CHAPTER 4

IRRADIATION UNIT AND RADIOISOTOPE PRODUCTION FACILITY DESCRIPTION

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Acronyms and Abbreviations

Acronym/Abbreviation	Definition
°C	degrees Celsius
°F	degrees Fahrenheit
µg/batch	micrograms per batch
µg/TSV batch	micrograms per TSV batch
µTorr	microtorr
AAEC	Australian Atomic Energy Commission
acfm	actual cubic feet per minute
ACI	American Concrete Institute
AGS	American Glovebox Society
AHA	acetohydroxamic acid
AHR	aqueous homogenous reactor
ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
[Proprietary Information]	[Proprietary Information]
Be	beryllium
BPVC	Boiler and Pressure Vessel Code
Btu	british thermal unit
Btu/hr	british thermal units per hour
Btu/hr-gal	british thermal units per hour per gallon
Btu/lb-°F	british thermal units per pound per degree Fahrenheit

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
CAAS	criticality accident alarm system
CAMS	continuous air monitoring system
CFD	Computational Fluid Dynamics
cm	centimeter
Ci	curie
D	deuterium
DBEQ	design basis earthquake
DI	deionized
DOE	U.S. Department of Energy
DPTE	double door for leaktight transfer
ECR	electron cyclotron resonance
EMI	electromagnetic interference
ESF	engineered safety feature
FBR	fluidized bed reactor
fl. oz.	fluid ounce
FPGA	field programmable gate array
FSAR	Final Safety Analysis Report
ft.	foot
ft/sec	feet per second
g	gram
g/cm^3	grams per cubic centimeter
g/L	grams per liter
g/TSV batch	grams per TSV batch
GHz	gigahertz

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
gpm	gallons per minute
gU/L	grams of uranium per liter
H_2	hydrogen gas
H_2SO_4	sulfuric acid
HNO_3	nitric acid
hp	horsepower
HVAC	heating, ventilation and air conditioning
HVPS	high voltage power supply
I	iodine
IF	irradiation facility
in.	inch
in. w.c.	inches of water column
in^3/min	cubic inches per minute
IU	irradiation unit
k_{eff}	effective neutron multiplication factor
kg	kilogram
kg/batch	kilograms per batch
kg/h	kilograms per hour
kg/L	kilograms per liter
KI	Kurchatov Institute
$KMnO_4$	potassium permanganate
kPa	kilopascal
Kr-85	krypton-85

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
Kr-85m	krypton-85 metastable
kV	kilovolt
kW	kilowatt
kWt	thermal kilowatt
kWt/L	thermal kilowatts per liter
L	liter
L/h	liters per hour
L/s	liters per second
L/TSV batch	liters per TSV batch
LANL	Los Alamos National Laboratory
lb.	pound
lb/batch	pounds per batch
lb/ft^3	pounds per cubic foot
lb/gal	pounds per gallon
lb/hr	pounds per hour
lb/in^3	pounds per cubic inch
lb/TSV batch	pounds per TSV batch
lbU/gal	pounds of uranium per gallon
LCO	limiting conditions of operations
LEU	low-enriched uranium
LFL	lower flammability limit
LOOP	loss of off-site power
lpm	liters per minute
LSSS	limiting safety system setting

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
LWPS	light water pool system
LWR	light water reactor
m	meter
m/s	meters per second
M	molar (moles per liter)
mA	milliamperes
mPa	millipascal
MCNP	Monte-Carlo N-Particle Transport Code
MEPS	molybdenum extraction and purification system
MeV	million electron volt
MeV/cm^2-sec	megaelectron volts per square centimeter per second
mg	milligram
mg/mL	milligram per milliliter
MIPS	molybdenum isotope product packaging system
mL	milliliter
mm	millimeter
Mo	molybdenum
Mo-99	molybdenum-99
mrem/hr	millirem per hour
n/s	neutrons per second
n/cm^2	neutrons per square centimeter (neutron fluence)
n/cm^2-sec	neutrons per square centimeter per second (neutron flux)

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
NaOH	sodium hydroxide
NDAS	neutron driver assembly system
NFDS	neutron flux detection system
NGRS	noble gas removal system
NH_4OH	ammonium hydroxide
NO_x	nitrogen oxide
NPH	natural phenomena hazard
NPSS	normal electric power supply system
NRC	U.S. Nuclear Regulatory Commission
NRTL	Nationally Recognized Testing Laboratory
OLWS	organic liquid waste storage and export
ORIGEN	Oak Ridge Isotope Generation
ORNL	Oak Ridge National Laboratory
oz/fl oz	ounce per fluid ounce
pcm	percent millirho
pcm/°C	percent millirho per degree Celsius
pcm/%void	percent millirho per percent void
pcm/(liter of water)	percent millirho per liter of water
PCLS	primary closed loop cooling system
PFBS	production facility biological shield system
PFD	process flow diagram
PNL	Phoenix Nuclear Laboratory
PPE	personal protective equipment
ppm	parts per million

