cell.

No.	Request for additional information

Selection of the specific hot cell enclosure fire suppression system and its discharged fire suppressant handling subsystems will be finalized in the RPF and hot cell final detailed designs, along with commitments to relevant design codes and standards. These final designs and commitments will be identified and provided in the FSAR as part of the Operating License Application.

RAI 9.3-2	*NMWI PSAR Section 9.3, "Fire Protection Systems and Programs," discusses the fire protection program for the NWMI facility, including the fire detection and suppression systems, as well as alarm systems and life safety concerns, however it does not address which building or fire codes the facility will be built to. Additional information is needed to determine if methods to detect, control, and extinguish fire are adequate.* *Identify the building and fire codes that NWMI is committing to follow with regards to the fire protection program, both for construction and maintenance of systems.*

Commitments to specific building and/or fire codes (e.g., NFPA 801) will be finalized and identified in the RPF final detailed design, both for facility construction and for fire protection program maintenance. This final detailed facility design and the relevant commitments to codes and standards will be identified and provided in the FSAR as part of the Operating License Application.

RAI 9.3-3	*While NWMI PSAR, Section 9.3, provides information on fire protection system design, it does not describe how high radiation areas may affect the performance of ionization detectors, while photoelectric detectors can be affected by the presence of dust or particulates. Additional information is needed to determine the fire detection capabilities of the facility.* *Provide the basis for how detectors will be chosen, as well as for the maintenance program that will be put in place to ensure their functionality.*

Fire detection systems for severe industrial environments (e.g., high radiation, high particulate concentrations) have been commercially available for years and include product designs compliant with NFPA and other relevant industry standards. Selection of the specific fire detection technologies for a particular severe environment will be guided by the recommendations offered in such standards (e.g., NFPA 801), along with the recommendations and requirements of the test and maintenance programs to confirm the reliable functionality of these systems. The fire detection systems selected for the RPF's fire-protected areas, and the corresponding test and maintenance programs, will be included in the final RPF detailed designs, along with commitments to design codes, standards, and other referenced documents, including any exceptions or exemptions to the identified requirements. The final designs, test and maintenance programs, and standards commitments will be identified and provided in the FSAR as part of the Operating License Application.

RAI 9.3-4	*NWMI PSAR Section 9.3.2.1, "Fire Suppression Subsystem" and Section 9.3.2.2, "Fire Detection and Alarm Subsystem," states that HEPA filter plenum deluge systems and heat detectors will be used to detect and suppress fires involving the HEPA filter system.* *Provide a description of the testing and maintenance of these systems, including information about any codes or standards that NWMI is committing to.*

High-efficiency particulate air (HEPA) filter fire protection (i.e., both detection and suppression) systems include the means for nondestructive in situ testing and maintenance, and are designed and constructed in compliance with appropriate NFPA or other relevant standards. The specific HEPA filter fire protection system(s) selected will determine the most appropriate corresponding test and maintenance program(s) to be implemented. HEPA filter fire protection will be included in the final RPF detailed designs, along with commitments to the relevant design codes, standards, and other referenced documents, including any exceptions or exemptions to the identified requirements. The final fire protection system designs, test and maintenance programs, and standards commitments will be identified and provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 9.3-5	*NUREG-1537, Part 1, Section 9.3, "Fire Protection Systems and Programs," states that the application should discuss passive design features required by the facility design characteristics to, in part, limit fire consequences. The facility should be designed and protective systems should exist to prevent the uncontrolled release of radioactive material if a fire should occur.* *NWMI PSAR Section 9.3.3.1, "Radioisotope Production Facility Fire Area," discusses fire hazards and ignition sources that were considered for the RPF, but does not describe which fire safety systems or management measures are necessary to prevent or mitigate high or intermediate consequence accidents. Additional information is required to determine if the plan for the prevention of fires is adequate.*
RAI 9.3-5a	*Identify which fire detection and suppression systems are necessary to prevent or mitigate high or intermediate consequence accidents in the RPF (i.e., IROFS).*

The fire protection system was not identified as an IROFS during preliminary design and Construction Permit Application stage. Design and/or operational approaches will be used to prevent and mitigate major RPF fires causing significant external and internal releases of energy and hazardous materials. The level of consequences is typically directly related to the quantity of energy and hazardous materials released during such accidents; therefore the design, construction, operation, and maintenance of the fire protection systems will focus on minimizing these release quantities. Inherent process and systems design features such as compartmentalization, isolation, batch size limitations, and early/rapid/redundant fire detection and suppression can serve to reduce the maximum possible magnitudes of energy and hazardous material releases resulting from fires. Once finalized, the detailed design of the facility and its systems (including the final designs for fire protection and the final list of key safety systems and components to address severe accidents), along with the management programs to maintain their reliability, will be identified and provided in the FSAR as part of the Operating License Application.

RAI 9.3-5b	*Describe any management measures that will assure that these systems and components are constructed, procured, installed, and tested to ensure that they will be available and reliable to perform their intended functions when needed.*

The fire protection system was not identified as an IROFS during preliminary design and Construction Permit Application stage. Design and/or operational approaches will be used to prevent and mitigate major RPF fires causing significant external and internal releases of energy and hazardous materials. The level of consequences is typically directly related to the quantity of energy and hazardous materials released during such accidents; therefore the design, construction, operation, and maintenance of the fire protection systems will focus on minimizing these release quantities. Inherent process and systems design features such as compartmentalization, isolation, batch size limitations, and early/rapid/redundant fire detection and suppression can serve to reduce the maximum possible magnitudes of energy and hazardous material releases resulting from fires. Once finalized, the detailed design of the facility and its systems (including the final designs for fire protection and the final list of key safety systems and components to address severe accidents), along with the management programs to maintain their reliability, will be identified and provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 9.3-6	*NUREG-1537, Part 2, Section 9.3, "Fire Protection Systems and Programs," states that the fire protection plan should discuss the prevention of fires, including limiting the types and quantities of combustible materials. However, the application did not provide any information of the combustibles in each fire area. Additional information is needed to determine if the facilities plan to limit the types of quantities of combustible materials is adequate.* *Provide additional information on the types of combustibles found in each fire area and how the combustible materials will be controlled, including any administrative controls and management measures.*

The RPF's combustible loading (i.e., detailed inventory and distribution of the types and quantities of combustibles throughout the facility, both fixed and transient), along with a commitment to an administrative program to control combustibles within the facility, requires the detailed specific quantitative data that can only be derived from the final facility design. In particular, the combustible quantities of various types and their distribution within facility areas depend on the detailed specification of process equipment and components, materials and their quantities, operating and maintenance procedures, and material locations during the various operational states of the facility. Thus, the combustible loading analysis results and the administrative program to control combustibles within the RPF will be finalized and provided along with the final detailed design information in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 9.4 – Communication System
RAI 9.4-1	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information maintained by the applicant be complete and accurate in all material respects.* *NUREG-1537, Part 1, Chapter 9, "Auxiliary Systems," states that the applicant should include the design bases for each auxiliary system.* *NWMI PSAR Section 9.4.1, "Design Basis," states that:* *The communications system design basis is to provide communications during normal and emergency conditions between vital areas of the RPF and the Administration Building. This communications capability will include the ability of operators or other designated staff members to announce an emergency in all areas of the RPF and provide two-way communications between all operational areas and the control room. Design of the telecommunication system also complies with Electronic Industries Alliance and Telecommunications Industry Association requirements. Additional information on the communications system design basis is provided in Chapter 3.0.* *However, PSAR Chapter 3.0, "Design of Structures, Systems, and Components," does not appear to provide any additional information on the communication system.* *Provide additional information to clarify this apparent gap in information.*

PSAR Section 9.4.1 will be modified, and the sentence, "Additional information on the communications system design basis is provided in PSAR Chapter 3.0." will be deleted.

No.	Request for additional information
	Section 9.6 – Cover Gas Control in Closed Primary Coolant Systems
RAI 9.6-1	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information maintained by the applicant be complete and accurate in all material respects.* *NUREG-1537, Part 1, Chapter 9, "Auxiliary Systems," states that the applicant should include the design bases for each auxiliary system.* *NWMI PSAR Section 9.6.1, "Design Basis," states that:* *The [cover gas control] system design basis is to ensure that the hydrogen concentration in the coolant system is less than 25 percent of the lower flammability limit of 5 percent H_2, so that no uncontrolled release of radioactive material can occur, and that gases within the system do not reach explosive levels. Additional information on the design basis of cover gas control in the closed primary coolant system is provided in Chapter 3.0 ['Design of Structures, Systems, and Components']."* *However, the only additional information on the cover gas control function or its design basis in PSAR Chapter 3.0 is in PSAR Section 3.5.2.7.17, "Process Chilled Water System," which lists one of the design basis functions of the process-chilled water system as providing cover gas to prevent flammable conditions.* *Provide additional information to clarify this apparent gap in information.*

The RPF does not have a separate cover gas system, as typically found in a reactor. There is a function applied to the process-chilled water system in PSAR Sections 3.5 and 9.7.1.1 to ensure that the process-chilled water system is below hydrogen flammability limits.

No.	Request for additional information
	Section 9.7 – Other Auxiliary Systems
RAI 9.7-1	*NUREG-1537, Part 1, Chapter 9, "Auxiliary Systems," states that the applicant should include the design bases for each auxiliary system.* *NWMI PSAR, Section 9.7.1, "Utility Systems," states, in part:* *The utility systems will provide heating, cooling, process water, compressed gases, instrument, motive force, and other functions to support uranium processing, waste handling, and ventilation. The utility systems will include the following subsystems:* • *Process steam* • *Process chilled water* • *Demineralized water* • *Plant and instrument air* • *Gas supply, which supplies nitrogen, helium, hydrogen, and oxygen* • *Purge/sweep gas* *The PSAR further states the utility systems are designed to ensure that any potential malfunctions do not cause accidents in the RPF or an uncontrolled release of radioactivity. The systems are designed to ensure that in the event radioactive material is released by the operation of one of these systems, potential radiation exposures would not exceed the limits of 10 CFR 20 and are consistent with the NWMI ALARA program. No function or malfunction of the auxiliary systems will interfere with or prevent safe shutdown of the RPF.* *PSAR Section 9.7.1.1, "Design Basis," states that:* *The utility systems design basis is to provide:* • *Saturated steam at 1.7 kilograms (kg)/square centimeter (cm^2) (25 pounds [lb]/square inch [in^2]) and 4.2 kg/cm^2 (60 lb/in^2) gauge to various process equipment* • *Chilled water to various process equipment at no greater than 10°C (50°F) during normal operations*

No.	Request for additional information
	• *Demineralized water to the process steam and cooling water systems* • *Plant air for instrument air, air-diaphragm pump power, mechanical tools, and grouting material conveyor* • *Instrument air for bubblers and valve actuation* • *Gas supply, including nitrogen, helium, and hydrogen to the reduction furnace, and nitrogen and oxygen to the dissolvers* • *Purge/sweep gases provide adequate flow such that the accumulation of combustible gases is below hazardous concentrations and reduces radiological hazards due to accumulation of gaseous fission products* *Additional information on the utility systems design basis is provided in Chapter 3.0.* PSAR Section 3.5.2.7.14, "Plant and Instrument Air System," states that the design basis functions of the plant and instrument air system are as follows: • Provide small, advective flows of plant air for several RPF activities (e.g., tool operation, pump power, purge gas in tanks, valve actuation, and bubbler tank level measurement) • Provide plant air receiver buffer capacity to make up difference between peak demand and compressor capacity • Provide plant air to instrument air subsystem for bubblers and valve actuation • Provide instrument air receiver buffer capacity to make up difference between peak demand and compressor capacity PSAR Section 3.5.2.7.14 lists the following design basis values of the plant and instrument air systems as: • 30-year design life with the exception of common replaceable parts (e.g., pumps) • Provide instrument air dried in regenerable desiccant beds to a dew point of no greater than -40°C (-40°F) and filtered to a maximum 40 micron (μ) particle size Thus, PSAR Section 9.7.1.1 appears inconsistent or not clearly correlated with PSAR Section 3.5.2.7.14, in that PSAR Section 3.5.2.7.14, lists three design basis functions for the plant and instruments air systems that are not listed among the two plant-and-instrument-air-system-related design basis elements in PSAR Section 9.7.1.1, specifically: • Provide small, advective flows of plant air for several RPF activities (e.g., tool operation, pump power, purge gas in tanks, valve actuation, and bubbler tank level measurement) • Provide plant air receiver buffer capacity to make up difference between peak demand and compressor capacity • Provide instrument air receiver buffer capacity to make up difference between peak demand and compressor capacity Additionally, PSAR Section 3.5.2.7.14, does not include two of the design bases given in PSAR Section 9.7.1.1 for the plant and instrument air systems. Also, PSAR Section 3.5.2.7.14 lists two design basis "values" that are not included among the design basis elements given in PSAR Section 9.7.1.1. While, PSAR Section 9.7.1.1 does indicate that PSAR Chapter 3.0 provides additional information on the utility systems design basis, it is not clear how each of the design basis functions and values in PSAR Section 3.5.2.7.14 supports or correlates with the design basis elements in PSAR Section 9.7.1.1. Additional information is needed to resolve inconsistent and incomplete design basis elements, functions, and values; inconsistent level of detail; unclear correlation of design basis functions and values with design basis elements. Lack of clarity in identifying which design basis elements belong to which system or subsystem make it difficult to identify system functional boundaries and to determine adequacy of the design basis.

No.	Request for additional information
RAI 9.7-1a	*Provide additional information to resolve or explain these apparent discrepancies in the design basis of utility systems.*

PSAR Sections 3.5.2.7 and 9.7 will be aligned with each other to enhance clarity and resolve discrepancies. The design basis bullets in PSAR Section 9.7.1.1 will be deleted, and the design basis description in PSAR Section 9.7.1 will be modified to cross-reference to the appropriate subsection of PSAR Section 3.5.2.7. Subsequent information in PSAR Section 9.7 will focus on the description of systems and components that satisfy the design basis functions.

No.	Request for additional information
RAI 9.7-1b	*Explain how NWMI will correct or clarify the apparent discrepancies in the PSAR.*

The design basis description in PSAR Section 9.7.1 will be modified to cross-reference to the appropriate subsection of PSAR Section 3.5. Subsequent information in PSAR Section 9.7 will focus on the description of systems and components that satisfy the design basis functions.

RAI 9.7-2

NUREG-1537, Part 1, Chapter 9, states that the applicant should include the design bases for each auxiliary system.

NWMI PSAR Section 9.7.1, "Utility Systems," states that the utility systems comprise the following subsystems:

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

PSAR Section 9.7.1.1 states, in part that one of the utility systems design bases is to provide chilled water to various process equipment at no greater than 10°C (50°F) during normal operations.

PSAR Section 3.5.2.7.17, "Process Chilled Water System," lists the process chilled water system design basis functions as follows:

- *Provide process chilled water loop for three secondary loops through plate-and-frame heat exchangers*
  - *One large geometry secondary loop in hot cell*
  - *One criticality-safe geometry secondary loop in hot cell*
  - *One criticality-safe geometry secondary loop in target fabrication area*
- *Provide monitoring of chilled water loops for loss of primary containment*
- *Provide cover gas to prevent flammable conditions*

PSAR Section 3.5.2.7.17 lists the process chilled water system design basis values as follows:

- *30-year design life with the exception of common replaceable parts (e.g., pumps)*
- *Chilled water to various process equipment at no greater than 10°C (50°F) during normal operations*

PSAR Section 3.5.2.7.18, "Facility Chilled Water System," lists the facility chilled water system design basis functions as follows:

- *Provide cooling media to heating, ventilation, and air conditioning (HVAC) system*
- *Supply HVAC system with cooling water that is circulated through the chilled water coils in air handling units*

PSAR Section 3.5.2.7.18 lists facility chilled water system design basis values as follows:

- *Provide cooling water at a temperature of 9°C (48°F) to the HVAC air-handling unit cooling coils*
- *30-year design life with the exception of common replaceable parts (e.g., pumps)*

PSAR Section 3.5.2.7.19, "Facility Heated Water System," lists facility heated water system design basis functions as follows:

- *Provide heated media to HVAC system*

No.	Request for additional information
	• *Supply the HVAC system with heated water that is circulated through the heated water coils in the air-handling units* *PSAR Section 3.5.2.7.19 lists facility heated water system design basis values as follows:* • *Provide heated water at a temperature of 82°C (180°F) to HVAC air-handling unit heating coils and reheat coil* • *30-year design life with the exception of common replaceable parts (e.g., pumps)* *PSAR Section 3.5.2.7, "Radioisotope Production Facility Specific System Design Basis Functions and Values," discusses the facility chilled water system (3.5.2.7.18) and the facility heated water system (3.5.2.7.19). However, neither of these systems is included among the utility systems listed in PSAR Section 9.7.1, or for which the design bases are given in PSAR Section 9.7.1.1. These systems are described briefly in PSAR Section 9.7.1.2.2, "Chilled Water." Additional information is needed to determine the relationship among these systems.* *Provide additional information to correct, explain or clarify this apparent discrepancy in discussions of the facility chilled water system and the facility heated water system.*

PSAR Sections 3.5.2.7 and 9.7 will be revised to enhance clarity and resolve discrepancies. The RPF system and subsystem designations will be used to align the utility systems. The design basis bullets in PSAR Section 9.7.1.1 will be deleted.