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
PSB	primary system boundary
psi	pounds per square inch
psig	pounds per square inch gage
Pu	plutonium
Pu-239	plutonium-239
PUREX	plutonium uranium extraction
PVVS	process vessel vent system
RAMS	radiation area monitoring system
RCA	radiologically controlled area
RDS	radioactive drain system
RLWE	radioactive liquid waste evaporation and immobilization
RLWS	aqueous radioactive liquid waste storage
RPCS	radioisotope process facility cooling system
RPF	radioisotope production facility
RSICC	Radiation Safety Information Computational Center
RVZ1	Radiological Controlled Area Ventilation System Zone 1
SASS	subcritical assembly support structure
SCALE	Standardized Computer Analyses for Licensing Evaluations
SCAS	subcritical assembly system
sccm	standard cubic centimeters per minute
scfm	standard cubic feet per minute
SHINE	SHINE Medical Technologies, Inc.

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
SL	safety limit
slpm	standard liters per minute
Sm-149	samarium-149
SNM	special nuclear material
SR	surveillance requirement
Sr-90	strontium-90
[Proprietary Information]	[Proprietary Information]
SRNL	Savannah River National Laboratory
SRWP	solid radioactive waste packaging
SSC	structure, system, and component
T	tritium
TBP	tri-butyl phosphate
Tc	technetium
Tc-99	technetium-99
TDN	thermal denitration
TK	tank
TLD	thermoluminescent dosimeter
TOGS	TSV off-gas system
TPCS	TSV process control system
TPS	tritium purification system
TRPS	TSV reactivity protection system
TS	target solution
TSPS	target solution preparation system
TSV	target solution vessel

Acronyms and Abbreviations (cont'd)

Acronym/Abbreviation	Definition
U	uranium
U-233	uranium-233
U-235	uranium-235
U-238	uranium-238
UN	uranyl nitrate
UNCS	uranyl nitrate conversion system
UNP	uranyl nitrate preparation
UO_x	uranium oxide
UPSS	uninterruptible electrical power supply system
UREX	uranium extraction
vol%	volume percent
W	watt
wt%	weight percent
Xe-133	xenon-133
Xe-135	xenon-135
yd^3	cubic yards

CHAPTER 4

IRRADIATION UNIT AND RADIOISOTOPE PRODUCTION FACILITY DESCRIPTION

4a1 HETEROGENEOUS REACTOR DESCRIPTION

The SHINE Medical Technologies, Inc. (SHINE) facility is not a reactor-based facility; therefore, this section does not apply to the SHINE facility.

4a2 IRRADIATION FACILITY DESCRIPTION

This section describes the SHINE irradiation units (IU) and supporting subsystems used for the irradiation of uranyl sulfate target solution as part of the irradiation facility (IF).

4a2.1 SUMMARY DESCRIPTION

An IU is an accelerator-driven subcritical operating assembly used for the irradiation of an aqueous uranyl sulfate target solution, resulting in the production of molybdenum-99 (Mo-99) and other fission products. An accelerator is used to create deuterium-tritium fusion reactions, resulting in the formation of 14 million electron volt (MeV) neutrons. These high-energy neutrons cause various multiplying reactions in the neutron multiplier, which increase the neutron population entering the target solution vessel (TSV). The neutron population in the TSV leads to fissioning of the uranium solution. Without operation of the accelerator, the fission process essentially terminates. Key operating parameters for the IU are provided in Table 4a2.1-1.

An IU cell is comprised of the following (see Figure 4a2.1-1):

- Biological shielding: The IUs are surrounded by a biological shield that protects facility workers from radiation levels inside the IU cell. A detailed description of the biological shield is provided in Section 4a2.5.

- Neutron driver assembly system (NDAS): An accelerator-based assembly is used to produce 14 MeV neutrons by the fusion of deuterium and tritium in the tritium target chamber. These neutrons then drive the fission reactions required for the production of Mo-99 and other fission products. Incorporated into the neutron driver is ample deuterium storage to allow for the normal operation of the driver between scheduled maintenance cycles. The neutron driver is suspended above the subcritical assembly on an operating platform attached to the IU cell walls and is intended to be regularly replaced over the operating life of the facility. A detailed description of the neutron driver is provided in Section 4a2.3.

- Light water pool system (LWPS): The LWPS serves two primary functions: shielding and cooling. The light water pool is the safety-related feature provided for heat removal from the subcritical assembly. In the event of a failure of the primary closed loop cooling system (PCLS), the thermal mass provided by the light water pool provides sufficient decay heat removal capacity to provide cooling for the TSV when the neutron driver is shut down. The light water pool also provides direct cooling to the neutron multiplier and tritium target chamber (neutron driver). The LWPS cooling loop circulates the pool water through a heat exchanger to maintain the normal operating temperature of the light water pool. See Section 5a2.2 for more information.

  Finally, the light water pool provides shielding necessary to allow manned entry to the IU cells when the IU is shut down, and reduces biological shield wall thickness requirements.

- Subcritical assembly system (SCAS): The SCAS consists of the TSV, neutron multiplier, subcritical assembly support structure (SASS), subcritical multiplication source, and other components to safely contain the target solution during the irradiation process.

- Neutron flux detection system (NFDS): Neutron flux monitors are located as shown in Figure 4a2.1-2. Details of the NFDS are provided in Subsection 7a2.4.3.

The systems supporting the IU cell include:

- Tritium purification system (TPS): The TPS is located outside the IU cell. The neutron driver is connected to the common tritium system by supply and return lines that pass through the IU cell wall boundary. During operation, purified tritium gas is supplied to the tritium target chamber, and mixed gas is returned to the tritium system for purification and re-use. A detailed description of the tritium system is discussed in Subsection 9a2.7.1.

- TSV off-gas system (TOGS): The majority of this system's components, purposed to maintain the hydrogen concentrations in the TSV below the lower flammability limits (LFL), are located outside, but adjacent to the IU cell. A detailed description of the TOGS is provided in Section 4a2.8.

- Primary closed loop cooling system (PCLS): The PCLS is an active coolant system designed to remove heat from the TSV during IU operation. The PCLS includes filters to remove activation products, plate-type heat exchangers for rejection of heat to the secondary coolant system, and deionizer units. This system also provides a fission product boundary between the TSV and the light water pool. A detailed description of the PCLS is provided in Section 5a2.2.

Subcritical Assembly

Each IU cell contains one neutron driver, which is coaxially located with the subcritical assembly. The neutron driver accelerates a deuteron beam into a tritium target chamber, which is located in the center of the multiplier annulus. The deuteron beam impinging on the tritium gas yields a neutron source with a shape similar to a vertical line distribution along the axis of the tritium target chamber. A small amount of tritium gas travels up the spacer tube leading to a small fraction of the reactions occurring above the target chamber. These fusion neutrons then move outward in all directions, with a nearly isotropic spectrum.

As the neutrons move outward, most of them enter the subcritical assembly. While the individual path of any single neutron is complicated, the overall effect of the subcritical assembly is to multiply the fusion neutrons, slow them down to thermal energy, and cause them to fission uranium-235 (U-235) atoms.

The multiplication of the neutrons happens in two primary locations: the neutron multiplier and the TSV. The neutron multiplier produces additional neutrons primarily through [Proprietary Information] reactions, while the TSV produces additional neutrons primarily through fission reactions. The slowing down of the neutrons occurs primarily in the TSV and the surrounding light water pool.

The subcritical assembly is composed of the following:

- Neutron multiplier: The neutron multiplier is located in the space between the neutron driver tritium target chamber and the target solution vessel. It is comprised of a [Proprietary Information], encapsulated within an aluminum jacket. A detailed description of the neutron multiplier is provided in Subsection 4a2.2.6.

- Target solution vessel: The TSV is the zircaloy-4 vessel that contains the uranyl sulfate target solution undergoing irradiation for the production of Mo-99 and other fission products. It is located within the SASS pressure boundary. A detailed description of the target solution vessel is provided in Section 4a2.2.

- Subcritical assembly support structure: The seismically qualified SASS establishes the location of the other subcritical assembly components relative to the bottom of the light water pool, and in relation to the neutron driver. A detailed description of the SASS is provided in Subsection 4a2.2.5. In addition, the SASS surrounds the TSV. While the SASS is not a complete second fission product boundary for the entire system, it does provide an additional degree of defense-in-depth for the components exposed to the highest neutron fluxes and most unique chemistry in the system.

  Cooling water supplied by the primary closed loop cooling system (PCLS) flowing through the subcritical assembly is the primary heat removal mechanism for the subcritical assembly during neutron driver operation, but forced convection of the coolant is not a safety-related function. In the event of a failure of the PCLS pump, the light water pool provides sufficient thermal capacity to remove the heat from a static primary loop after the shutdown of the neutron driver, and is credited as providing passive decay heat removal for design basis accidents.

- Subcritical multiplication source: The subcritical multiplication source is a fixed neutron source in the subcritical assembly that is used to monitor the state of the reactivity of the assembly when the neutron driver is not producing neutrons. This is a small neutron source (in comparison to the neutron driver) that facilitates the performance of 1/M measurements during the startup process. It provides a stable level of background neutrons so that neutron multiplication in the subcritical assembly can be accurately and reliably measured while filling the TSV with target solution. Details of the subcritical multiplication source are provided in Subsection 4a2.2.4.