CHAPTER 11.0 – RADIATION PROTECTION AND WASTE MANAGEMENT

No.	Request for additional information
	Section 11.1 – Radiation Protection
(Applies to 11.1-1 through 11.1-2)	*Paragraph 20.1101(d) of 10 CFR Part 20 states that a constraint on emissions of radioactive material shall be established by licensees such that individual members of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent (TEDE) in excess of 10 millirem (mrem) per year from these emissions.*
RAI 11.1-1	*NUREG 1537, Part 2, Section 11.1.1, "Radiation Sources," states the applicant should identify models and assumptions that are used for predicting and calculating the dose rates and accumulative doses from such radionuclides as Argon-41, Nitrogen-16 and airborne radioactive particulates in both restricted, controlled, and unrestricted areas. The analysis should contain conservative best estimates of the predicted annual total doses to at least the following in the unrestricted areas: (1) the maximum exposed individual, (2) the nearest permanent residence, and (3) any location of special interest.* *NWMI PSAR Section 11.1.1.1.2, "Release of Airborne Radionuclides," provides Table 11-2, "Radionuclide Stack Release Source Term Input to COMPLY" to determine the constraint on release of airborne radionuclides to the environment.* *PSAR Section 1.2.2, "Consequences from the Operation and Use of the Facility," states the anticipated radionuclide inventory in the RPF is based on a weekly throughput of eight MURR targets processed a specified time after end of irradiation (EOI). Section 4.1.2.1 states the RPF is designed to have a nominal operational processing capability of one batch per week of up to 12 targets from MURR for up to 52 weeks per year and approximately 30 targets from the Oregon State University (OSU) TRIGA Reactor (OSTR) or a third university reactor for 8 weeks per year per reactor. It is not clear whether the calculation of airborne release of radionuclides is based on 8 or 12 MURR targets per week.*
RAI 11.1-1a	*Clarify the basis of the calculation of airborne release of radionuclides, i.e. which target processing throughput is the basis of the computation for maximum dose to the public described in the section.*

The calculations of airborne release in PSAR Section 11.1.1.1.2, "Release of Airborne Radionuclides," are based on the processing of eight targets at MURR. This section will be updated and described in the FSAR as part of the Operating License Application. The basis will be consistent with nominal operating conditions. The primary dose contributor is the xenon noble gas, and the offgas system is designed to retain the xenon to below release limits and bound the range of target processing.

RAI 11.1-1b	*If the nominal processing capability cited in PSAR Section 4.1.2.1 was not used for the computation, re-evaluate the maximum dose to the public.*

PSAR Section 11.1.1.1.2 conditions were slightly more conservative than those described in PSAR Section 4.1.2.1. PSAR Sections 4.1.2.1 and 11.1.1.1.2 operating conditions will be aligned in the FSAR as part of the Operating License Application.

RAI 11.1-2	*NUREG-1537, Part 2, Section 11.1.1, "Radiation Sources," states the applicant should present the best estimates of the maximum annual dose and the collective doses for major radiological activities during the full range of normal operations for facility staff and members of the public. The doses shall be shown to be within the applicable limits of 10 CFR Part 20.* *NWMI PSAR Section 11.1.2, "Radiation Protection Program," states NWMI management is committed to protecting RPF workers, the public, and environment from unacceptable exposure to radiation sources. The NWMI RPF administrative exposure limits have been set below the limits specified in 10 CFR Part 20 to ensure that regulatory radiation exposure limits are not exceeded and to emphasize ALARA principles. Table 11-5, "Estimated Radioisotope Production Facility Controlled and Restricted Area Dose Rates," provides dose rates for a variety of work areas to include the administrative spaces. However, a basis was not provided for does rates, designation of radiation areas, and assignment of dosimetry. This information is needed to determine ALARA and compliance with does limits in 10 CFR 20.*

No.	Request for additional information
RAI 11.1-2a	*Provide the basis of the dose rates established in Table 11-5 of the PSAR.*

The dose rates in PSAR Chapter 11.0, Table 11-5, were either based on actual shielding calculations or were the goals/ endpoints of the shielding analysis. This table will be updated in the FSAR as part of the Operating License Application when the final shielding design and calculations are completed. Areas identified as controlled access areas, restricted areas, radiation areas, and high radiation areas will be designated based definitions provided in 10 CFR 20, "Standards for Protection Against Radiation," and the predicted doses rates presented by the shielding analysis. Although the Radiation Protection Plan has not yet been developed (i.e., this plan will be supplied with the Operating License Application), dosimetry is anticipated to be required in any restricted area.

RAI 11.1-2b	*Describe if there are any areas within the RPF that would be designated radiation areas or high radiation areas.*

Radiation or high radiation areas are shown in PSAR Section 11.1.3.2, Figures 11-2 through 11-4, and discussed further in PSAR Section 11.1.5.5.

RAI 11.1-2c	*Describe the NWMI basis for assigning personnel dosimetry to staff.*

Personnel dosimetry will provide a means to measure, assess, and record personnel exposures to ionizing radiation from external sources. Exposure to external sources of radiation will be performed by individual monitory devices such as thermoluminescent dosimeters (TLD), optically stimulated luminescence (OSL), CR-39, activation foils, or direct reading pocket dosimeters. Use of personnel dosimetry will be required for all personnel entering the "restricted areas." Use of direct reading personnel dosimetry and criticality monitoring will be required for all personnel entering "high radiation areas" and "very high radiation areas."

RAI 11.1-2d	*Explain what is meant by "dose investigation level of 5 mSv/yr" in Section 11.1.2 of the PSAR.*

A dose investigation level of 5 millisievert (mSv)/year (yr) (500 millirem [mrem]/yr) is the total effective dose equivalent (TEDE) above which would trigger an investigation by the Radiation Protection staff to determine why an individual received such a dose equivalent. The routine TEDE to workers is not anticipated to approach this level. An investigation might entail interviews with the individual and the immediate supervisor, review of radiation work permits (or equivalent), review of procedures, review of ALARA (as low as reasonably achievable) approaches, and providing feedback to management with recommendations on how to proceed.

RAI 11.1-3	*Subparagraph 20.2001(a)(2) of 10 CFR Part 20 states that one of the methods that a licensee shall dispose of licensed material is decay-in-storage.* *NUREG 1537, Part 2, Section 11.1.5, "Radiation Exposure Control and Dosimetry," states the design bases of radiation shielding, ventilation, and remote handling and decontamination equipment should be planned so radiation doses are maintained ALARA and should be within the regulatory limits.* *NWMI PSAR Section 4.2.1.1, "Biological Shield Function," provides a general description of the biological shield and the intent to reduce radiation dose rates and accumulated doses to not exceed the limits of 10 CFR Part 20. It is not clear on the access controls for entering the RPF from the administrative support area. This information is need to ensure that radiation doses are maintained ALARA and within applicable limits of 10 CFR Part 20.*

No.	Request for additional information
RAI 11.1-3a	*Describe the requirements (dosimetry, personal protection equipment, etc.) and access controls for entering the RPF from the administrative support area.*

As shown in PSAR Section 11.1.5.5.2, Figure 11-5, the entire RFP is considered a "controlled area." PSAR Chapter 11.0, Figure 11-2, shows five doors from the outside of the RFP to entrances into the "restricted area." Each door will have two-credential access control (e.g., fob/PIN, fob/bio, or bio/PIN). The RPF Radiation Protection Program will require personnel to access assigned dosimetry and portable survey instrumentation (as needed based on the work authorized) from an as-yet unspecified location within the RPF administrative area before entering the restricted area. Portal survey monitoring will be in-place at the exit from the restricted area into the administrative area. The specifics on the type and instrument used will be described in the FSAR as part of the Operating License Application and will either be a control that allows standing passive detection or hand and foot monitors.

No.	Request for additional information
RAI 11.1-3b	*Describe the anticipated radiation levels and occupancy status of the corridor separating from the tank hot cell.*

This corridor, which is not continuously manned, will normally have a dose rate less than 0.5 mrem. However, this corridor provides access to the solid waste drums in the manipulator hot cells and therefore, at times, may have a higher dose rate and require radiation controls consistent with the planned activity.

No.	Request for additional information
RAI 11.1-4	*Subparagraph 20.1501(a)(2) of 10 CFR Part 20 states that each licensee shall make or cause to be made, surveys of areas, that are reasonable under the circumstances to evaluate the potential radiological hazards of the radiation levels and residual radioactivity detected.* *NUREG 1537, Part 2, Section 11.1.1, "Radiation Sources," states the applicant should present the best estimates of the maximum annual dose and the collective doses for major radiological activities during the full range of normal operations for facility staff and members of the public. The doses shall be shown to be within the applicable limits of 10 CFR Part 20.* *NWMI PSAR Section 4.3.2.2.6, "Radiological Hazards," describes the anticipated radionuclide inventory to be managed following irradiated target receipt, which excludes trace and decayed products of the irradiation process. The basis of this inventory is a weekly throughput of 8 MURR targets processed at a specified time following end of irradiation.* *PSAR Section 4.3.2.2.5, "Special Nuclear Material Description," describes processing a potential 30 irradiated targets per week from the OSTR. It is not clear how these additional irradiated targets impact the radiation dose to the public.* *Explain how these additional 30 irradiated targets impact the inventory and the maximum dose to the public described in Section 11.1.1.1.2 of the PSAR.*

At discharge from the Oregon State University TRIGA Reactor (OSTR) (or third) reactor, the 30 targets will have essentially the same amount of radioactivity as eight targets being discharged from MURR. Since the OSTR targets are not going to be received for 48 hours, the total radioactivity is significantly less than the eight MURR targets received in 8 hours. Therefore, other than grams of uranium, the radiation source for the 30 OSTR targets is lower.

No.	Request for additional information
(Applies to RAI 11.1-5 through 11.1-9)	*Paragraph 20.1101(a) of 10 CFR Part 20 states that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part.*
RAI 11.1-5	*NUREG 1537, Part 2, Section 11.1.1, "Radiation Sources," states the applicant should identify models and assumptions that are used for conservative best estimates of the predicted annual total doses to the maximum exposed individual.* *NWMI PSAR Section 13.2.1, "Maximum Hypothetical Accident," shows in Tables 13-18, Table 13-19, and Table 13-20, the distance-dependent inhalation, exposure, and total receptor MHA doses, respectively, versus distance from the RPF stack for an assumed bounding 2-hr exposure. NWMI provides the TEDE due to the MHA for the maximally exposed individual but does not provide the Committed Dose Equivalent (CDE) to the thyroid.* *Provide the CDE to the thyroid from the presented scenario and the impact to the public.*

The committed dose equivalent (CDE) to the thyroid was 97 percent of the impact to the public in the MHA. The MHA is being deleted from the PSAR, consistent with the response to RAI G-3.

No.	Request for additional information
RAI 11.1-6	*NUREG-1537, Part 1, Section 11.1.3, "ALARA Program," states the facility's ALARA program should require that radiation dose received by facility staff and the public are maintained ALARA, economic factors having been taken into account.* *NWMI PSAR Section 11.1.2, "Radiation Protection Program," states the NWMI administrative exposure limits have been set below the limits specified in 10 CFR Part 20 to ensure that regulatory radiation exposure limits are not exceeded and to emphasize ALARA principles. Table 11-5, "Estimated Radioisotope Production Facility Controlled and Restricted Area Dose Rates," states the dose rate in the administration and support area and utility area will be 0 mrem/hr. It is not clear how NWMI will demonstrate these areas will be maintained at 0 mrem/hr (unrestricted area).* *Describe how NWMI will demonstrate that the administration and utility areas are no different than unrestricted areas.*

PSAR Section 11.1.2, Table 11-5, provides estimated dose rates based on the RPF design. Although a dose rate of zero may not be achievable in the controlled areas, this is the goal. As stated in PSAR Section 11.1.5.5.2, an area monitoring program will be established in the controlled area to demonstrate compliance with public exposure limits in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 11.1-7	*NUREG-1537, Part 1, Section 11.1.4, "Radiation Monitoring and Surveying," states that a complete range of radiation monitoring and sampling equipment, appropriate to the facility, should be employed throughout the facility, including equipment employed by experimental and operations support personnel, including remote area monitors.* *NWMI PSAR Section 11.1.5.5.2, "Controlled Area," states that area monitoring will demonstrate compliance with public exposure limits for visitors. Area monitoring equipment is not described in Section 11.1.4 of the PSAR.* *Provide a description of the area monitoring equipment NWMI intends to use to demonstrate compliance with public exposure limits for visitors.*

Details on the area monitoring program will be provided in the FSAR as part of the Operating License Application. Area monitoring is anticipated to comprise a combination of passive (e.g., TLD or OSL monitors changed out monthly or quarterly) and active (e.g., energy-compensated G-M detector systems with local and remote monitoring capability) monitoring systems located at points in the controlled area that would provide reasonable assurance that radiation areas are not present in the controlled area. The selection of specific instrumentation, range of detection, and alert/alarm setpoints will be consistent with the intent to detect radiation areas where they should not be and alert personnel to this changing condition.

No.	Request for additional information
RAI 11.1-8	*NUREG-1537, Part 1, Section 11.1.7, "Environmental Monitoring," states the methods and techniques to sample analyze the radiological effect of facility operation should be complete, applicable, and of sufficient validity that the environmental impact can be unambiguously assessed.* *NWMI PSAR Section 4.1.3.6.1, "Waste Handling System Process Overview," states that a portion of the low-dose liquid fraction is expected to be suitable for recycle to selected hot cell systems as processed water. Water that is not recycled will be adjusted and then mixed with an adsorbent material in 55-gallon drums. Further, waste streams will be containerized, stabilized as appropriate, and shipped offsite for treatment and disposal. Figure 4-23, "Low-Dose Liquid Waste Disposition Process," and Figure 4-25, "Low Dose Liquid Waste Evaporation Facility Location," describe an evaporation system and facility. It is not clear how the evaporation effluent is controlled from the low-dose wastewater tanks.* *Describe the endpoint of the evaporation effluent from this facility and clarify whether any evaporate will be exhausted to the environment. Explain, if NWMI's intends to exhaust to the environment, what type of effluent sampling and documentation of results will be performed.*

The final facility design strategy is to route the air stream from the evaporation tanks into the Zone I exhaust system. The Zone I exhaust stack will have continuous monitoring.

RAI 11.1-9	*NWMI PSAR Section 4.3.2.2.5, "Special Nuclear Material Description," states weekly cask shipments from the OSTR will consist of two transport casks, not to exceed a total of 30 targets. Weekly cask shipments from the MURR will consist of two transport casks containing four targets. PSAR Section 4.2.3.1, "Initial Source Term," states the photon source strength for the NWMI shielding analysis was determined based on the activity associated with eight MURR irradiated targets. Additional information is required to understand the process of target receipt and processing and the impact to the shielding analysis.* *Clarify the anticipated target receipt inventory and explain how the additional irradiated targets from OSTR will bear on the shielding analysis.*

At discharge from OSTR (or third) reactor, the 30 targets will have essentially the same amount of radioactivity as eight targets being discharged from MURR. Since the OSTR targets are not going to be received for 48 hours due to transportation time, their total radioactivity is significantly less than the eight MURR targets received in 8 hours. The OSTR targets at receipt are estimated to have only approximately 40 percent of the MURR targets at receipt. Therefore, other than grams of uranium, the radiation source for the 30 OSTR targets is lower and is not used in the shielding analysis.

RAI 11.1-10	*Paragraph 20.1101(d) of 10 CFR Part 20 states that a constraint on emissions of radioactive material shall be established by licensees such that individual members of the public likely to receive the highest dose will not be expected to receive a TEDE in excess of 10 millirem (mrem) per year from these emissions.* *NUREG-1537, Part 2, Section 11.1.1, "Radiation Sources," Acceptance Criteria, states, in part, that all sources of radiation should be discussed by the applicant. This discussion should include the physical and chemical form, type (e.g., neutron, gamma), curie strength or exposure rates, energy level, encapsulation (sealed or unsealed), use, storage conditions and locations, and planned program for disposal of all radioactive material subject to the reactor license.* *NWMI PSAR, Section 4.4.1, "Processing of Irradiated Special Nuclear Material," provides an overview of the uranium recovery system and states it is sized to purify approximately 22 kg/week for recycle to the target fabrication system. Section 4.4.1 also states the 22 kg/week is based on processing 30 targets.* *PSAR Section 1.2.2, "Consequences from the Operation and Use of the Facility," states the anticipated radionuclide inventory in the RPF is based on a weekly throughput of eight MURR targets processed at a specified time after the end of irradiation. PSAR Section 4.1.2.1 states the RPF is designed to have a nominal operational processing capability of one batch per week of up to 12 targets from MURR for up to 52 weeks per year and approximately 30 targets from the OSTR or a third university reactor for 8 weeks per year per reactor.*

No.	Request for additional information
RAI 11.1-10a	*Identify whether or not this recyclable material is a part of the inventory described in Table 1.1.*

The SNM being recycled to target fabrication is included in PSAR Chapter 1.0, Table 1-1. The recyclable material is held in the U decay tanks until needed for processing into targets.

No.	Request for additional information
RAI 11.1-10b	*Clarify the radionuclide inventory, maximum dose to the public, and other radiological hazards based on planned processing capabilities.*

The response to RAI 11.1-4 explains why the OSTR targets are not used for the radioactive source term calculations. The basis for the maximum dose to the public is derived from NWMI-2013-CALC-011, *Source Term Calculations*. As discussed in PSAR Chapter 13.0, the MHA uses 12 MURR targets to estimate the gaseous source term. Due to batch size and receipt timing at the RPF, eight MURR targets (8 hours after discharge) bound the liquid accidents.