- TSV dump valves: The TSV dump valves are provided to drain the TSV, whether as part of the process prior to transferring the target solution downstream for processing, or as a safety-related feature utilized as part of a planned response in the event of an IU abnormal condition. The valves are located in redundant flow paths and fail to a safe (open) position. Both the TSV reactivity protection system (TRPS) and TSV process control system (TPCS) can open both dump valves.

- TSV dump tank: The TSV dump tank is normally utilized as a hold tank for decay prior to transfer to the radioisotope production facility (RPF). The TSV dump tank is physically located below the TSV to allow for the gravity transfer of the target solution from the TSV to a passively cooled, criticality-safe storage geometry in the event of an abnormal condition or accident scenario.

  This assembly is supported from the floor of the light water pool, directly beneath the neutron driver, and interfaces with (but is not directly coupled to) the tritium target chamber. The subcritical assembly is designed such that routine replacement of its constituent components is not necessary.

Table 4a2.1-1 Irradiation Unit Key Operating Parameters

Operating Parameter	Value and/or Description
Thermal Power Level of the TSV	Up to [Proprietary Information]
Accelerator Neutron Production Rate	[Proprietary Information] per second
Neutron Flux in the TSV	[Proprietary Information] n/cm^2-sec during irradiation
Neutron Multiplier Material	[Proprietary Information], aluminum-clad, [Proprietary Information].
Target Solution Uranium Type and Enrichment	Aqueous uranyl sulfate solution, [Proprietary Information] uranium concentrations, 19.75% U-235 enrichment.
Primary Cooling System Configuration	Forced convection cooling (de-ionized water), submersed in a heat-removing light water pool.
Moderator Type	Light water
Reflector Type	Light water
Operating Type	Subcritical steady-state operation. Sustained nuclear reaction is not possible without the neutron addition provided by the accelerator when operated within technical specification limits.
Negative Temperature Coefficient	Results in inherently limited power excursions (decreased reactivity with a power increase).
Negative Void Coefficient	Results in inherently limited power excursions (decreased reactivity with a power increase).
TOGS	Maintains radiolytic hydrogen concentrations in the TSV below the LFL.

4a2.2 SUBCRITICAL ASSEMBLY

The primary systems that comprise an IU are an NDAS and an SCAS. The neutron driver is an accelerator-based neutron generator suspended above the light water pool and is described in detail in Section 4a2.3.

The subcritical assembly is submerged in the light water pool, centered on a vertical centerline shared with the neutron driver. The subcritical assembly is primarily comprised of the TSV that contains the target solution during the irradiation process and a neutron multiplier, both of which are supported and positioned by the SASS. See Figure 4a2.1-2 for a depiction of the subcritical assembly.

4a2.2.1 TARGET SOLUTION

4a2.2.1.1 Description

The target solution is a uranyl sulfate solution that is loaded into the TSV to be irradiated. This subsection describes the chemical and radiological characteristics of the target solution and its interface with TSV-related systems. Descriptions of the preparation, storage, and handling of the target solution can be found in Sections 9a2.2 and 9b.2 and Subsections 4b.4.1 and 4b.4.2.

4a2.2.1.2 Chemical Properties

Table 4a2.2-1 shows the chemical and physical properties of the fresh/recycled target solutions loaded into the TSV.

4a2.2.1.3 Isotope Properties

Table 4a2.2-2 shows the radionuclide inventory for a single TSV, [Proprietary Information] after end of irradiation, for select isotopes in the target solution that has been irradiated for [Proprietary Information] cycles.

There are no significant quantities of fission products or transuranics in the fresh target solution because it has not been irradiated.

4a2.2.1.4 Primary System Boundary

The primary system boundary (PSB) includes the TSV, TOGS, TSV dump tank, and associated components. The PSB provides the primary fission product barrier between the target solution and the surrounding area. PSB components are designed to be compatible with the target solution to avoid corrosion and other unwanted metallurgical effects that could lead to the PSB being compromised. The TSV is located in the IU cell, which provides confinement protection to the facility, workers, and the public.

4a2.2.1.5 Off-Gas Formation

During target solution irradiation, noble gases are formed by fission and decay and hydrogen and oxygen are formed by radiolysis. The hydrogen concentration in the TSV headspace is maintained below LFL by volume by using air as a sweep gas. The off-gas is passed through

silver zeolite packed beds to remove approximately 95 percent of the iodine isotopes and then through a catalytic recombiner to recombine the hydrogen with oxygen before being circulated back to the TSV.

A detailed description of the TOGS can be found in Section 4a2.8.

The formation rate of hydrogen and oxygen is approximately up to [Proprietary Information] standard cubic feet per minute (scfm) ([Proprietary Information] standard liters per minute [slpm]) of 67 percent hydrogen and 33 percent oxygen. The overall circulation rate of the TOGS is approximately [Proprietary Information] pounds per hour (lb/hr) ([Proprietary Information] kilograms per hour [kg/h]) to normally maintain the hydrogen concentration below 2 percent by volume.

The PSB is a closed system during irradiation and water is retained within the system. The condensate return to the TSV is approximately [Proprietary Information] and [Proprietary Information].

The headspace within the TSV when the target solution is present is between 9 percent and 32 percent depending on the fill volume determined during the startup process.

4a2.2.1.6 TSV Operating Conditions

The normal operating conditions within the TSV target solution are 68 to 176 degrees Fahrenheit (°F) (20 to 80 degrees Celsius [°C]) and approximately -0.5 inches of water column (in. w.c.) (-0.1 kilopascal [kPa]). See Section 4a2.8 for more information on TSV off-gas operating conditions. Operating specification limits of the target solution are discussed in Subsection 4a2.2.1.7

There is no mechanical mixing in the TSV. The target solution is mixed using natural convection and bubbling during irradiation. The temperature of the target solution rises from room temperature (approximately 68°F [20°C] to approximately 140°F [60°C]) during full-power irradiation. The pressure of the TSV remains constant at the pressure of the TOGS circulation loop. There is no precipitation out of the target solution. There are no significant pH changes expected during irradiation. The bulk void fraction within the target solution is expected to be less than 5 percent. The target solution has a negative void coefficient and increases in void leads to lower power in the TSV.

4a2.2.1.7 Target Solution Operating Limits

The operating limits of the target solution are shown in Table 4a2.2-3.

4a2.2.1.8 Sampling and Testing

The volume, uranium density, and pH of the target solution are determined and set in the target solution preparation system, in the uranyl sulfate preparation tank (1-TSPS-01T). Sampling and analysis of the target solution in the target solution hold tank (1-TSPS-03T) verifies that the target solution is within specification before being transferred to the TSV. See Section 9b.2 for more target solution preparation information.

Target solution is sampled and analyzed from the target solution hold tank prior to being loaded into the TSV to ensure the required chemical property specifications are met. Operators make adjustments to the target solution to meet specifications, as necessary.

SCAS instrumentation monitors the temperature of the target solution during irradiation to ensure safe operation.

4a2.2.1.9 Chemical and Physical Changes in Target Solution

Physical changes in the target solution as the irradiation cycle proceeds during operation are generally small. Refer to Table 4a2.2-3 for the list of chemical property operating range limits of the target solution.

Short term chemical and physical property changes of the target solution are limited to temperature, concentration, and fission gas release. There is a small buildup of fission and activation products in the target solution during irradiation. Calculations indicate that plutonium buildup and fission product poisons have a minimal effect on this system due to the low power and short irradiation cycles.

There is no significant change related to pH in the target solution during irradiation. The temperature of the target solution rises from room temperature (approximately 68°F [20°C] to approximately 140°F [60°C]) during irradiation. The concentration of the target solution is slightly affected by the holdup of hydrogen, oxygen, and water in the TOGS piping during operation. The TOGS will be designed such that the holdup of water is less than 1 percent of the total TSV volume. Radiolytic fission gas is produced during irradiation (hydrogen, oxygen, water vapor) and is drawn into the TOGS to recombine the hydrogen and oxygen and return condensate back to the TSV to limit water loss during operation.

After each irradiation cycle, the target solution is sent to the molybdenum-99 extraction and purification system (MEPS) for removal of medical isotopes. After the target solution has been through the irradiation cycle [Proprietary Information] times it is reconstituted in the uranyl nitrate conversion system (UNCS). The spent uranyl sulfate target solution is reacted to uranyl nitrate and is processed through the uranium extraction (UREX) system and thermal denitration to generate uranium oxide (UO_x) to be used in creating new target solution batches. There are no long term chemical or physical effects that are a detriment to the uranyl sulfate target solution. This is supported by research conducted by Argonne National Laboratory (ANL) (ANL, 2012). See Subsection 4b.4.1 for more information on the UNCS.

4a2.2.1.10 TSV Physical Structure

The TSV is sized to allow for up to [Proprietary Information] per TSV batch (gal/TSV batch) ([Proprietary Information] per TSV batch [L/TSV batch]) of target solution to be irradiated. Up to 1.9 gallons (gal.) (7 liters [L]) are contained within the TSV dump lines during irradiation. The TSV inner radius is approximately [Proprietary Information], the outer radius is approximately [Proprietary Information], and the height of the target solution is between [Proprietary Information]. The TSV contains the uranyl sulfate, sulfuric acid, and fission products during operation and credible accidents. The TSV is capable of withstanding the pressure excursions encountered during a credible deflagration of the radiolytic gases.