No.	Request for additional information
	Section 11.2 – Radioactive Waste Management
RAI 11.2-1	*NUREG-1537, Part 1, Section 11.2, "Radioactive Waste Management," introduces the expectations for the content of the PSAR. It includes the statement, "The magnitude and nature of the effort required should depend upon the size and complexity of both the reactor facility and its utilization programs. Therefore, the nature and details of the radioactive waste management program should also be commensurate with those factors." Based on the annual generation rate of high-dose Class C waste presented in PSAR Table 11-6, "Waste Produced in the Radioisotope Production Facility," and the capacity of the container proposed for Class C waste presented in PSAR Table 4-17, "Waste Container Geometric Data," there will be one or more Class C shipments for disposal every week. The complexity of waste processing is indicated by both the number of chemical operations and adjustments required, as discussed in PSAR Sections 4.1.3.6, "Waste Handling," and 9.7.2, "Control and Storage of Radioactive Waste," and by the number of potential accidents involving waste operations, as identified in NWMI PSAR Chapter 13, "Accident Analysis Methodology and Preliminary Hazards Analysis." Given the amount of waste and the complexity of waste management operations, the "...nature and details of the radioactive waste management program..." should be well explained.* *NUREG-1537, Section 11.2.1, "Radioactive Waste Management Program," delineates specific expectations for the description of the radioactive waste management program. NUREG-1537, Part 2, Section 11.2.1, "Radioactive Waste Management Program," states that factors addressed by the applicant should include organization of the management function, program staffing and position descriptions, and program personnel responsibilities and qualifications as discussed in the format and content guide.* *PSAR Section 11.2.1.3.2, "Waste Management Lead," identifies the individual responsible to the Plant Manager for waste management activities, including self-assessments. PSAR Chapter 12, "Conduct of Operations," does not include the position "Waste Management Lead," nor is it on any of the management figures. PSAR Section 11.1.2.1.3, "Radiation Protection Manager," states that the Radiation Protection Manager will be responsible for overseeing handling and disposal of radioactive wastes. This PSAR inconsistency results in uncertainty regarding the management function, program staffing and position descriptions, and program personnel responsibilities.*

No.	Request for additional information
RAI 11.2-1a	*Provide a more complete and comprehensive description of the radioactive waste management organization and responsibilities within the NWMI management structure.*

The Waste Management Lead (not the Radiation Protection Manager) has responsibility for oversight, handling, and disposal of radioactive wastes. PSAR Section 11.1.2.1.3 will be modified to delete "overseeing handling and disposal of radioactive wastes;" this information will be added to PSAR Section 11.2.1.3.2.

Radioactive waste management responsibilities within the NWMI management structure include:

- Implements waste management policy
- Develops waste management procedures for the processing, packaging, and shipment of radioactive waste from the facility
- Processes, packages, and ships radioactive waste from the facility
- Provides technical input to the design of equipment and processes
- Provides technical input to the waste management training program
- Establishes and maintains contractual relationships with waste disposal sites and radioactive waste carriers
- Maintains working knowledge of the waste acceptance criteria, standards, guides, and codes with respect to waste disposal
- Conducts self-assessments of waste management practices and compliance with procedures in accordance with the waste management self-assessment program

These responsibilities will be added to PSAR Section 11.2.1.3.2.

RAI 11.2-1b	*Clarify the line of authority for radioactive waste management.*

The waste management program will be coordinated with the radiation protection program, and program management will report to the Plant Manager. PSAR Section 11.1, "Radiation Protection," describes the program and procedures for controlling and assessing radioactive exposures associated with radioactive sources, including radioactive waste streams. The goal of the waste management program is to minimize waste generation, minimize exposure of personnel, and to protect the public and environment. An official charter describing the authority, duties, and responsibilities of personnel in the waste management organization will be described in the FSAR as part of the Operating License Application.

RAI 11.2-2	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information maintained by the applicant be complete and accurate in all material respects.* *NUREG-1537, Part 1, Section 11.2.2, "Radioactive Waste Controls," states that the applicant should identify and discuss the plans for management of all forms of radioactive waste.* *NWMI PSAR Section 11.2.3, "Release of Radioactive Waste," Table 11-6 presents estimates of the annual generation rates of specific waste streams, but without adequate specificity to ascertain what these generation rates include and whether all potential waste streams have been considered.* *PSAR Section 9.7.2.2.1 states that caustic solution is added, as needed, to the high-dose liquid stream.*
RAI 11.2-2a	*Clarify if the values presented in PSAR Table 11-6 include this added volume of caustic solution.*

Caustic soda (NaOH) is included in the waste volume estimates in PSAR Chapter 11.0, Table 11-6.

RAI 11.2-2b	*Confirm if mixing ratios of waste liquid to solidification agent have been included in PSAR Table 11-6 values for solidified waste volume.*

The solidification agent content is included in the waste volume estimates in PSAR Chapter 11.0, Table 11-6.

No.	Request for additional information
RAI 11.2-2c	*PSAR Section 4.4.1.4, "Special Nuclear Material Description," contains references to drains to geometrically safe locations. There appears to be no discussion of how these streams are controlled after collection and whether they will be treated as waste.* *Provide a discussion of how these streams are controlled after collection, whether they are treated as waste, and if so, whether that volume is included in PSAR Table 11-6.*
Material discussed in PSAR Section 4.4.1.4 will not be considered waste. The material will be returned to the U recovery and recycle system, purified, and reused. This material is not included in PSAR Chapter 11.0, Table 11-6.	
RAI 11.2-2d	*PSAR Section 4.1.2.1, "Process Design Basis," states that the RPF is designed to have a nominal operational processing capability of one batch per week of up to 12 targets from MURR for up to 52 weeks per year and approximately 30 targets from the OSTR or a third university reactor for 8 weeks per year per reactor." Therefore, the nominal operational processing capability of the RPF is 1,104 targets per year. This section also states that the overall process functional requirement for the handle waste function is, "Providing the capability to handle waste generated from processing up to 120 irradiated targets per month." PSAR Table 11-6, Waste Produced in the Radioisotope Production Facility, is based on processing only 8 MURR targets per week, or about 38% of the RFP nominal operational processing capability.* *Given these statements, justify that, the data presented in PSAR Table 11-6, which is based on processing only 8 MURR targets per week is reasonable basis for waste production in the RPF.*
There was no basis identified for the values listed in PSAR Chapter 11.0, Table 11-6. Waste volume projections are based on the composite values from the MURR and OSTR mass balance calculations that assume an eight-target/week MURR processing rate plus a 30-target/week OSTR processing rate and will bound the planned operations.	
RAI 11.2-3	*Paragraph 20.1101(d) of 10 CFR Part 20 states that a constraint on emissions of radioactive material shall be established by licensees such that individual members of the public likely to receive the highest dose will not be expected to receive a TEDE in excess of 10 millirem (mrem) per year from these emissions.* *NUREG-1537, Part 2, Section 11.2.1, "Radioactive Waste Management Program," states the program should be designed to address all technical and administrative functions necessary to limit radiation hazards related to radioactive waste.* *PSAR Section 4.3.4.1, "Process Description," states that the overall process concept for radioactive noble gases is to delay gas release so that decay will reduce the radioisotope content sufficiently to allow the decayed noble gases to be safely discharged to the stack. The PSAR states that a 60-day period will effect that delay, driven by Xenon-133. The PSAR does not address how longer-lived radioactive isotopes are limited to reduce radiation hazards related to radioactive waste.* *Describe how NWMI intends to address longer-lived radioisotopes that generate hundreds or thousands of curies of activity in this process, identified in Table 4-33, "Target Disassembly In-Process Radionuclide Inventory," such as Promethium-143 and 145, Cesium-137, Cerium-144, etc.*
The nongaseous long-lived radioisotopes are contained in the high-dose liquid waste stream that is solidified and eventually sent offsite for disposal.	
RAI 11.2-4	*Section 50.9, "Completeness and accuracy of information," of 10 CFR Part 50 requires that information submitted, or information required to be maintained by the applicant be complete and accurate in all material respects.* *NWMI PSAR Section 11.2.2.1, "Waste Designation," states that the RPF will generate Class A, B, and C low-level radioactive waste. Table 11-6, "Waste Produced in the Radioisotope Production Facility," identifies several items as Class C waste. High-dose and other types of waste are identified as Class C, and annual generation volumes are provided.*

No.	Request for additional information
	In its November 2, 2015, response to environmental RAI WM-R-1, (ADAMS Accession number ML15328A071) NWMI stated that high-dose and encapsulated waste are projected to be Class B waste, and provided bounding annual generation rates (in kilograms per year). Additionally, in the response to RAI WM-R-1, NWMI stated that volume reduction has the potential to change the disposed waste classification from Class B to class C as a result of optimization activities, and results will be described in the Operating Permit Application, but did not identify any specific generation rates of Class C waste. Additional information is needed to resolve this apparent discrepancy between Table 11-6 and NWMI's response to RAI WM-R-1. *Provide additional information to resolve the apparent discrepancy of whether Class C waste will be produced, or may potentially be produced, in Table 11-6 and NWMI's response to RAI WM-R-1.*

The solidified high-dose liquid waste from the RPF will be either Class B or Class C waste. As a result of reducing the waste volume and minimizing disposal costs, the liquid waste concentration endpoint may result in a change in the final waste classification from Class B to Class C.

RAI 11.2-5	*NUREG-1537, Part 1, Section 11.2.2, "Radioactive Waste Controls," states that the applicant should describe the plans and procedures for managing solid radioactive wastes generated during operations, research, and utilization of the reactor. This description should include how solid radioactive materials are generated and where they enter the waste control and treatment systems. Additionally, NUREG-1537, Part 1, Section 11.2.3, "Release of Radioactive Waste," states that the applicant should identify all radioactive materials for which transfer to other parties for disposal is planned.* *NWMI PSAR Table 19-13, "Solid Waste Produced at the Radioisotope Production Facility," includes an estimate of 40,000 L of potentially contaminated waste (e.g., decontamination materials, PPE). PSAR Section 11.2 does not identify personal protective clothing and dry active waste from decontamination and maintenance activities as a waste stream and does not provide information regarding the estimated amount or handling of dry active wastes generated at the RPF. It is not clear whether this waste stream is included in PSAR Table 11-6, "Waste Produced in the Radioisotope Production Facility." PSAR Chapters 4 and 9 do not contain information regarding the proposed collection, volume reduction, packaging or storage of this waste steam. PSAR Section 9.7.2.2.8, "Waste Staging and Storage Building," provides the volume of the Waste Staging and Shipping Building but provides no details regarding the processes occurring within that building.* *More information is needed to provide reasonable assurance that radioactive wastes will be controlled at all times in a manner that protects the environment and the health and safety of the facility staff and the public.*
RAI 11.2-5a	*Provide the basis for the estimate of the generation rate and discuss the processes used to minimize the volume stored prior to disposal, and include the waste stream in PSAR Table 11-6.*

The estimates for the laboratory facilities or facility support waste volume projections in PSAR Chapter 19.0, Table 19-13, have no definitive basis and will be further defined in the NWMI Operating License Application. The estimated facility support waste values in Table 19-13 will be added to PSAR Chapter 11.0, Table 11-6. Shipment will be made in a timely manner to minimize the inventory of stored waste at the RPF. The Class A waste will typically be shipped monthly or when a trailer of drums has accumulated. The high-dose waste will be decayed to meet shipping cask limits and then transported to the disposal site. Solid waste encapsulated in cement (55-gal drums) will typically be shipped when 10 drums are available to fill the shipping cask. The other (low volume) waste streams will be disposed of shortly after.

RAI 11.2-5b	*Clarify which waste processing steps are accomplished in the RPF and which are accomplished in the Waste Staging and Shipping Building.*

The Waste Staging and Shipping Building does not contain any waste processing steps. The building will be used to store incoming drums (consumables) and to store and stage filled drums (Class A waste) prior to shipment to an approved disposal facility.

No.	Request for additional information
	Section 11.3 – Respiratory Protection Program
RAI 11.3-1	*Subparagraph 20.2001(a)(1) of 10 CFR Part 20 states that a licensee shall dispose of licensed material by transfer to an authorized recipient, as provided in 10 CFR 20.2006.* *The ISG Augmenting NUREG-1537, Part 2, Section 11.3, "Respiratory Protection Program," states that the applicant shall describe surveillance requirements, including preventive and corrective maintenance and performance testing, to ensure that the ventilation and containment systems operate when required and are within their design specifications.* *NWMI PSAR Section 4.1.4.5, "Irradiated Target Receipt Area," states that the irradiated target receipt bay will be designed to operate as a Zone II airspace during fuel element unloading procedures and when the hot cell cover block is removed for maintenance. Table 4-5, "Facility Areas and Respective Confinement Zones," in Section 4.1.4.3 states the bay will operate as a Zone IV area. More information is needed to understand zone management to determine the adequacy of confinement and radioactive material contamination control.*
RAI 11.3-1a	*Describe how these zonal differences are managed, how the changes will be evaluated, and posted.*

The preliminary design and analysis of the RPF ventilation system ensures that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur during normal operational states and to mitigate the consequences of design basis accidents (e.g., maintaining a series of cascading pressure zones to draw air from cleanest area to the most contaminated area of the RFP). In addition, the preliminary design indicates that the distribution and concentrations of any airborne radionuclides are limited by operation of the ventilation system so that during the full range of facility operations, no potential occupation exposures would exceed the design bases (e.g., 10 CFR 20) derived in PSAR Chapter 11.0. The pressure relationship between the four ventilation zones and ambient atmospheric pressure is presented below.

Zone IV will be the cleanest zone and is slightly positively pressurized with respect to atmosphere. Zone IV is independent of the other three ventilation zones. Zones I, II, and III will potentially be contaminated areas, with Zone III being the cleanest of the potentially contaminated areas, and each subsequent zone being more contaminated and having lower pressures, as shown below:

$$P_{\text{Zone I}} < P_{\text{Zone II}} < P_{\text{Zone III}}$$

The Irradiated Target Receipt Area and the Irradiated Target Truck Bay are two different areas in the RPF. The truck bay is where trailers will be rinsed before entering the receipt area, where the cask will be removed from the trailer. The Irradiated Target Truck Bay is Zone IV, while the Irradiated Target Receipt Area is normally Zone III. Details of how the Irradiated Target Receipt Area will transition between Zone II and III during operating/maintenance activities will be provided in the FSAR as part of the Operating License Application.

RAI 11.3-1b	*Describe the occupancy status when operating as a Zone II area.*

The Irradiated Target Receipt Area will be occupied by at least two workers during cask sampling and movement of the cask to the cask transfer tunnel.

No.	Request for additional information
RAI 11.3-2	*Section 20.1902 of 10 CFR Part 20 establishes posting requirements for radiation areas, airborne radioactivity areas and areas or rooms in which licensed material is used or stored.* *The ISG Augmenting NUREG-1537, Part 2, Section 11.3, states that the applicant will install "appropriately sized ventilation and containment systems in areas of the plant identified as having potential airborne concentrations of radionuclides that could exceed the occupational derived air concentration values specified in 10 CFR Part 20, "Standards for Protection against Radiation," Appendix B, "Annual Limits on Intake (ALI) and Derived Air Concentrations (DAC) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."* *NWMI PSAR Section 9.1.2, "System Description," states that the facility ventilation system will maintain a series of cascading pressure zones to draw air from the cleanest areas of the facility to the most contaminated areas. Zone IV will be a clean zone that is independent of the other ventilation zones. Zone III will be the cleanest of the potentially contaminated areas, with each subsequent zone being more contaminated and having lower pressures. More information is needed to determine the adequacy of the confinement and radioactive material contamination control.*
RAI 11.3-2a	*Describe the process whereby NWMI will maintain these ventilation zones in regards to contamination.*

The preliminary design and analysis of the RPF ventilation system ensures that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur during normal operational states and to mitigate the consequences of design basis accidents (e.g., maintaining a series of cascading pressure zones to draw air from cleanest area to the most contaminated area of the RFP). In addition, the preliminary design indicates that the distribution and concentrations of any airborne radionuclides are limited by operation of the ventilation system so that during the full range of facility operations, no potential occupation exposures would exceed the design bases (e.g., 10 CFR 20) derived in PSAR Chapter 11.0. The pressure relationship between the four ventilation zones and ambient atmospheric pressure is presented below.

Zone IV will be the cleanest zone and is slightly positively pressurized with respect to atmosphere. Zone IV is independent of the other three ventilation zones. Zones I, II, and III will potentially be contaminated areas, with Zone III being the cleanest of the potentially contaminated areas, and each subsequent zone being more contaminated and having lower pressures, as shown below:

$$P_{\text{Zone I}} < P_{\text{Zone II}} < P_{\text{Zone III}}$$

The RFP will maintain ventilation zones in regard to contamination and radioactive material contamination control. In the context of describing ventilation zones, it is common to separate zones commensurate with the work being performed, radioactive materials present, and the potential for radioactive contamination within each zone. Details of the facility ventilation system provided in PSAR Section 9.1.2 were intended to meet this philosophy. PSAR Section 3.1 provides the codes and standards that the ventilation system will be designed to.