The TSV is constructed of zircaloy-4, an alloy of zirconium that offers exceptional corrosion resistance under irradiation and a low neutron absorption cross section (around 0.1 barns for thermal neutrons, and around 0.01 barns for fast neutrons). Zircaloy-4 is widely used throughout the nuclear industry where corrosion resistance and neutron economy are important. The solution also comes in contact with stainless steel piping and vessels. Historical data has shown the compatibility of the utilized stainless steel and the zirconium alloys with uranyl sulfate solutions. SHINE will perform testing to validate acceptable operating conditions for material and target solution compatibility.

The TSV is cooled through both side walls, [Proprietary Information] the external walls of the TSV are actively cooled by the PCLS during normal operations.

Detailed material and structural information for the TSV and auxiliary components can be found in the other subsections of Section 4a2.2. These subsections include discussions on dimensions, materials of construction, and special features (reflectors, geometrical designs) that relate to safety and the physical integrity of the TSV.

4a2.2.1.11 Physical Properties Significant to Safety

Physical properties significant for safety of the target solution that are significant to the thermal hydraulic analyses include:

- Solution density (see Table 4a2.2-1).
- Power density (see Table 4a2.2-1).
- Temperature (see Table 4a2.2-1).
- pH (see Table 4a2.2-1).
- Pressure (see Subsection 4a2.2.1.6).
- Heat capacity (see Table 4a2.2-1).
- Gas evolution (see Subsection 4a2.2.1.5).
- Changes in void fraction (see Subsection 4a2.2.1.6).
- Precipitation of uranium or fission product complexes (see Table 4a2.2-1).
- Sweep gas (see Subsection 4a2.2.1.5).

4a2.2.1.12 Target Solution Preparation

The uranyl sulfate target solution is derived from dissolving low-enriched uranium (LEU) (19.75 ± 0.2 percent U-235) metal in 12 molar (M) nitric acid, which is then processed into uranium oxide by the thermal denitration subsystem to UNCS. The uranium oxide is then dissolved in sulfuric acid to form the target solution. The preparation of target solution is discussed further in Section 9b.2.

Trace amounts of plutonium-239 (Pu-239) are present in the recycled target solution because it has been previously irradiated. Table 4.2.2-2 shows the predicted amount of Pu-239 present in the recycled target solution TSV batches.

4a2.2.1.13 Target Solution History

Concept designs of solution reactors utilizing uranyl sulfate as fuel were first proposed in the 1940s (IAEA, 2008). Both Oak Ridge National Laboratory (ORNL) and Los Alamos National Laboratory (LANL) initiated research activities on uranyl sulfate during that period (Secoy, 1948; Baker, 1944). These national laboratory research activities on uranyl sulfate continued during the 1950s, (ORNL, 1952; Lee, 1952; Lane, 1958; Beall, 1954). Concurrent with the research performed at the national laboratories, uranyl sulfate research was performed at universities with research reactors (NCSU, 1955), Atomics International (Wilson, 1958), and elsewhere (Parkins, 1958; Secoy, 1955; Silverman, 1961; Gamble, 1959). Experience has demonstrated that uranyl sulfate can be used in fission systems successfully and safely. Research on uranyl sulfate as a fuel solution has since continued and supports the concept of using uranyl sulfate as a target solution for the SHINE medical isotope facility.

4a2.2.1.14 Technical Specifications

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the target solution are provided in Chapter 14.

4a2.2.2 REACTIVITY CONTROL MECHANISMS

There are seven variables affecting reactivity that are controlled by the SHINE design. Three, once established during filling of the TSV, are not significantly altered during IU operation:

- Uranium concentration in solution
- Uranium enrichment
- TSV fill volume

Four factors that are controlled to ensure that they remain stable during operational modes are:

- Target solution temperature
- Target solution pressure
- Temperature of the light water pool
- Temperature of the PCLS

A detailed discussion of the systems used to control system reactivity is provided in Subsection 7a2.4.

During TSV operation, it is possible to open the TSV dump valves, and gravity drain the entire contents of the TSV to TSV dump tank (1-SCAS-01T). The TSV dump tank (1-SCAS-01T) has been designed to be criticality-safe for the most reactive uranium concentrations, regardless of temperature, and has sufficient capacity to hold the entire contents of the target solution hold tank (1-TSPS-03T).

The concentration of uranium in solution is measured and independently verified to ensure that concentration values remain within the limits prescribed by SHINE. Uranium concentration is prepared and measured to ensure it is within 1 percent of desired concentration. The SHINE system provides a predictable and precisely controlled system response as the TSV fill volume rises above a fill height of approximately [Proprietary Information]. Final monitoring and

verification of target solution occurs in the target solution hold tank (1-TSPS-03T), prior to its transfer into the TSV. Target solution characteristics and allowable operating ranges are discussed in Subsection 4a2.2.1.

The uranium enrichment is verified when received, and no means are provided to increase the enrichment in the process design. The allowable operating ranges are identified in Subsection 4a2.2.1.

A number of design features are provided to establish TSV fill-volumes:

Fixed TSV configuration: The volume and geometry of the TSV are known and fixed.

Level indication instrumentation: Instrumentation provides an inferred measurement of the TSV fill volume.

Feed pump transfer volume settings: The target solution feed pumps are used to transfer target solution from the target solution hold tank (1-TSPS-03T) to the TSV (1-SCAS-01S). Pump performance characteristics, line hold-up data calculated and verified during commissioning testing, and control system architecture are used to estimate a TSV fill volume, which is confirmed by liquid level instrumentation.

The elevations of the target solution hold tank (1-TSPS-03T), TSV (1-SCAS-01S), and TSV dump tank (1-SCAS-01T) are established to preclude the gravity transfer of target solution into the TSV (see Figure 4a2.2-1). Positive action by the operator and control systems are required by the target solution feed pump to add target solution to the TSV.

Reactivity measurements: The flux monitors are used to determine system reactivity during filling of the TSV. Results are evaluated against calculated values to verify technical specification limits on system reactivity parameters are not exceeded, and yet ensure that the variables affecting molybdenum (Mo) production are optimized to the extent practicable by ensuring consistent cold effective neutron multiplier (k_{eff}) values during the fill process.

The design of the TSV prevents pressurization of the target solution during normal operations. The TOGS maintains the pressure over the target solution negative with respect to atmospheric. The pressure over the target solution is controlled by the TSV off-gas blower (1-TOGS-01C) that removes air and fission gases from the headspace of the TSV, and recirculates it back to the TSV (see Figure 4a2.2-2) after it has been processed through the catalytic recombiner. Details of the TOGS are provided in Section 4a2.8.

The liquid condensed by the TOGS is returned to the TSV via gravity drains to minimize the change in uranium concentration.

The PCLS is monitored and controlled to ensure that the temperature remains stable within the system's established operating limits. The PCLS is used to remove approximately up to [Proprietary Information] of heat from the TSV during full power irradiation.

In the event of a loss of PCLS active heat removal capacity due to loss of flow or increased temperature above acceptable limits, TRPS initiates a dump of the solution and de-energizes the neutron driver high voltage power supply (HVPS). Decreased heat removal would result in a

reactivity decrease, along with a drop of power in the target solution. Details of the PCLS are provided in Section 5a2.2. PCLS temperatures below acceptable limits result in a TRPS trip.

The large thermal mass of the light water pool serves to stabilize temperature effects by conduction through the SASS. PCLS flow rates and PCLS cooler (1-PCLS-01A) outlet temperatures are maintained stable during startup and irradiation modes of operation.

The TRPS is described in Section 7a2.4.

Additional information with respect to final operating parameters, design margins, and allowable temperature variability will be provided in the FSAR.

4a2.2.2.1 Technical Specifications

Potential variables, conditions or other items that will be probable subjects of a technical specification associated with the target solution are provided in Chapter 14.

4a2.2.3 NEUTRON MODERATOR AND REFLECTOR

The light water pool, which surrounds the TSV, provides neutron moderation and reflection to reduce leakage from the system. The materials of construction, geometry, and other design considerations for the light water pool are described in Section 4a2.4.

The neutron multiplier is a [Proprietary Information] that serves to improve the neutron population in the TSV. The design features of the multiplier are described in Subsection 4a2.2.6.

No additional neutron moderators or reflectors are included in the design.

4a2.2.3.1 Technical Specifications

There are no potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the neutron moderator.

4a2.2.4 SUBCRITICAL MULTIPLICATION SOURCE

The subcritical multiplication source is a fixed neutron source in the subcritical assembly that is used to monitor the state of the reactivity of the assembly when the neutron driver is not producing neutrons. This is a neutron source with an output several orders of magnitude less than the neutron driver that facilitates the performance of 1/M measurements during the startup process. It provides a stable level of background neutrons so that neutron multiplication in the subcritical assembly can be accurately and reliably measured while filling the TSV with target solution.

The neutron driver produces an excessively large neutron flux for startup measurements and does not have the desired degree of stability to produce the accuracy required.

4a2.2.4.1 Type of Nuclear Reaction

This information will be provided in the FSAR. The source is expected to be an alpha-emitting radioisotope coupled with a neutron-emitting light isotope, such as plutonium (Pu) – beryllium (Be).

4a2.2.4.2 Energy Spectrum of Neutrons

The information for this subsection will be provided in the FSAR.

4a2.2.4.3 Source Strength

The exact source strength required will depend on the neutron detectors chosen and source placement. The information for this subsection will be provided in the FSAR.

4a2.2.4.4 Interaction with the System

The exact placement of the source has not been determined yet. Two areas being considered are in the subcritical assembly support plate (below the tritium target chamber) or in [Proprietary Information].

4a2.2.4.4.1 Physical Environment

Information on the chemical, thermal, and radiation environment will be provided in the FSAR.

4a2.2.4.5 Verification of Integrity and Performance

The information for this subsection will be provided in the FSAR.