The RPF preliminary design of ventilation and containment systems was developed to ensure the sufficiency of the principal design criteria, design bases, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. The final facility design of the ventilation and confinement system will be described in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 11.3-2b	*Describe the gradients of potential contamination separating each zone.*

The preliminary design and analysis of the RPF ventilation system ensures that no uncontrolled release of airborne radioactive material to the unrestricted environment could occur during normal operational states and to mitigate the consequences of design basis accidents (e.g., maintaining a series of cascading pressure zones to draw air from cleanest area to the most contaminated area of the RFP). In addition, the preliminary design indicates that the distribution and concentrations of any airborne radionuclides are limited by operation of the ventilation system so that during the full range of facility operations, no potential occupation exposures would exceed the design bases (e.g., 10 CFR 20) derived in PSAR Chapter 11.0. The pressure relationship between the four ventilation zones and ambient atmospheric pressure is presented below.

Zone IV will be the cleanest zone and is slightly positively pressurized with respect to atmosphere. Zone IV is independent of the other three ventilation zones. Zones I, II, and III will potentially be contaminated areas, with Zone III being the cleanest of the potentially contaminated areas, and each subsequent zone being more contaminated and having lower pressures, as shown below:

$$P_{\text{Zone I}} < P_{\text{Zone II}} < P_{\text{Zone III}}$$

Details of the facility ventilation system provided in PSAR Section 9.1.2 were intended to meet this philosophy. PSAR Section 3.1 provides the codes and standards that the ventilation system will be designed to.

The RPF preliminary design of ventilation and containment systems was developed to ensure the sufficiency of the principal design criteria, design bases, general arrangement, and approximate dimensions sufficient to provide reasonable assurance that the final design will conform to the design basis. The final facility design of the ventilation and confinement system will be described in the FSAR as part of the Operating License Application.

RAI 11.3-3	*Section 20.1701 of 10 CFR Part 20 states that the applicant shall use, to the extent practical, process or other engineering controls to control radioactive material in air.* *The ISG Augmenting NUREG-1537, Part 2, Section 11.3, states that the applicant will describe the criteria for the ventilation and containment systems, including minimum flow velocity at openings in these systems, maximum differential pressure across filters, and types of filters to be used.* *NWMI PSAR Section 9.1, "Heating, Ventilation, and Air Conditioning Systems," states that the RPF design features ensure that airflow and relative pressure will prevent inadvertent diffusion or other uncontrolled release of airborne radioactive material from the RPF. The facility is also designed and operated to ensure that no uncontrolled release of airborne radioactive material to the unrestricted environment can occur. More information is needed to determine the adequacy of the confinement and radioactive material contamination control.* *Describe the criteria for the ventilation and containment systems, including minimum flow velocity at openings in these systems, maximum differential pressure across filters, and types of filters to be used.*

PSAR Section 3.1 provides the codes and standards that the ventilation system will be designed to. The detailed ventilation system criteria, including minimum flow velocity at openings in each zone, maximum differential pressure across filters, and types of filters to be used (e.g. HEPA, high-efficiency gas adsorption [HEGA]), will be provided in the FSAR as part of the Operating License Application.

CHAPTER 12.0 – CONDUCT OF OPERATIONS

No.	Request for additional information
	Appendix 12C, Section 2.2 – Quality Control Program
RAI C.2.2-1	*Subparagraph 50.34(a)(7) of 10 CFR Part 50 requires a description of the quality assurance (QA) program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility.* *Section 70.4 of 10 CFR Part 70 defines IROFS as "structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in §70.61 or to mitigate their potential consequences." The ISG augmenting NUREG-1537, Part 1, states that meeting the performance requirements in 10 CFR 70.61 are not required by regulation, but that their use as accident consequence and likelihood criteria could be found acceptable by the NRC staff.* *NWMI's QA program defines QA Level 1 items as including those items in which failure or malfunction could indirectly result in a condition that adversely affects workers, the public, and/or environment as described in 10 CFR 70.61. The failure of a QA Level 2 item, in conjunction with the failure of an additional item, could result in a high or intermediate consequence. All building and structure IROFS associated with credible external events are QA Level 1. QA Level 1 items also include those attributes of items that could interact with IROFS due to a seismic event, and result in high or intermediate consequences as described in 10 CFR 70.61.* *NWMI PSAR description of QA Level 1 IROFS appears to only include structures. Additional information is needed to understand the applicability of the QA plan to other IROFS besides structures.* *Clarify the scope of IROFS addressed by the applicant's QA program and describe the applicability of different QA levels to IROFS to factors including systems, equipment, components and activities.*

The QA Plan will be revised to clarify the difference between QL-1 and QL-2. Section 3.5.1.3 was modified to reflect the changes in the quality level definitions (Attachment A). The basis for the difference in QL-1 and QL-2 is a graded approach to quality, by which the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance. The graded approach permits the implementing organization to focus resources on those activities that are deemed, by qualitative analysis, to reduce the associated risks and hazards. The activities and tasks are performed in accordance with approved implementing procedures.

The graded approach to quality is a process by which the level of analysis, documentation, and actions necessary to comply with a requirement is commensurate with the safety significance. A graded approach permits the implementing organization to focus resources on those activities that are deemed, by qualitative analysis, to reduce the associated risks and hazards.

Activities and tasks are performed in accordance with the quality level definitions.

- Quality Level 1 will be applied to IROFS (SSCs and activities). IROFS are QL-1 items in which failure or malfunction could directly result in a condition that adversely affects workers, the public, and/or environment, as described in 10 CFR 70.61.
- Quality Level 2 will be applied to safety SSCs that are non-QL-1 SSCs. Some of the required characteristics may be examined less rigorously than for QL-1 items.
- Quality Level 3 items include those items that are not classified as QL-1 or QL-2. QL-3 items are controlled in accordance with standard commercial practices.

No.	Request for additional information
	Appendix 12C, Section 2.3 – Design Control
RAI C.2.3-1	*ANS 15.8 states that the need for or the use of qualification tests shall be defined in a formal test plan that shall include appropriate acceptance criteria and shall demonstrate the adequacy of performance under conditions that simulate the most adverse design conditions. Test results shall be documented and evaluated by the responsible design organization to assure that test requirements have been met. NWMI's PSAR did not provide how this criteria will be met and how it is incorporated on the QA program.* *Clarify whether NWMI intends to perform qualification testing or the need to demonstrate the adequacy of performance of systems, structures or components under conditions that simulate the most adverse design conditions. For example, if it is required for any type of qualification testing for an identified IROFS during the construction period, how will this be stated in the QA program.*

Qualification testing will be performed to demonstrate the adequacy of performance of SSCs under conditions that simulate the most adverse design conditions. Formal testing or analysis will be required to verify conformance of designated SSCs to specified requirements and demonstrate satisfactory performance for service or to collect data in support of design or fabrication. Test results will be documented and evaluated by a responsible authority to ensure that test requirements have been satisfied. Computer programs used for operational control will be tested in accordance with an approved verification and validation plan and will demonstrate required performance over the range of operation of the controlled function or process.

RAI C.2.3-2	*ANS 15.8 states that where a significant design change is necessary because of an incorrect design, the design process and verification procedure should be reviewed and modified as necessary.* *NWMI's QA program does not provide details on how design changes are captured, documented, and monitor during the construction period and how this will transition to the operation period.* *Clarify whether NWMI intends to provide that when a significant design change is necessary because of an incorrect design, the design process and verification procedure should be reviewed and modified as necessary. For example how will any design change be documented or tracked during the construction period, and how will this be documented thru the QA program.*

Engineering change control procedures (NWMI-ENG-PRO-002, *Engineering Change Control*) have been developed for the RPF design and construction to ensure that modifications to safety-related SSCs, or computer codes, will be based on a defined "as-exists" design. Changes to verified designs will be documented, justified, and subject to design control measures commensurate with those applied to the original design. The control measures will include assurance that the design analyses for the SSC, or computer code, are still valid. Where a significant design change is necessary because of an insufficient design, the design process and verification procedure will be reviewed and modified as necessary.

No.	Request for additional information
	Appendix 12C, Section 2.11 – Test Control
RAI C.2.11-1	*ANS 15.8 states, in part, that testing shall include prototype qualification tests, proof tests prior to installation, and functional tests.* *NWMI's QA program does not provide the extent or applicability of the QA program to prototype testing pre and post installation.* *Clarify the scope of testing activities (e.g., prototype qualification tests, proof tests prior to installation, functional tests) the applicant intends to conduct under the QA program.*

Testing activities (e.g., prototype qualification tests, proof and functional tests) will be completed under the QA program of the organization that is completing the work. For example, the LEU prototypic target fabrication will be completed under the Oak Ridge National Laboratory, High Flux Isotope Reactor QA program, and the fabrication of LEU targets for irradiation and processing at MURR will be completed under the MURR QA program.

No.	Request for additional information
RAI C2.11-2	*NWMI PSAR, Chapter 12, Section C2.11.2.4, "Computer Software," states, in part, that "Testing shall include verification tests, hardware integration tests, in use tests, or other tests as specified by the customer, as appropriate."* *NWMI's QA program does not provide the extent or applicability of the QA program to software applicability on these tests or the authority responsible for such verification.* *Clarify whether the computer software testing is to be done by another entity other than NWMI, and whether these software testing controls are applicable to NWMI.*

Computer software testing will be required by all suppliers to verify and provide evidence of the quality of their software products. In addition, methods to control and approve supplier-generated documents will be established. Based on the complexity of the product and importance to safety, NWMI will independently verify the quality of supplier's product using source surveillances, inspections, audits, and review of supplier's nonconformances, dispositions, waivers, and corrective actions. NWMI-QA-PRO-029, *Testing*, identifies the process by which computer software testing will be completed.

The software requirements review will be performed at the completion of the software requirements documentation and will ensure that the requirements are complete, verifiable, consistent, and technically feasible. The review will also ensure that the requirements will result in feasible and usable code.

During software testing, the design as implemented in code will be exercised by executing the test cases. Failure to successfully execute the test cases will be reviewed to determine if modifications of the requirements, design, implementation, and/or test plans and cases are required. The code will be validated and verified to ensure adherence to the requirements and that the software produces correct results for the test cases. To evaluate technical adequacy, the software test case results can be compared to results from alternative methods, including analysis without computer assistance, experiments and tests, standard problems with known solutions, or confirmed published data and correlations.

No.	Request for additional information
	Appendix 12C, Section 2.15 – Control of Nonconforming Items
RAI C2.15-1	*Part 21, "Reporting of Defects and Noncompliance," of 10 CFR states that this part applies, except as specifically provided other in Parts 31, 34, 35, 39, 40, 60, 61, 63, 70, or Part 72 of this chapter.* *NWMI PSAR, Chapter 12, Section C2.15.2.4, "Nonconforming Condition," states, "When required by contract or specification, the nonconforming conditions shall be transmitted to the customer for evaluation as a potentially reportable condition under 10 CFR 21."*
RAI C2.15-1a	*Clarify the applicability of 10 CFR Part 21 to the applicant. In the statement above from the PSAR, it appears the applicability of 10 CFR Part 21 requirements apply to another entity.*

The requirements in 10 CFR 21, "Reporting of Defects and Noncompliance," apply to NWMI as the responsible party for the RPF Construction Permit Application. This will also be true for the Operating Permit Application. PSAR Chapter 12.0, Section C2.15.2.4, while correctly stating that NWMI will notify a specific vendor (as applicable), did not intend to imply that reportability requirements to the NRC under 10 CFR 21 would be passed to the vendor. Those reportability requirements fall under the responsibility of NMWI. NWMI-QA-PRO-035, *Identification and Control of Nonconforming Items*, identifies the process by which nonconforming conditions will be identified and controlled.

RAI C2.15-1b	*Clarify the circumstances in which 10 CFR Part 21 and 50.55(e) will be applicable to another entity.*

NWMI, as the license holder of the RFP, has responsibility for reporting defects and noncompliance under 10 CFR 21. NWMI-QA-PRO-035 identifies the process by which nonconforming conditions will be identified and controlled.

No.	Request for additional information
	Appendix 12C, Section 2.17 – Quality Records
RAI C2.17-1	*ANS 15.8 states that the [quality] records shall include as a minimum: inspection and test results, results of quality assurance reviews, quality assurance procedures, and engineering.* *NWMI PSAR, Chapter 12, Section C2.17.2, "Requirements," states that what is considered a quality record will be provided in implementing procedures. PSAR Section 12.6 "Records," states that the records management program will define process for managing records and will be consistent with the requirement of applicable regulations. It also states that the identification, generation and authentication, maintenance, and disposition of records will be provide in the Operating License Application. There were no additional details on how the records for the construction, installation and other documentation are going to be maintained. Furthermore, there is no clarification on the retention of period of the documents listed above.* *Clarify if the implementing procedures, at a minimum, address the above listed quality records and retention time for such records.*

NWMI-QA-PRO-017 (Rev 1), *Quality Records*, identifies the process by which quality records are identified and maintained. Items identified in Section 6.1 of the procedure as quality documents are relevant to the final design and construction phase. These include:

- Contracts and specifications (including any modifications)
- Drawings
- Procurement records
- Test procedures
- Test reports
- Engineering reports (including calculations, and software verification and validation reports)
- Inspection reports
- Assessment reports
- Supplier evaluation reports
- Training records
- Project-specific Quality Assurance Plan
- Corporate Environmental, Safety, and Health Program Plan
- Corporate Quality Assurance Program Plan
- Implementing procedures
- Material test reports
- Certifications of conformance
- Personnel qualification/certification records
- Design review reports
- Project-specific procedures
- Calibration records
- Nonconformance reports
- Corrective Action reports
- Stop work requests

All quality records will be retained for the life of the RPF.

No.	Request for additional information
RAI C2.17-2	*ANS 15.8 states that some records shall be maintained by or for the plant owner for the life of the particular item while it is installed in the plant or stored for future use. Such records shall be classified in accordance with the following criteria: (a) those which would be of value in demonstrating capability for safe operation; (b) those which would of value in maintaining, reworking, repairing, replacing, or modifying an item; (c) those which would be of value in determining the cause of results of an accident or malfunction of a safety-related item; (d) those which provide required baseline data for in-service inspections; or (e) those which would be of value in planning for facility decommissioning.* *NWMI PSAR, Chapter 12, Section C2.17.2.2, "Classifications," states in part, that what is considered lifetime records will be delineated within implementing procedures. However, it does not state if NWMI will be following ANS 15.8 guidance regarding lifetime records.* *Clarify whether the lifetime records, at a minimum, will be classified in accordance with ANS 15.8 criteria.*

Lifetime records will be classified consistent with the recommendations found in ANS 15.8, *Quality Assurance Program Requirements for Research Reactors*.

No.	Request for additional information
	Appendix 12C, Section 2.19 – Experimental Equipment
RAI C2.19-1	*ANS 15.8 states that the QA program shall provide controls over the design, fabrication, installation, and modification of experimental equipment to the extent that these impact safety-related items.* *The NWMI PSAR does not appear to state whether the QA program will provide controls over the design, fabrication, installation, and modification of experimental equipment to the extent that these impact safety-related items.* *Clarify whether the applicant intends to provide controls over the design, fabrication, installation, and modification of experimental equipment to the extent that these impact safety-related items or where this information is stated in the PSAR.*

NWMI does not intend to have any experimental equipment; therefore, there will be no potential impacts to safety-related items from such equipment. Item 19, Experimental Equipment, in Table C-1 (Section C1.3) is listed as "Not applicable; no experimental equipment."

CHAPTER 13.0 – ACCIDENT ANALYSIS

No.	Request for additional information
	Section 13.1 – Accident Analysis Methodology and Preliminary Hazards Analysis
RAI 13.1-2	*Subparagraph 50.35 (a)(2) of 10 CFR Part 50 requires that further technical or design information as may be required to complete the safety analysis and which can be left for later consideration, will be supplied in the final safety analysis report.* *NWMI PSAR, Section 13.1, page 13-28, "Uranium Recovery Open Item," identifies two potential accident sequences for the uranium recovery operation preliminary hazard analyst items 4.2.4.8 and 4.3.4.8 (tentative accident sequence S.R.14) that involves an interaction between a high temperature nitric acid solution and the uranium purification ion exchange media. The application states that adverse events need to be further researched, but is not clear on the specific nature of this research or when it will be completed.*
RAI 13.1-2a	*Describe what additional work has occurred or is planned to understand the likelihood and potential consequences of such an accident sequence.*

Several laboratory resin tests are being completed to determine the interactions between the solutions and resin as a function of temperature. The results from these tests will help define the hazard and accident controls if needed. The testing is planned for the first quarter of 2017.

RAI 13.1-2b	*Describe how the research and development results might influence the final design of the RPF and the uranium recovery process equipment.*

A uranium ion exchange column represents a closed system (for the purposes of safety evaluation) operating at approximately 45 pounds (lb)/square inch (in.^{2}) gauge to address pressure drop through the media bed at proposed operating flow rates. A rupture disk vent path has been included in current equipment descriptions as the feature that mitigates a potential accident resulting in exothermic reactions within the ion exchange column. The research and development results will either:

1. Confirm that a pressure relief system is feasible to design for an ion exchange column operating at approximately 45 lb/in^2 gauge and the uranium separation process approach will continue, or
2. Require a design change to the system or implementation of additional controls/process parameters to reduce the likelihood of a reaction or change of separation technology.

RAI 13.1-2c	*The staff needs to understand NWMI's assessment of the potential for exothermic reactions involving the chemical extractant in the ion exchange columns resin and other process equipment which may handle the resin including the evaporators that are downstream of the ion exchange system. The PSAR states the operating temperature range of the uranium concentrators in Table 4-50, "Uranium Recovery and Recycle Process Equipment" (UR-Z-320 and UR-Z-520) and the nitric acid concentration of the first concentrator (UR-Z-320) in Section 4.4.1.1.* *Explain whether the research and development includes an assessment of the potential for thermal and radiolytic decomposition of the chemical extractant which is the active component of the resin. Information in technical literature discusses exothermic decomposition at temperatures around 100°C; conclude that the chemical extractant has resistance to radiation and nitric acid than tributylphosphate.*

Release of diamylamylphosphonate (DAAP) from the ion exchange column media during operation must be evaluated as part of the research and development program. Limited data described in Appendix E of NWMI-2013-034, *Uranium Recovery and Recycle Process Descriptions, PFD and P&ID*, indicates that the media beads have the potential to swell when the adsorbed dose exceeded 10×10^6 rad. Swollen beads have the potential to release DAAP from the media skeleton to other process vessels. Release of DAAP is considered an issue from both a thermal/radiolytic decomposition perspective (e.g., in concentrators) and represents a potential criticality issue if DAAP were to collect as a separate phase in a non-geometrically favorable vessel.

No.	Request for additional information
	Section 13.2 – Analysis of Accidents with Radiological and Criticality Safety Consequences
RAI 13.2-2	*The ISG Augmenting NUREG-1537, Part 1, Section 13b.2, "Analyses of Accidents with Radiological Consequences," states that the radiation dose estimates should include estimates for the operating staff throughout the event and during recovery operations and also for the maximally exposed individual in the uncontrolled areas and at the nearest permanent residence.* *NWMI PSAR Section 13.2.1, "Maximum Hypothetical Accident," describes the accident scenario and presents the calculated TEDE results in PSAR Table 13-20, "Maximum Hypothetical Accident Total Effective Dose Equivalent," and PSAR Figure 13-2, "Total Effective Dose Equivalent (Inhalation plus External) for 2-Hour Ground Level Exposure from Maximum Hypothetical Accident," for a range of distances from the accident release site. PSAR Table 13-20 identifies the maximum dose as 22.6 rem at a distance of 1,100 m from the accident release site. No results or discussion is provided pertaining to operating staff dose or dose to the public at the nearest permanent residence. The location of the nearest permanent residence is not identified.* *Provide results and discussion pertaining to operating staff dose, including worker stay time. Identify the location of the nearest permanent residence and provide results and discussion pertaining to the dose to the public at the nearest permanent residence.*

The MHA discussion was inappropriately included in PSAR Section 13.2.1. NWMI interpreted the ISG augmenting NUREG-1537 as requiring an analysis consistent with both 10 CFR 70.61 and with the MHA approach. Additionally, NWMI combined the two in its approach to the MHA, producing an analysis inconsistent with the intended requirements of an MHA analysis. As only one analysis is required, the MHA discussion in PSAR Section 13.2.1 will be removed from the PSAR.

See response to RAI G-3 for further discussion.

RAI 13.2-3	*The ISG Augmenting NUREG-1537, Part 1, Section 13b.2, "Analyses of Accidents with Radiological Consequences," states that the radiological consequences should be in terms of TEDE.* *NWMI PSAR Figure 13-2 is titled "Total Effective Dose Equivalent (Inhalation plus External) for 2-Hour Ground Level Exposure from Maximum Hypothetical Accident." However, the title on the top of the graph states, "Total CEDE [committed effective dose equivalent] Dose Results for 25 ft Stack."* *Confirm if CEDE or TEDE is actually presented in PSAR Figure 13-2.*

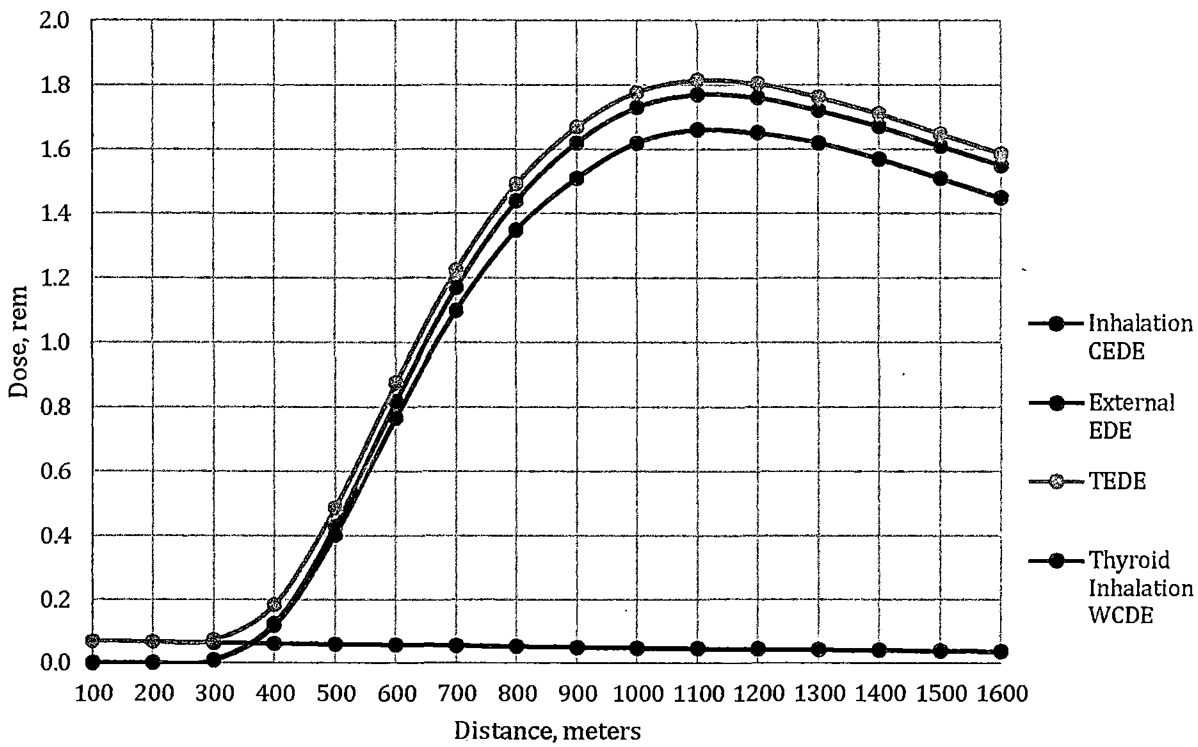
The data in Figure 13-2 of PSAR Section 13.2.1 is in units of TEDE. The labels on the figure will be corrected.

RAI 13.2-4	*The ISG Augmenting NUREG-1537, Part 1, Section 13b.2 states that the radiation dose estimates should include estimates for the operating staff throughout the event and during recovery operations and also for the maximally exposed individual in the uncontrolled areas and at the nearest permanent residence.* *NWMI PSAR Section 13.2.2, "Liquid Spills and Sprays with Radiological and Criticality Safety Consequences," describes the accident scenario and presents the calculated TEDE dose results in PSAR Table 13-24, "Spray Release Consequence Summary." PSAR Table 13-24 identifies the maximum unmitigated TEDE dose as 12 rem to the public. PSAR Section 13.2.2.7.2, "Confinement Release Consequence," contains a discussion of the mitigated dose consequence value of 0.97 rem to the public. No results or discussion is provided pertaining to operating staff dose or dose to the public at the nearest permanent residence. The location of the nearest permanent residence is not identified.*

No.	Request for additional information
RAI 13.2-4 (cont)	*Provide results and discussion pertaining to operating staff dose, including worker stay time. Identify the location of the nearest permanent residence and provide results and discussion pertaining to the dose to the public at the nearest permanent residence.*

In the Construction Permit Application, NWMI originally used both RASCAL and RSAC to model off-site accident consequences. Since the submission of the application, NWMI has selected RSAC for off-site accident consequence modeling. For the liquid spills and spray accident in PSAR Section 13.2.2, NWMI has rerun the off-site dose calculations using RSAC. The nearest permanent resident (432 m [0.27 mi]) unmitigated dose estimate is 300 mrem, while the maximum receptor location (1,100 m [0.68 mi]) has a TEDE of 1.8 rem (Figure 1).

Figure 1. Offsite Dose Calculation of Spray Leak Accident as a Function of Distance



The RPF operating staff should not receive an occupational exposure from a spray leak or spill in the hot cells due to the shielding walls and ventilation flow rate.

The third accident scenario in PSAR Section 13.2.2 is a spill of molybdenum-99 (^{99}Mo) product during container loading operations. This scenario will be reevaluated in the Operating Licensing Application. The current scenario assumes three to four times the curie content of a shipping cask and does not take in to account the inner container that would also reduce or eliminate the spill. Operating staff dose estimates and worker stay time (if needed) for accident scenarios will be provided in the FSAR as part of the Operating Licensing Application.

No.	Request for additional information
RAI 13.2-5	*The ISG Augmenting NUREG-1537, Part 1, Section 13b.2 states, "These analyses should include accident scenarios within the operating categories listed in Section 13.b.1.1 and, as a minimum, include accidents caused by those initiating events listed in Section 13b.a.2 within each operating category." The ISG Augmenting NUREG-1537, Part 1, Section 13.b.1.2, "Accident – Initiating Events," states "...should include the following initiating events: Loss of electrical power."* *NWMI PSAR, Section 13.2.5, "Loss of Power," describes the accident scenario. PSAR Section 13.2.5.7, "Evaluation of Potential Radiological Consequences," does not present any dose consequence results for the loss of power accident scenario and states, in part, that "A detailed evaluation of potential radiological consequences will be developed for the Operating License Application."* *The purpose of Chapter 13 of the construction application is to provide adequate accident analysis that results in identifying necessary safety SSCs and IROFs to proceed with design and construction. Since a Loss of Power event would be an anticipated (not unlikely) event, potentially affecting operational processes involving radioactive and fissile materials; a Loss of Power event analysis, results, and identification of derived safety SSCs and IROFs are important to preliminary facility and process design.* *Provide an analysis, results, and discussion pertaining to the potential radiological consequences from a postulated loss of power accident.*

Loss of power was identified as an initiating event in numerous RPF accident sequences. NWMI concluded that no additional radiological accidents were present beyond what was identified in the hazard analysis and the quantitative risk analysis. No additional IROFS were identified from loss of power. The summary of radiological consequences from the analysis of other accidents where loss of power was an initiator will be provided in the FSAR as part of the Operating License Application.

RAI 13.2-6	*The ISG Augmenting NUREG-1537, Part 2, Section 13b.1.2, "Accident Initiating Events, External Events," states, in part, that "Consequences of natural external events that cause facility damage (e.g., seismic events that damage the confinement or containment) are within the bounds discussed for other accidents in this chapter."* *NWMI PSAR, Section 13.2.6, "Natural Phenomena Events," states that:* *The RPF is designed to withstand the effects of natural phenomena events. Consequences of natural phenomena accident sequences have been evaluated. Sections 13.2.6.1 through 13.2.6.6 provide event descriptions and identify any additional controls required to protect the health and safety of workers, the public, and environment.* *Subsequent PSAR Sections 13.2.6.1 through 13.2.6.6 each provide a brief discussion of the qualitative dose consequence resulting from each specific natural phenomena event, but do not provide any comparison of dose consequences to the dose consequences from the bounding accidents previously analyzed and documented in PSAR Sections 13.2.2 through 13.2.5.* *Provide a comparison of PSAR Sections 13.2.6.1 through 13.2.6.6 of the dose consequence values for each natural phenomena event versus the associated bounding event dose consequence analyzed in PSAR Sections 13.2.2 through 13.2.5, supporting why the natural phenomena events dose consequences are bounded by the dose consequences of other analyzed accidents in Chapter 13.*

Dose consequences were not determined for the RPF natural phenomena events. Using ISA methodology and since the IROFS and RPF processing areas are designed to withstand DBEs (highly unlikely events), off-site dose calculation were not completed for the Construction Permit Application. The worker dose estimates for a seismic event during target cask unloading will be developed and provided in the FSAR as part of the Operating License Application.

No.	Request for additional information
RAI 13.2-7	*The ISG Augmenting NUREG-1537, Part 1, Section 13.b.2 states, in part, "Evaluate the potential radiological consequences using realistic methods."* *NWMI PSAR Section 13.2.2.7.2, "Confinement Release Consequence," discusses the dose consequence results for the liquid spills and sprays event category and documents the calculated dose consequence results in PSAR Table 13-24. The discussion text section and Table 13-24 demonstrate that the unmitigated dose consequence for the bounding event is 12 rem to the public, and the mitigated dose consequence is 0.97 rem to the public. In this same section, PSAR Table 13-23, "Release Consequence Evaluation RASCAL Code Inputs," documents the analysis inputs and assumptions used to calculate the dose consequences for this bounding event. PSAR Table 13-23 shows that a receptor distance of 100 m is selected in the analysis. The selection of a receptor distance of 100 m for the accident analysis consequences documented in PSAR Section 13.2.2.7.2 is not consistent with the identification of the maximum dose location at 1,100 m in PSAR Table 13-20 and Table 13-26.* *Provide additional discussion of why a receptor distance of 100 m would result in the limiting public dose for this event, when other event dose consequence results in PSAR Table 13-20 and Table 13-26, "Target Dissolver Offgas Accident Total Effective Dose Equivalent," demonstrate that the maximum dose to the public occurs at a distance of 1,100 m.*
RSAC has been selected as the model platform for all accident release and dose calculations for the RPF. The accidents described in PSAR Section 13.2.2.2 have been reevaluated using RSAC (instead of RASCAL). The maximum dose to the public occurs at a distance of 1,100 m (0.68 mi). The response to RAI 13.2-4 provides additional dose information.	
(Applies to RAI 13.2-8 through 13.3-1)	*Subparagraph 50.35(a)(1) of 10 CFR Part 50 requires that the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.*
RAI 13.2-8	*The ISG augmenting NUREG-1537, Part 2, Section 13b.2, states that the application should describe the potential accidents caused by process deviations or other events internal to the facility and credible external events.* *NWMI PSAR Section 4.3.5.1, "Process Description," identifies the ion exchange material in the secondary ion exchange column (MR-IS-225) as an ion exchange resin which will be eluted with nitric acid. Table 4-44 identifies the normal operating temperature but the process uses an active cooler (Chiller MR-Z-230) to maintain a specified temperature.* *There is experience of energetic reactions with anion exchange resins in nitric acid media. Most of the accident experience is for nitric acid concentration higher than that proposed for elution, but it is also noted that resin degradation rates vary with resin type, temperature, and acid concentration. There appears to be some potential for anion exchange reactions under upset conditions or off normal conditions at the NWMI facility. The practical guidelines information sheet for ion exchange resins identifies a temperature maximum on the order of 30 to 60°C and nitric acid as a material that can cause explosive reactions.* *The accident analysis for the molybdenum recovery system in Table 13-11, "Adverse Event Summary for Molybdenum Recovery and Identification of Accident Sequences Needing Further Evaluation," of the PSAR does not identify an accident involving overheating and reaction of the anion exchange material and there is no discussion of why this accident is not considered given the presence of decay heat and the use of nitric acid as an eluting agent.* *Information is needed on why this type of accident was not or does not need to be considered in the accident analysis for the molybdenum recovery system or the waste management systems that would handle the spent anion exchange resin. The description of waste management systems in Chapter 4 of the PSAR does not describe the handling of spent resin from the molybdenum recovery and purification operations.*

No.	Request for additional information
RAI 13.2-8a	*Describe NWMI's assessments of accidents involving the anion exchange media used in the molybdenum purification system.*

The anion exchange columns in the RPF molybdenum system are very small and are single use. There is a chiller to maintain process conditions. After elution and rinsing, the column and resin are discarded as solid waste. No additional hazards have been identified for the anion exchange columns/media.

No.	Request for additional information
RAI 13.2-8b	*Provide a discussion of any potential accidents involving this material that were identified and evaluated and any design features considered reasonable for managing this hazard have been adequately defined.*

The process hazard analysis (PHA) tables for the RPF molybdenum system and waste handling will be updated for hazards associated with the molybdenum resin as part of the ongoing ISA process and will be reflected in the operating licenses. Hazards/accidents will include changing temperature, flow and acid conditions, and their impacts on the anion resin.

No.	Request for additional information
RAI 13.2-9	*The ISG Augmenting NUREG-1537, Part 2, Section 13b, discusses facility accident analysis for the radioisotope production facility. The section states that NUREG-1520, Section 3.4, provides additional criteria for adherence to the safety program and integrated safety analysis (ISA) performance. Section 3.4.3.1.(5)(b) of NUREG-1520 states that process hazard analysis methods (i.e., ISA Methods) are acceptable if the method addresses all modes of operation, including startup, normal operation, shutdown, and maintenance.* *The NWMI PSAR has analyzed potential accidents that might occur during normal process operations as part of its process for identifying major features or components incorporated into the design for the protection of the health and safety of the public. These major features are included in PSAR Tables 1 8, 1-12, 6-1, 6-2 and Table 4-21 of the ISA Summary. These analyses are discussed in Chapter 13 and the ISA Summary of the PSAR and are the basis for design features intended to protect the health and safety of workers and the public for these potential accident conditions (e.g., to protect the plant personnel during normal operations (i.e., manipulator-accessed hot cells and the larger tank hot cell).* *The PSAR Chapter 13 accident analysis does not address other operating modes (e.g., startup, maintenance, extended shutdown) that can introduce different types of hazards. For example maintenance operations can involve reduced protective barriers between the hazardous materials and workers. Extended shutdown might create a situation where radiolytic hydrogen would be generated over extended periods and could accumulate in unusual locations. Extended shutdown could also result in organic ion exchange resins degradation making it more prone to react exothermically upon process startup.*
RAI 13.2-9a	*Describe how the accident analysis evaluated non-routine situations to determine if there are additional hazards that have to be accommodated by design features.*

The technical specification will define modes and limiting conditions for operation (and maintenance). As suggested in the RAI, maintenance activities (e.g., removing a cover block to replace a piece of failed equipment) could change the configuration of the facility. For these situations, limits on operations activities or acceptable inventories will be defined and implemented. Procedures will be developed to place equipment/plant conditions in a proper configuration for outages, including surveillance and monitoring activities. The radioactive inventory after a one-month shutdown is only 5 percent of the incoming inventory of eight MURR targets 8 hours after end of irradiation. After a long outage, a water run/test would typically be performed to check the equipment and processes.

No.	Request for additional information
RAI 13.2-9b	*Describe how NWMI will address an extended RPF shutdown to ensure the protection of health and safety of workers and the public.*

The technical specification will define modes and limiting conditions for operation (and maintenance). As suggested in the RAI, maintenance activities (e.g., removing a cover block to replace a piece of failed equipment) could change the configuration of the facility. For these situations, limits on operations activities or acceptable inventories will be defined and implemented. Procedures will be developed to place equipment/plant conditions in a proper configuration for outages, including surveillance and monitoring activities. The radioactive inventory after a one-month shutdown is only 5 percent of the incoming inventory of eight MURR targets 8 hours after end of irradiation. After a long outage, a water run/test would typically be performed to check the equipment and processes.

No.	Request for additional information
RAI 13.2-10	*The ISG Augmenting NUREG-1537, Part 2, Section 13b.2, states that the application should describe the potential accidents caused by process deviations or other events internal to the facility and credible external events.* *Section 13.3.2, "Nitric Acid Fume Release," of the NWMI PSAR discusses nitric acid fume release. The NRC staff performed an independent verification of the Areal Locations of Hazardous Atmospheres calculations and confirms that the scenario modeled would exceed the thresholds of high consequence for a worker and for the public. The NRC staff needs to understand any design features that NWMI plans to rely on to protect workers and the public from this high consequence event.* *Identify and describe any design features NWMI plans to rely on to prevent or mitigate a nitric acid fume release.*

General RPF design features intended to prevent/mitigate a nitric acid fume release include RPF building containment and nitric acid storage tank construction and venting. Specific features will be addressed in the FSAR as part of the Operating License Application.

No.	Request for additional information
	Section 13.3 – Analysis of Accidents with Hazardous Chemicals
RAI 13.3-1	*The ISG Augmenting NUREG-1537, Part 2, Section 13b.2, states that the application should provide that the applicant's facility design, operations, and safety controls for chemical safety provide reasonable assurance that they will function as intended and ensure the safe handling of licensed material at the facility.* *NWMI PSAR, Chapter 4, Section 4.3.4.6, "Chemical Hazards," describes the chemical protection provisions for each process. For processes having the potential of a chemical accident, the PSAR states that those features preventing release of radioactive material and limiting radiation exposure will also protect workers and the public from exposures to hazardous chemicals. The staff needs to understand any design features that NWMI plans to rely on to protect workers and the public from chemical hazards.*
RAI 13.3-1a	*Identify and describe the specific design features that NWMI plans to rely on to protect workers and public from exposures to hazardous chemicals.*

Specific RPF design features relied on to protect workers and the public from hazardous chemical exposure are the RPF building structure (walls, roof) that provide physical barriers to chemical releases, and the building ventilation system that uses zone-based negative differential pressures with respect to the RPF surroundings to prevent release of hazardous chemical vapors. Note that these design features are the same features that protect against radioactive material releases.

No.	Request for additional information
RAI 13.3-1b	*Identify and describe the accident scenarios that will discuss these safety features.*

Detailed RPF accident scenarios for chemical hazards will be developed, analyzed, and documented in the FSAR as part of the Operating License Application. Note that the building structure and general containment features, including the ventilation system, serve as barriers to protect workers and the public against both chemical and radioactive material hazards.

No.	Request for additional information
RAI 13.3-2	*Per 10 CFR 50.35(a)(2), a construction permit will be issued if the Commission finds that further technical or design information as may be required to complete the safety analysis and which can be left for later consideration, will be supplied in the final safety analysis report.* *The ISG Augmenting NUREG-1537, Part 2, Section 13b.2, states that the application should provide that the applicant's facility design, operations, and safety controls for chemical safety provide reasonable assurance that they will function as intended and ensure the safe handling of licensed material at the facility.* *NWMI PSAR Section 9.7.4.3, "Operational Analysis and Safety Function," discusses operational analysis and safety function and identifies two IROFS that will be implemented CS-18, "Backflow Preventive Device and CS-19, "Safe Geometry Day Tanks" to ensure criticality and chemical safety. The chemical safety roles of these IROFS is not described in PSAR, Section 13.3 or the ISA Summary. The NRC staff needs to understand the chemical safety role NWMI plans for these two IROFS.* *Provide additional information that identifies the specific chemical safety accidents these IROFS will be preventing or mitigating.*

Specific chemical safety accidents will be developed, analyzed, and documented in the FSAR as part of the Operating License Application, along with identification of relevant technical specifications. IROFS CS-18, Backflow Prevention Device, and IROFS CS-19, Safe Geometry Day Tanks, protect against general classes of chemical accidents in which hazardous chemicals (and radioactive materials) could enter systems not designed for them, or could be released via tank overflow events or other chemical losses/spills. IROFS C-19 further protects against accidental nuclear criticality events that could initiate or exacerbate hazardous chemical releases.

CHAPTER 14.0 – TECHNICAL SPECIFICATIONS

No.	Request for additional information
	Chapter 14.0 – Technical Specifications
RAI 14.0-1	*Paragraph 50.34(a), "Preliminary safety analysis report," of 10 CFR Part 50 states that each application for a construction permit shall include a preliminary safety analysis report. Section 50.34(a)(5) states that the minimum information to be included shall consist of the identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design.* *The ISG Augmenting NUREG-1537, Part 1, Chapter 14, "Technical Specifications," states, in part, that:* *NUREG-1537, Part 1, Chapter 14 of the format and content guide, as augmented by this ISG, is applicable to providing a description of the technical specifications for the licensing of a radioisotope production facility and a non-power reactor facility. Whenever the term "reactor" appears, it is understood to mean a "non-power reactor facility," a "radioisotope production facility," or both, as applicable.* *NWMI PSAR Section 3.0, "Design of Structures, Systems, and Components," states that the design information for the complete range of normal operating conditions for various facility systems is provided throughout the construction permit application, and includes potential conditions or other items that will be probable subjects of technical specifications associated with the RPF structures and design features are discussed in Chapter 14.0.* *PSAR Chapter 14.0, states, in part, that:* *This chapter describes the process by which the Northwest Medical Isotopes, LLC (NWMI) Radioisotope Production Facility (RPF) technical specifications will be developed and written. For the Construction Permit Application, NWMI has prepared the strategy and content of what will be required for technical specifications during RPF operations.* *However, PSAR Chapters 3.0 and 14.0 do not provide the stated design information.* *In accordance with 10 CFR 50.34(a)(5), identify and justify the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design.*

The variables or conditions in Table 11 are probable subjects of technical specifications based on their involvement with preventing release of radioactive materials routinely or in the event of an accident. Table 11 will be added to the PSAR Chapter 14.0. Technical specifications on these items are planned for inclusion in sections that address limiting conditions of operation and surveillance/maintenance in the FSAR as part of the Operating License Application.

Table 11. Potential Technical Specifications

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

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

RAM = radioactive material.
SNM = special nuclear material.

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Attachment A

Section 3.5, "Systems and Components" of NWMI-2013-021, *Construction Permit Application for Radioisotope Production Facility*

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3.5 SYSTEMS AND COMPONENTS

Certain systems and components of the RPF are considered important to safety because they perform safety functions during normal operations or are required to prevent or mitigate the consequences of abnormal operational transients or accidents. This section summarizes the design basis for design, construction, and operating characteristics of safety-related SSCs of the RPF.

3.5.1 General Design Basis Information

3.5.1.1 Classification of Systems and Components Important to Safety

The RPF systems and components will be classified according to their importance to safety, quality levels, and seismic class. The guidance used in developing these classifications during preliminary design with the support of regulatory guidance reviews, hazards and operability analysis, accident analysis, integrated safety analysis, and national consensus code requirements is presented below.

The RPF systems identified in Table 3-1 and their associated subsystems and components are discussed in the subsections that follow.

3.5.1.2 Classification Definitions

The definitions used in the classification of SSCs include the following.

In accordance with 10 CFR 50.2, "Definitions," design basis refers to information that identifies the specific functions to be performed by an SSC of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be:

- Restraints derived from generally accepted state-of-the-art practices for achieving functional goals
- Requirements derived from analysis (e.g., calculation, experiments) of the effects of a postulated accident for which a SSC must meet its functional goals

These reference bounds are to include the bounding conditions under which SSCs must perform design basis functions and may be derived from normal operation or any accident or events for which SSCs are required to function, including anticipated operational occurrences, design basis accidents, external events, natural phenomena, and other events specifically addressed in the regulations.

Safety-related is a classification applied to items relied on to remain functional during or following a design basis event (DBE) to ensure the:

- Integrity of the facility infrastructure
- Capability to shut down the facility and maintain it in a safe shutdown condition
- Capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the applicable guideline exposures set forth in 10 CFR 70.61, "Performance Requirements," as applicable

Design basis accident is a postulated accident that a nuclear facility must be designed and built to withstand, without loss to the SSCs necessary to ensure public health and safety.

Design basis event (DBE) is an event that is a condition of normal operation (including anticipated operational occurrences), a design basis accident, an external event, or natural phenomena for which the facility must be designed so that the safety-related functions are achievable.

Design basis accidents and transients are those DBEs that are accidents and transients and are postulated in the safety analyses. The design basis accidents and transients are used in the design of the facility to establish acceptable performance requirements for SSCs.

Single failure is considered a random failure and can include an initiating event (e.g., component failure, natural phenomenon, external man-made hazard) or consequential failures. Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence. This type of design verifies that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions. Mechanical, instrumentation, and electrical systems and components are designed to ensure that a single failure, in conjunction with an initiating event, does not result in the loss of the RPF's ability to perform its intended safety function. Design techniques such as physical separation, functional diversity, diversity in component design, and principles of operation, will be used to the extent necessary to protect against a single failure.

An initiating event is a single occurrence, including its consequential effects, that places the RPF (or some portion) in an abnormal condition. An initiating event and its resulting consequences are not considered a single failure.

Active components are devices characterized by an expected significant change of state or discernible mechanical motion in response to an imposed demand on the system or operation requirements (e.g., switches, circuit breakers, relays, valves, pressure switches, motors, dampers, pumps, and analog meters). An active component failure is a failure of the component to complete its intended safety function(s) on demand.

Passive components are devices characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design basis load demand on the system.

Defense-in-depth is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous material through the creation of multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied on. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

The RPF structure and system designs are based on defense-in-depth practices. The RPF design incorporates:

- Preference for engineered controls over administrative controls
- Independence to avoid common mode failures
- Other features that enhance safety by reducing challenges to safety-related components and systems

Safety-related systems and components identified in this section are described in Chapters 4.0; 5.0, "Coolant Systems;" 6.0; 7.0; 8.0, "Electrical Power Systems;" and 9.0, "Auxiliary Systems," as appropriate.

3.5.1.3 Nuclear Safety Classifications for Structures, Systems, and Components

SSCs in the RPF are classified as safety-related and non-safety-related. The safety-related SSCs include IROFS to meet the performance requirement of 10 CFR 70.61 and other safety related SSCs to meet the requirements of 10 CFR 20. The purpose of this section is to classify SSCs according to the safety function being performed.

In addition, design requirements will be placed on SSCs to ensure the proper performance of their safety function, when required.

- **Safety-related IROFS** – SSCs identified through accident analyses as required to meet the performance requirements of 10 CFR 70.61 (see Table 3-2)
- **Safety-related** – SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of workers, the public, and environment, includes SSCs to meet 10 CFR 20 normal release or exposure limits
- **Non-safety-related** – SSCs related to the production and delivery of products or services that are not in the above safety classifications

3.5.1.3.1 Quality Group Classifications for Structures, Systems, and Components

The assignment of safety-related classification and use of codes and standards conforms to the requirements NWMI's Quality Assurance Program Plan (QAPP) for the development of a Quality Group classification and the use of codes and standards. The classification system provides a recognizable means of identifying the extent to which SSCs are related to safety-related and seismic requirements, including ANS nuclear safety classifications, NRC quality groups, ASME Code Section III classifications, seismic categories, and other applicable industry standards, as shown in Table 3-7.

Quality assurance (QA) requirements are defined in the NWMI QAPP (Chapter 12.0, "Conduct of Operations," Appendix C). The definitions of QA Levels 1, 2, and 3 are provided below.

QA Level 1 will implement the full measure of the QAPP and will be applied to IROFS. IROFS are QA Level 1 items in which failure or malfunction could directly result in a condition that adversely affects workers, the public, and/or environment, as described in 10 CFR 70.61. Examples include:

- Items to prevent nuclear criticality accidents (e.g., preventive controls and measures to ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical)
- Items credited to withstand credible design-bases external events (e.g., seismic, wind)
- Items to prevent degradation of structural integrity (e.g., failure or malfunction of facility)

QA Level 2 will be applied to non-QA Level 1 safety SSCs. The QA program is important to the acceptability and suitability of the item or service to perform as specified. Acceptance methods shall be specified (including acceptance and other applicable performance criteria), documented, and verified before use of the item or service. Some of the required characteristics may be examined less rigorously than for QA Level 1. Examples of QA Level 2 items include:

- SSCs to meet 10 CFR 20 normal release or exposure limits
- Fire protection systems
- Seismic detection systems
- Safeguards and security systems
- Material control and accountability systems

QA Level 3 will include non-safety-related quality activities performed by NWMI that are deemed necessary to ensure the manufacture and delivery of highly reliable products and services to meet or exceed customer expectations and requirements. QA Level 3 items include those items that are not classified as QA Level 1 or QA Level 2. QA Level 3 items are controlled in accordance with standard commercial practices.

These quality activities are embodied in NWMI's QAPP and will be further specified in the Operating License Application, and when necessary.

3.5.1.3.2 Seismic Classification for Structures, Systems, and Components

SSCs identified as IROFs will be designed to satisfy the general seismic criteria to withstand the effects of natural phenomena (e.g., earthquakes, tornados, hurricanes, floods) without loss of capability to perform their safety functions. ASCE 7, Chapter 11, sets forth the criteria to which the plant design bases demonstrate the capability to function during and after vibratory ground-motion associated with the safe-shutdown earthquake conditions.

The seismic classification methodology used for the RPF complies with the preceding criteria, and with the recommendations stated in Regulatory Guide 1.29, *Seismic Design Classification*. The methodology classifies SSCs into three categories: seismic Category I (C-I), seismic Category II (C-II), and non-seismic (NS).

Seismic C-I applies to both functionality and integrity, while C-II applies only to integrity. SSCs located in the proximity of IROFs, the failure of which during a safe-shutdown earthquake could result in loss of function of IROFs, are designated as C-II. Specifically:

- C-I applies to IROFs. C-I also applies to those SSCs required to support to shut down the RPF and maintain it in a safe shutdown condition.
- C-II applies to SSCs designed to prevent collapse under the safe-shutdown earthquake. SSCs are classified as C-II to preclude structural failure during a safe-shutdown earthquake, or where interaction with C-I items could degrade the functioning of a safety-related SSC to an unacceptable level or could result in an incapacitating injury to occupants of the main control room.
- NS SSCs are those that are not classified seismic C-I or C-II.

3.5.2 Radioisotope Production Facility

Systems and components within the RPF are presented in Section 3.5.1. The RPF design basis evaluated the general design criteria from 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities." This evaluation is presented in Table 3-23. These general design criteria provide a rational basis from which to initiate design but are not mandatory. There are some cases where conformance to a particular criterion is not directly measurable. For each of the criteria, a specific assessment of the RPF design is made, and a complete list of references is included to identify where detailed design information pertinent to each criterion is treated. The Chapter 13.0 accident sequences for credible events define the DBE. The safety-related parameter limits ensure that the associated design basis is met for the events presented in Chapter 13.0.

Table 3-23. Design Criteria Requirements (4 pages)

Design criteria and description	Application and compliance
10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities"[a]	
Quality standards and records • Develop and implement design in accordance with management measures to ensure that IROFS are available and reliable to perform their function when needed. • Maintain appropriate records of these items by or under the control of the licensee throughout the life of the facility.	• SSCs important to safety will be designed, fabricated, erected, tested, operated, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they will be identified and evaluated to determine their applicability, adequacy, and sufficiency and will be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. • NWMI's QAPP will be established and implemented to provide adequate assurance that SSCs satisfactorily perform their safety functions. • Appropriate records of design, fabrication, erection, and testing of SSCs important to safety will be maintained by or under control of NWMI for the life of RPF. • NWMI will use a graduated QAPP that links quality classification and associated documentation to safety classification and to the manufacturing and delivery of highly reliable products and equipment. • The NWMI QAPP will provide details of the procedures to be applied, including quality and safety level classifications.
Natural phenomena hazards Provide for adequate protection against natural phenomena, with consideration of the most severe documented historical events for the site.	• SSCs important to safety will be designed, fabricated, erected, tested, operated, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they will be identified and evaluated to determine their applicability, adequacy, and sufficiency and will be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. • The design basis for these SSCs will include: – Appropriate consideration of the most severe natural phenomena that have been historically reported for the RPF site and surrounding area, including sufficient margin for limited accuracy, quantity, and period of time for which historical data has been accumulated – Appropriate combinations of natural phenomena effects during normal and accident operating conditions – Importance of the safety functions to be performed • Specific RPF design criteria and NRC general design criteria are discussed in Sections 3.1 and 3.5, respectively.

Table 3-23. Design Criteria Requirements (4 pages)

Design criteria and description	Application and compliance
Fire protection Provide for adequate protection against fires and explosions	• SSCs important to safety will be designed and located throughout the RPF to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. • Noncombustible and heat resistant materials will be used wherever practical throughout the RPF, particularly in locations such as confinement and the control room. • Fire detection and suppression systems of appropriate capacity and capability will be provided and designed to minimize the adverse effects of fires on SSCs important to safety. • Firefighting systems will be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs. • Where necessary, within zoned areas or where criticality and access are an issue, required systems will be manually initiated by operations after review of a detection signal. • RPF fire protection system will be designed such that a failure of any component will not impair the ability of safety-related SSCs to safely shut down and isolate the RPF or limit the release of radioactivity to provide reasonable assurance that the public will be protected from radiological risks resulting from RPF operations. • RPF fire protection system will be designed to provide reasonable assurance that the public will be protected from radiological risks resulting from RPF operations (e.g., failure of any component will not impair the ability of safety-related SSCs to safely shutdown and isolate the RPF or limit the release of radioactivity). • Chapters 6.0 and 9.0 provide additional information.
Environmental and dynamic effects Provide for adequate protection from environmental conditions and dynamic effects associated with normal operations, maintenance, testing, and postulated accidents that could lead to loss of safety functions	• SSCs important to safety are designed to accommodate effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Due to low temperature and pressure RPF processes, dynamic effects due to pipe rupture and discharging fluids are not applicable to the RPF.
Chemical protection Provide for adequate protection against chemical risks produced from licensed material, facility conditions that affect the safety of licensed material, and hazardous chemicals produced from licensed material	• Chemical protection in the RPF will be provided by confinement isolation systems, liquid retention features, and use of appropriate personal protective equipment. • Chapter 6.0, Section 6.2.1, provides additional information.
Emergency capability Provide for emergency capability to maintain control of: • Licensed material and hazardous chemicals produced from licensed material • Evacuation of on-site personnel • On-site emergency facilities and services that facilitate the use of available off-site services	• Emergency procedures will be developed and maintained for the RPF to control SNM and hazardous chemicals produced from the SNM. • A preliminary Emergency Preparedness Plan is provided in Chapter 12.0, Appendix B.

Table 3-23. Design Criteria Requirements (4 pages)

Design criteria and description	Application and compliance
Utility services Provide for continued operation of essential utility services	• The RPF is designed for passive, safe shutdown and to prevent uncontrolled release of radioactive material if normal electric power is interrupted or lost. • A standby diesel generator will be provided for asset protection of selected RPF systems. • Uninterruptable power supplies will automatically provide power to systems that support the safety functions protecting workers and the public. • A combination of uninterruptable power supplies and a standby electrical power system will provide emergency electrical power to the RPF. A 1,000 kW (1,341 hp) diesel generator will provide facility electric power. • Chapter 8.0, Section 8.2 provides additional information.
Inspection, testing, and maintenance Provide for adequate inspection, testing, and maintenance of IROFS to ensure availability and reliability to perform their function when needed	• The RPF is designed to provide access and controls for testing, maintenance, and inspection of safety-related SSCs, as needed, throughout the RPF. • Chapters 4.0, 6.0, 7.0, and 9.0 provide additional information.
Criticality control Provide for criticality control, including adherence to the double-contingency principle	• The RPF design will provide adequate protection against criticality hazards related to the storage, handling, and processing of SNM, which will be accomplished by: – Including equipment, facilities, and procedures to protect worker and public health and to 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 • Compliance with the requirements of criticality control, including adherence to the double-contingency principle, are described in detail in Chapter 6.0, Section 6.3.
Instrumentation and control The design must provide for inclusion of I&C systems to monitor and control the behavior of items relied on for safety.	• RPF SNM processes will be enclosed predominately by hot cells and glovebox designs except for the target fabrication area. • The FPC system will provide monitoring and control of safety-related components and process systems within the RPF. • The BMS (a subset of the FPC system) will monitor the RPF ventilation system and mechanical utility systems. • 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. • RPF designs are based on defense-in-depth practices and incorporate a preference for engineered controls over administrative controls, independence to avoid common mode failures, and incorporate other features that enhance safety by reducing challenges to safety-related components and systems. • The FPC system will provide the capability to monitor and control the behavior of safety-related SSCs. These systems ensure adequate safety of process and utility service operations in connection with their safety function. Controls are provided to maintain these variables and systems within the prescribed operating ranges under all normal conditions. • The FPC system is designed to fail to a safe-state or to assume a state demonstrated to be acceptable if conditions such as loss of signal, loss of energy or motive power, or adverse environments are experienced. • Chapter 7.0 provides additional I&C system information. Safety-related SSCs are described in Section 3.5 and Chapters 4.0, 5.0, 6.0, 7.0, and 8.0.

Table 3-23. Design Criteria Requirements (4 pages)

Design criteria and description	Application and compliance
Defense-in-depth[b]: Base facility and system design and facility layout on defense-in-depth practices. The design must incorporate, to the extent practicable: • Preference for the selection of engineered controls over administrative controls to increase overall system reliability • Features that enhance safety by reducing challenges to IROFS	• Defense-in-depth is a design philosophy that NWMI has applied from the beginning of the project and will continue through completion of a design that is based on providing successive levels of protection such that health and safety are not wholly dependent on any single element of the design, construction, maintenance, or operation of the RPF. • NWMI's risk insights obtained through performance of the accident analysis will be used to supplement the final design by focusing attention on the prevention and mitigation of the higher risk potential accidents. • Chapter 6.0 and 13.0 provide additional information.

[a] 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," *Code of Federal Regulations*, Office of the Federal Register, as amended.

[b] As used in 10 CFR 70.64, requirements for new facilities or new processes at existing facilities, defense-in-depth practices means a design philosophy, applied from the outset and through completion of the design, that is based on providing successive levels of protection such that health and safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of the facility. The net effect of incorporating defense-in-depth practices is a conservatively designed facility and system that will exhibit greater tolerance to failures and external challenges.

BMS	=	building management system.	NRC	=	U.S. Nuclear Regulatory Commission.
CFR	=	Code of Federal Regulations.	NWMI	=	Northwest Medical Isotopes, LLC.
ESF	=	engineered safety feature.	QAPP	=	quality assurance program plan.
FPC	=	facility process control.	RPF	=	Radioisotope Production Facility.
I&C	=	instrumentation and control.	SNM	=	special nuclear material.
IROFS	=	items relied on for safety.	SSC	=	structures, systems, and components.

The criteria are generic in nature and subject to a variety of interpretations; however, they also establish a proven basis from which to provide for and assess the safety of the RPF and develop principal design criteria. The general design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety (i.e., SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of workers, the public, and environment).

Safety-related SSCs for the RPF will be designed, fabricated, erected, and tested as required by the NWMI QAPP, described in Chapter 12.0, Appendix C. In addition, appropriate records of the design, fabrication, erection, procurement, testing, and operations of SSCs will be maintained throughout the life of the plant.

The RPF design addresses the following:

- Radiological and chemical protection
- Natural phenomena hazards
- Fire protection
- Environmental and dynamic effects
- Emergency capability (e.g., licensed material, hazardous chemicals, evacuation of on-site personnel, on-site emergency facilities/off-site emergency facilities)
- Utility services
- Inspection, testing, and maintenance
- Criticality safety
- Instrumentation and controls
- Defense-in-depth

Safety-related systems and components will be qualified using the applicable guidance in the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE 323, *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*. The qualification of each safety-related system or component needs to demonstrate the ability perform the associated safety function:

- Under environmental and dynamic service conditions in which they are required to function
- For the length of time the function is required

Additionally, non-safety-related components and systems will be qualified to withstand environmental stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the safety-related functions.

The RPF instrumentation and control (I&C) system (also known as the facility process control [FPC] system) will provide monitoring and control of the process systems within the RPF that are significant to safety over anticipated ranges for normal operations and abnormal operations. 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.

The FPC system will also ensure that process and utility systems operate in accordance with their safety function. Controls will be provided to maintain variables and systems within the prescribed operating ranges under all normal conditions. In addition, the FPC system is designed to fail into a safe state or to assume a state demonstrated to be acceptable if conditions such as loss of signal, loss of energy or motive power, or adverse environments are experienced.

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.

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 fire protection system will have its own central alarm panel. The fire protection system will report the status of the fire protection equipment to the central alarm station and the RPF control room.

This integrated control system will be isolated from safety-related components consistent with IEEE 279, *Criteria for Protection Systems for Nuclear Power Generating Stations*. In addition, the RPF is designed to meet IEEE 603, *Standard Criteria for Safety Systems for Nuclear Power Generating Stations*, for separation and isolation of safety-related systems and components. Chapter 7.0 provides additional details on the integrated control system.

3.5.2.1 System Classification

The RPF is classified as a non-reactor nuclear production facility per 10 CFR 50. In addition, a portion of the RPF will fabricate LEU targets, similar to fuel fabrication per 10 CFR 70. Due to the nature of the work performed within facility, a hazardous occupancy applies. Table 3-24 provides the RPF classification for hazards occupancy, construction, risk, and seismic design categories.

Table 3-24. System Classifications

Classification description	Classification	Source
Hazard category	Intermediate hazard	NRC
Occupancy type	Mixed, A-2, B, F-1, H-3 and H-4	IBC 2012[a]
Construction type	II-B	IBC 2012[a]
Risk category	IV	ASCE 7[b]
Seismic design category	C	ASCE 7[b]

[a] IBC 2012, "International Building Code," as amended, International Code Council, Inc., Washington, D.C., February 2012.
[b] ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, American Society of Civil Engineers, Reston, Virginia, 2013.
NRC = U.S. Nuclear Regulatory Commission.

3.5.2.2 Classification of Systems and Components Important to Safety

RPF SSCs, including their foundations and supports, designed to remain functional in the event of a DBE are designated as C-I. SSCs designated IROFS are also classified as C-I. SSCs co-located with C-I systems are reviewed and supported in accordance with II over I criteria. This avoids any unacceptable interactions between SSCs.

C-I structures should be designed using dynamic analysis procedures, or when justified, equivalent static procedures using both horizontal and vertical input ground motions. For dynamic analyses, either response spectra or time history analyses approaches may be used. Dynamic analysis should be performed in accordance with the procedures of ASCE 4, with the exception of the damping limitations presented in Section 3.4.1.

Table 3-25 lists the RPF SSCs and associated safety and seismic classifications and quality level group for the top-level systems. Subsystems within these systems may be identified with lower safety classifications. For example, the standby power supply (UPS) is an IROFS, while the standby diesel generator is classified as safety-related.

Table 3-25. System Safety and Seismic Classification and Associated Quality Level Group (2 pages)

System name (code)	Highest safety classification[a]	Seismic classification[b]	Quality level group
Facility structure (RPF)	IROFS	C-I	QL-1
Target fabrication (TF)	IROFS	C-I	QL-1
Target receipt and disassembly (TD)	IROFS	C-I	QL-1
Target dissolution (DS)	IROFS	C-I	QL-1
Mo recovery and purification (MR)	IROFS	C-I	QL-1
Uranium recovery and recycle (UR)	IROFS	C-I	QL-1
Waste handling (WH)	IROFS	C-I	QL-1
Criticality accident alarm (CA)	IROFS	C-I	QL-1
Radiation monitoring (RM)	IROFS	C-I	QL-1
Standby electrical power (SEP)	IROFS	C-I	QL-1
Normal electrical power (NEP)	SR	C-I	QL-1

Table 3-25. System Safety and Seismic Classification and Associated Quality Level Group (2 pages)

System name (code)	Highest safety classification[a]	Seismic classification[b]	Quality level group
Process vessel ventilation (PVV)	IROFS	C-I	QL-1
Facility ventilation (FV)	IROFS	C-I	QL-1
Fire protection (FP)	SR	C-II	QL-2
Plant and instrument air (PA)	NSR	C-II	QL-2
Emergency Purge gas (PG)	IROFS	C-I	QL-1
Gas supply (GS)	NSR	C-II	QL-2
Process chilled water (PCW)	IROFS	C-I	QL-1
Facility chilled water (FCW)	NSR	C-II	QL-2
Facility heated water (HW)	NSR	C-II	QL-2
Process steam (boiler)	IROFS	C-II	QL-2
Demineralized water (DW)	NSR	C-II	QL-2
Chemical supply (CS)	NSR	C-II	QL-2
Biological shield (BS)	IROFS	C-I	QL-1
Facility process control (FPC)	SR	C-II	QL-2

[a] Safety classification accounts for highest classification in the system. Systems that are classified as safety-related may include both safety-related and non-safety-related components. Only safety-related components will be used to satisfy the safety functions of the system, whereas non-safety-related components can be used to perform non-safety functions. For example, there are non-safety-related components, such as fans, within the safety-related ventilation systems that perform non-safety-related functions.

[b] Seismic category may be locally revised to account for II over I design criteria and to eliminate potential system degradation due to seismic interactions.

[c] Ventilation zone classifications vary –Ventilation Zone I and II are considered safety-related, C-I and QL-1; Ventilation Zone III and IV are considered non-safety-related, C-II and QL-2.

IROFS = items relied on for safety.
NSR = non-safety related.
RPF = Radioisotope Production Facility.
SR = safety-related (not IROFS).

SSCs that must maintain structural integrity post-DBE, but are not required to remain functional are C-II. All other SSCs that have no specific NRC-regulated requirements are designed to local jurisdictional requirements for structural integrity and are C-III. All C-I SSCs are analyzed under the loading conditions of the DBE and consider margins of safety appropriate for that earthquake. The margin of safety provided for safety-class SSCs for the DBE are sufficient to ensure that their design functions are not put at risk. Table 3-26 presents the likelihood index limit guidelines and associated event frequency and risk index limits.

Table 3-26. Likelihood Index Limit Guidelines

	Likelihood category	Event frequency limits	Risk index limits
Likely normal facility process condition	4	Multiple events per year	> or = 0
Not unlikely (frequent facility process condition)	3	More than 10^{-4} per event, per year	>-4 <0
Unlikely (infrequent facility process condition)	2	Between 10^{-4} and 10^{-5} per event, per year	-4 to 5
Highly unlikely (limiting facility process condition)	1	Less than 10^{-5} per event, per year	< -5

3.5.2.3 Design Basis Functions, Values, and Criteria

The design basis for systems and components required for safe operation and shutdown of the RPF are established in three categories, which are described below. The preliminary design basis functions and values for each major system are provided in the following subsections.

Design Basis Functions

- License conditions, orders, or technical specifications
- Functions credited in the safety analysis to ensure safe shutdown of the facility is achieved and maintained, prevent potential accidents, or mitigate the potential consequences of accidents that could result in consequences greater than applicable NRC exposure guidelines

Design Basis Values

- Values or ranges of values of controlling parameters established as reference bounds for RPF design to meet design basis function requirements
- Values may be established by an NRC requirement, derived from or confirmed by the safety analysis, or selected by the designer from an applicable code, standard, or guidance document

Design Basis Criteria

- Code-driven requirements established for the RPF fall into seven categories, including fabrication, construction, operations, testing, inspection, performance, and quality
- Codes include national consensus codes, national standards, and national guidance documents
- Design of safety-related systems (including protection systems) is consistent with IEEE 379, *Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*, and Regulatory Guide 1.53, *Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems*
- Protection system is designed to provide two or three channels for each protective systems and functions and two logic train circuits:
  - Redundant channels and trains will be electrically isolated and physically separated in areas outside of the RPF control room
  - Redundant design will not prevent protective action at the system level

3.5.2.4 System Functions/Safety Functions

The NWMI RPF will provide protection against natural phenomena hazards for the personnel, SNM, and systems within the facility. The facility will also provide protection against operational and accident hazards to personnel and the public. Table 3-2 lists the IROFS defined by the preliminary hazards analysis.

3.5.2.5 Systems and Components

3.5.2.5.1 Mechanical

RPF C-I mechanical equipment and components (identified in Table 3-25) will be qualified for operation under the DBEQ seismic conditions by prototype testing, operating experience, or appropriate analysis.

The C-I mechanical equipment is also designed to withstand loadings due to the DBEQ, vibrational loadings transmitted through piping, and operational vibratory loading, such as floor vibration due to other operating equipment, without loss of function or fluid boundary. This analysis considers the natural frequency of the operating equipment, the floor response spectra at the equipment location, and loadings transmitted to the equipment and the equipment anchorage.

The qualification documents and all supporting analysis and test reports will be maintained as part of the permanent plant record in accordance with the requirements of the NWMI QAPP.

The safety-related equipment and components within the RPF will be required to function during normal operations and during and following DBEs. This equipment will be capable of functioning in the RPF environmental conditions associated with normal operations and design basis accidents. Certain systems and components used in the ESF systems will be located in a controlled environment. This controlled environment is considered an integral part of the ESF systems.

3.5.2.5.2 Instrumentation and Electrical

C-I instrumentation and electrical equipment (identified in Table 3-25) is designed to resist and withstand the effects of the postulated DBEQ without functional impairment. The equipment will remain operable during and after a DBEQ. The magnitude and frequency of the DBEQ loadings that each component experiences will be determined by its location within the RPF. In-structure response curves at various building elevations have been developed to support design. The equipment (e.g., batteries and instrument racks, control consoles) has test data, operating experience, and/or calculations to substantiate the ability of the components and systems to not suffer loss of function during or after seismic loadings due to the DBEQ.

This certification of compliance with the specified seismic requirements, including compliance with the requirements of IEEE 344, is maintained as part of the permanent plant record in accordance with the NWMI QAPP.

3.5.2.6 Qualification Methods

Environmental qualification of safety-related mechanical, instrumentation, and electrical systems and components is demonstrated by tests, analysis, or reliance on operating experience. Testing will be the preferred method of qualification. Qualification testing will be accomplished either by tests on the particular equipment or by type tests performed on similar equipment under environmental conditions at least as severe as the specified conditions. The equipment will be qualified for normal and accident environments. Qualification data will be maintained as part of the permanent plant record in accordance with the NWMI QAPP.

3.5.2.7 Radioisotope Production Facility Specific System Design Basis Functions and Values

The design basis functions and values for each system identified in Table 3-1 are discussed in the following subsections. Additional details for each system described below will be updated during the development of the Operating License Application.

3.5.2.7.1 Target Fabrication System

An overview and detailed description of the target fabrication system are provided in Chapter 4.0, Sections 4.1.3.1 and 4.4, respectively.

Design Basis Functions

- Store fresh LEU, LEU target material, and new LEU targets
- Produce LEU target material from fresh and recycled LEU material
- Assemble, load, and fabricate LEU targets
- Reduce or eliminate the buildup of static electricity
- Minimize uranium losses through target fabrication
- Safety-related functions:
  - Maintain subcriticality conditions within target fabrication system
  - Prevent flammable gas composition within target fabrication system
  - Limit personnel exposure to hazardous chemicals and offgases

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs

3.5.2.7.2 Target Receipt and Disassembly System

An overview and detailed description of the target receipt and disassembly system are provided in Chapter 4.0, Section 4.1.3.2, and Sections 4.3.2/4.3.3, respectively.

Design Basis Functions

- Handle irradiated target shipping cask, including all opening, closing, and lifting operations
- Retrieve irradiated targets from a shipping cask
- Disassemble targets and retrieving irradiated target material from targets
- Reduce or eliminate the buildup of static electricity
- Safety-related functions:
  - Provide radiological shielding during receipt and disassembly activities
  - Maintain subcriticality conditions within target receipt and disassembly system
  - Prevent radiological materials from being released during target receipt and disassembly operations to limit the exposure of workers, the public, and environment to radioactive material
  - Maintain positive control of radiological materials (LEU target material and radiological waste)
  - Protect personnel and equipment from industrial hazards associated with system equipment (e.g., moving parts)

Design Basis Values

- 30-year design life
- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- Crane designed for anticipated load (e.g., hot cell cover block) of approximately 68 metric tons (MT) (75 ton)

3.5.2.7.3 Target Dissolution (DS)

An overview and detailed description of the target dissolution system are provided in Chapter 4.0, Sections 4.1.3.3 and 4.3.4, respectively.

Design Basis Functions

- Fill the dissolver basket with the LEU target material
- Dissolve the LEU target material within dissolver basket
- Treat the offgas from the target dissolution system
- Handle and package solid waste created by normal operational activities
- Safety-related functions:
  - Provide radiological shielding during target dissolution activities
  - Control and prevent flammable gas from reaching lower flammability limit conditions
  - Maintain subcriticality conditions through inherently safe design of target dissolution equipment
  - Maintain positive control of radiological materials (LEU target material and radiological waste)

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- Prevent radiological materials from being released during target dissolution operations to limit the exposure of workers, the public, and environment to radioactive material per 10 CFR 20

3.5.2.7.4 Molybdenum Recovery and Purification (MR)

An overview and detailed description of the Mo recovery and purification system are provided in Chapter 4.0, Sections 4.1.3.4 and 4.3.5, respectively.

Design Basis Functions

- Recovery of Mo product from a nitric acid solution created from dissolved irradiated uranium targets
- Purification of the recovered Mo product to reach specified purity requirements, followed by shipment of the Mo product
- Safety-related functions:
  - Maintain subcriticality conditions through inherently safe design of components that could handle high-uranium content fluid
  - Prevent radiological materials from being released by containing fluids in appropriate tubing, valves, and other components

- Control and prevent flammable gas from reaching lower flammability limit conditions
- Maintain positive control of radiological materials (^{99}Mo product, intermediate streams, and radiological waste)
- Provide appropriate containers and handling systems to protect personnel from industrial hazards such as chemical exposure (e.g., nitric acid, caustic, etc.)

Design Basis Values

- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Replace consumables after each batch

3.5.2.7.5 Uranium Recovery and Recycle (UR)

An overview and detailed description of the uranium recovery and recycle system are provided in Chapter 4.0, Sections 4.1.3.5 and 4.3.6, respectively.

Design Basis Functions

- Receive and decay impure LEU solution
- Recover and purify impure LEU solution
- Decay and recycle LEU solution
- Transfer process waste
- Safety-related functions:
  - Provide radiological shielding during uranium recovery and recycle system activities
  - Prevent radiological release during uranium recovery and recycle system activities
  - Maintain subcriticality conditions through inherently safe design of the uranium recovery and recycle equipment
  - Control and preventing flammable gas from reaching lower flammability limit conditions
  - Maintain positive control of radiological materials
  - Protect personnel and equipment from industrial hazards associated with the system equipment, such as moving parts, high temperatures, and electric shock

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs

3.5.2.7.6 Waste Handling

An overview and detailed description of the waste handling system are provided in Chapter 4.0, Section 4.1.3.6 and Chapter 9.0, Section 9.7.2, respectively.

Design Basis Functions

- Receive liquid waste that is divided into high-dose source terms and low-dose source terms to lag storage
- Transfer remotely loaded drums with high-activity solid waste via a solid waste drum transit system to a waste encapsulation cell

- Encapsulate solid waste drums
- Load drums with solidification agent and low-dose liquid waste
- Load high-integrity containers with solidification agent and high-dose liquid waste
- Handle and load a waste shipping cask with radiological waste drums/containers
- Safety-related functions:
  - Maintain subcriticality conditions by maintaining mass limits
  - Prevent spread of contamination to manned areas of the facility that could result in personnel exposure to radioactive materials or toxic chemicals
  - Provide shielding, distance, or other means to minimize personnel exposure to penetrating radiation

Design Basis Values

- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- 30-year design life with the exception of common replaceable parts (e.g., pumps)

3.5.2.7.7 Criticality Accident Alarm System

Chapter 6.0, Section 6.3.3.1, and Chapter 7.0, Section 7.3.7, provide descriptions of the criticality accident alarm system.

Design Basis Functions

- 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.
- Provide both local and remote annunciation of a criticality excursion
- Remain operational during DBEs

Design Basis Values

- 30-year design life
- Capable of detecting a criticality accident that produces an absorbed dose in soft tissue of 20 absorbed radiation dose (rad) of combined neutron or gamma radiation at an unshielded distance of 2 m from reacting material within one minute (except for events occurring in areas not normally accessed by personnel and where shielding provides protection against a criticality)

3.5.2.7.8 Continuous Air Monitoring System

Chapter 7.0, Section 7.6, and Chapter 11.0, Section 11.1.4, provide detailed descriptions of the RPF continuous air monitoring system.

Design Basis Functions

- Provide real-time local and remote annunciation of airborne contamination in excess of preset limits
- Provide real-time local and remote annunciation of radiological dose of excess of preset limits
- Provide environmental monitoring of nuclear radioactive stack releases
- Provide the capability to collect continuous samples
- Remain operational during DBEs

Design Basis Values

- Activate when airborne radioactivity levels exceed predetermined limits
- Activate when radiological dose levels exceed predetermined limits
- Adjust volume of air sampled to ensure adequate sensitivity with minimum sampling time

3.5.2.7.9 Standby Electrical Power

Chapter 8.0, Section 8.2 provides a detailed description of the RPF standby electrical power (SEP) system.

Design Basis Functions

SEP includes two types of components: uninterruptible power supplies (UPS) and a standby diesel generator:

- **UPS** – Provides power when normal power supplies are absent
- **Standby diesel generator** – Provides power when normal power supplies are absent to allow continued RPF processing

Design Basis Values

- 30-year design life
- Maintain power availability for a minimum of 120 min post-accident (UPS)
- Maintain power availability for 12 hr (diesel generator)

3.5.2.7.10 Normal Electrical Power

Chapter 8.0, Section 8.1 provides a detailed description of the RPF normal electrical power (NEP) system.

Design Basis Functions

- Provide facility power during normal operations

Design Basis Values

- 30-year design life

3.5.2.7.11 Process Vessel Ventilation System

Chapter 9.0, Section 9.1 provides a detailed description of the process vessel ventilation system.

Design Basis Functions

- Provide primary system functions to protect on-site and off-site personnel from radiological and other industrial related hazards
- Collect air in-leakage sweep from each of the numerous vessels and other components in main RPF processes and maintain hydrogen concentration process tanks and piping below lower flammability limit
- Minimize reliance on administrative or complex active engineering controls to provide a confinement system as simple and fail-safe as reasonably possible

Design Basis Values

- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- 30-year design life
- Contain and store noble gases generated in the RPF to meet 10 CFR 20 requirements

3.5.2.7.12 Facility Ventilation System

Chapter 9.0, Section 9.1 provides a detailed description of the facility ventilation system.

Design Basis Functions

- Provide confinement of hazardous chemical fumes and airborne radiological materials and conditioning of RPF environment for facility personnel and equipment
- Prevent release and dispersal of airborne radioactive materials (e.g., maintain pressure gradients to ensure proper flow of air from least potentially contaminated areas to most potentially contaminated areas) to protect health and minimize danger to life or property
- Maintain dose uptake through ingestion to levels as low as reasonably achievable (ALARA)
- Provide makeup air and condition the RPF environment for process and electrical equipment
- Process exhaust flow from the process vessel ventilation system
- Provide confinement of airborne radioactive materials by providing for the rapid, automatic closure of isolation dampers within confinement zones for various accident conditions
- Provide conditioned air to ensure suitable environmental conditions for personnel and equipment in RPF

Design Basis Values

- Maintain primary fission product boundary during and after normal operations, shutdown conditions, and DBEs
- Provide an integrated leak rate for confinement boundaries that meets the requirements of accident analyses that complies with 10 CFR 20 dose limits
- Ensure that air exhausted to the atmosphere meets 40 CFR 61 (NESHAP) and applicable State law
- 30-year design life
- Maintain occupied space at 24 degrees Celsius (°C) (75 degrees Fahrenheit [°F]) (summer) and 22°C (72°F) (winter), with active ventilation to support workers and equipment
- Maintain air quality that complies with 10 CFR 20 dose limits for normal operations and shutdown

3.5.2.7.13 Fire Protection System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF fire protection system.

Design Basis Functions

- Provide detection and suppression of fires
- Generate alarm signals indicating presence and location of fire

- Execute commands appropriate for the particular location of the fire (e.g., provide varying levels of notification of a fire event and transmitting notification to RPF central alarm station and RPF control room)
- Provide fire detection in RPF and initiate fire-rated damper closures
- Remain functional during DBEs

Design Basis Values

- 30-year design life
- Provide a constant flow of water to an area experiencing a fire for a minimum of 120 min based on the size of the area per International Fire Code (IFC, 2012)
- Provide sprinkler systems, when necessary, per National Fire Protection Association (NFPA) 13, *Standard for the Installation of Sprinkler Systems*

3.5.2.7.14 Plant and Instrument Air System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF plant and instrument air system.

Design Basis Functions

- Provide small, advective flows of plant air for several RPF activities (e.g., tool operation, pump power, purge gas in tanks, valve actuation, and bubbler tank level measurement)
- Provide plant air receiver buffer capacity to make up difference between peak demand and compressor capacity
- Provide plant air to instrument air subsystem for bubblers and valve actuation
- Provide instrument air receiver buffer capacity to make up difference between peak demand and compressor capacity

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Provide instrument air dried in regenerable desiccant beds to a dew point of no greater than -40°C (-40°F) and filtered to a maximum 40 micron (μ) particle size

3.5.2.7.15 Emergency Purge Gas System

Chapter 6.0, Section 6.2.1.7.5 provides a detailed description of the emergency purge gas system.

Design Basis Functions

- Provide nitrogen to emergency purge gas system to the required process tanks
- Remain functional during DBEs

Design Basis Values

- 30-year design life with the exception of common replaceable parts
- Maintain hydrogen gas (H_2) concentrations less than flammability limit

3.5.2.7.16 Gas Supply System

Chapter 9.0, Section 9.7.1 provides a detailed description of the gas supply system.

Design Basis Functions

- Provide helium, hydrogen, and oxygen in standard gas bottles
- Provide nitrogen from a tube truck to the chemical supply room where manifold piping will be used to distribute the gas
- Provide adequate flow to ensure that the accumulation of combustible gases is below hazardous concentrations and reduces radiological hazards due to accumulation of gaseous fission products

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Provide standard gas bottles, with capacity of approximately 8,495 L (300 cubic feet [ft^3])

3.5.2.7.17 Process Chilled Water System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF chilled water system.

Design Basis Functions

- Provide process chilled water loop for three secondary loops through plate-and-frame heat exchangers
  - One large geometry secondary loop in hot cell
  - One criticality-safe geometry secondary loop in hot cell
  - One criticality-safe geometry secondary loop in target fabrication area
- Provide monitoring of chilled water loops for loss of primary containment
- Provide cover gas to prevent flammable conditions

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Chilled water to various process equipment at no greater than 10°C (50°F) during normal operations
- The hydrogen concentration in the coolant system is maintained at less than 25 percent of the lower flammability limit of 5 percent H_2

3.5.2.7.18 Facility Chilled Water System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF facility chilled water system.

Design Basis Functions

- Provide cooling media to heating, ventilation, and air conditioning (HVAC) system
- Supply HVAC system with cooling water that is circulated through the chilled water coils in air-handling units

Design Basis Values

- Provide cooling water at a temperature of 9°C (48°F) to the HVAC air-handling unit cooling coils
- 30-year design life with the exception of common replaceable parts (e.g., pumps)

3.5.2.7.19 Facility Heated Water System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF heated water system.

Design Basis Functions

- Provide heated media to HVAC system
- Supply the HVAC system with heated water that is circulated through the heated water coils in the air-handling units

Design Basis Values

- Provide heated water at a temperature of 82°C (180°F) to HVAC air-handling unit heating coils and reheat coil
- 30-year design life with the exception of common replaceable parts (e.g., pumps)

3.5.2.7.20 Process Steam System – Boiler

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF process steam system for the boiler.

Design Basis Functions

- Generate low- and medium-pressure steam using a natural gas-fired package boiler
- Provide a closed loop steam system for the hot cell secondary loops that meets criticality control requirements
- Provide monitoring of steam condensate for loss of primary containment
- Limit sludge or dissolved solids content with automatic and makeup water streams in the boiler

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Provide saturated steam at 1.7 kg/square centimeters (cm^2) (25 lb/square inch [$in.^2$]) and 4.2 kg/cm^2 (60 $lb/in.^2$) gauge to various process equipment

3.5.2.7.21 Demineralized Water System

Chapter 9.0, Section 9.7.1 provides a detailed description of the RPF demineralized water system.

Design Basis Functions

- Provide demineralized water to RPF except for administration and truck bay areas
- Remove mineral ions from municipal water through an ion exchange (IX) process and accumulate in a storage tank
- Provide regenerable IX media using a strong acid and a strong base
- Feed acids and bases from local chemical drums by toe pumps

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)
- Provide the water at 4.2 kg/cm^2 (60 $lb/in.^2$) gauge

3.5.2.7.22 Chemical Supply System

Chapter 9.0, Section 9.7.4 provides a detailed description of the chemical supply system.

Design Basis Functions

- Provide storage capability for nitric acid, sodium hydroxide, reductant, and nitrogen oxide absorber solutions, hydrogen peroxide, and fresh uranium IX resin
- Segregate incompatible chemicals (e.g., acids from bases)
- Provide transfer capability for chemical solutions mixed to required concentrations and used in target fabrication, target dissolution, Mo recovery and purification, and waste management systems

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., pumps)

3.5.2.7.23 Biological Shielding System

Chapter 4.0, Section 4.2, provides a detailed description of the RPF biological shielding.

Design Basis Functions

- Provide biological shielding from radiation sources in the hot cells for workers in occupied areas of the RPF
- Limit physical access to hot cells
- Remain functional through DBEs without loss of structural integrity

Design Basis Values

- 30-year design life
- Provide dose rates consistent with ALARA goals for normally occupied areas

3.5.2.7.24 Facility Process Control System

Chapter 7.0, Section 7.2.3 provides a description of the FPC system.

Design Basis Functions

- Perform as overall production process controller
- Monitor and control process instrumented functions within the RPF (e.g., process fluid transfers, controlled inter-equipment pump transfers of process fluids)
- Provide monitoring of safety-related components while BMS (a subset of the FPC system) monitors ventilation system and mechanical utility systems
- Ensure ESF systems operate independently from FPC system or BMS
- Use hard-wired analog controls/interlocks for each ESF safety function to protect workers, public, and environment
- Integrate into and monitor ESF parameters and alarm functions by FPC system or BMS
- Initiate actuation of isolation dampers for hot cell area or analytical area on receipt of signals from fire protection system

Design Basis Values

- 30-year design life with the exception of common replaceable parts (e.g., controllers)

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