4a2.2.4.6 Technical Specifications

There are no potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the subcritical multiplication source.

4a2.2.5 SUBCRITICAL ASSEMBLY SUPPORT STRUCTURE

The TSV maintains the location and shape of the aqueous target solution during irradiation. The SASS positions the TSV relative to the neutron driver, as well as the relative location of the neutron multiplier, subcritical multiplication source, and neutron flux monitors as shown in Figure 4a2.1-2. Any deviations from process limits for criticality, temperature, pressure, and others result in a safety-related gravity discharge of the TSV target solution into a criticality-safe by geometry TSV dump tank (Figure 4a2.2-1), which is below the TSV in the LWPS. The SASS contains the TSV and supports TSV dump lines, TSV overflow lines, TOGS piping, and associated instrumentation.

The SASS provides two other functions. It is connected to the primary coolant loop, forcing coolant to pass [Proprietary Information] along the TSV inner and outer walls. It also forms an additional fission product boundary in the event of a breach in the TSV.

The SASS and supported components are designed to withstand the design basis loads, including thermal, seismic, and hydrodynamic loads imposed by the light water pool during a seismic event. In addition, the SASS and components are designed to withstand normal operating loads imposed by the primary coolant loop and target solution, including hydraulic and thermal stresses. The entire assembly is submerged in the light water pool. In the event of a loss of the PCLS in a design basis event, the LWPS provides passive safety-related cooling of the TSV and target solution.

The SASS is supported from the floor of the light water pool via seismic anchorages that establish the alignment of the SASS relative to the neutron driver. The SASS provides vertical and lateral support for the subcritical assembly components. Nozzle loads imposed on the SASS (target solution dump, PCLS, LWPS, and TOGS lines) are accommodated by the design of the SASS ensuring that displacements resulting from these lateral loads do not result in stresses that exceed design allowables, or affect the fit, function, or neutronics of the overall system.

The SASS is operated at pressures near atmospheric (a slight pressure differential is required across the SASS to provide cooling water flow). The SASS is designed for an internal pressure of 100 pounds per square inch (psi) (689 kPa) to accommodate forces resulting from a hydrogen deflagration event followed by a failure of the TSV integrity.

The PCLS is attached to the SASS upper and lower plenums. The PCLS is used to remove up to approximately [Proprietary Information] of heat from the TSV during full power operation.

Near TSV elevation, the SASS is exposed to neutron fluxes of approximately [Proprietary Information]. The SASS does not normally contact the target solution. In the event of a breach in the TSV, the SASS provides a boundary between the target solution and the light water pool. This boundary is designed to provide for safe holdup of the target solution until it is recovered and removed from the SASS and PCLS.

Materials of construction for the SASS and associated fixtures used to locate the neutron flux monitors are 316 stainless steel. The properties and behavior of 316 stainless steel in the expected neutron fluences are well-documented, and will be accounted for in the design of the SASS, ensuring its safe reliable operation over its 30 year design life.

Neutron flux monitors are supported using brackets attached to the outer walls of the SASS. These brackets serve to locate the flux monitors in a fixed location relative to the TSV, ensuring flux profiles are measured consistently. Three flux monitors are positioned around the SASS at nominally 120 degree intervals.

The design of the TSV, TSV dump tank, target solution hold tank, and associated piping and valves ensures that target solution is never present in excess of one batch volume in these three vessels (see Figure 4a2.8-1). The target solution hold tank (1-TSPS-03T) is the same size as the liquid fill volume of the TSV. To fill the TSV, the target solution hold tank is isolated from target solution feeds to the tank. Once isolated from a source of additional uranyl sulfate, and with TSV dump tank (1-SCAS-01T) verified empty, the contents from the target solution hold tank are pumped in discrete steps into the TSV. Note that the elevation of the criticality-safe target solution hold tank prevents the gravity transfer of target solution to the TSV.

During irradiation, the target solution hold tank and the TSV dump tank remain empty, and no additional solution is added to the TSV. After the irradiation cycle is completed, or if tripped, the TSV gravity drains to the TSV dump tank. The TSV dump tank provides criticality-safe geometry for storage of the target solution. The light water pool provides a mechanism for removal of decay heat from the target solution after a transfer from the TSV.

4a2.2.5.1 Technical Specifications

There are no potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the SASS.

4a2.2.6 NEUTRON MULTIPLIER

The neutron multiplier is an annulus (approximately [Proprietary Information] [Security-Related Information] tall) of aluminum-clad [Proprietary Information] that serves to moderate and multiply the fast neutrons coming from the fusion reactions initiated by the neutron driver. The multiplier consists of [Security-Related Information][Proprietary Information], clad in aluminum, nominally 0.25 in. (0.64 cm) thick. The design life of the neutron multiplier is 30 years, but it is designed to allow remote replacement should physical damage occur to it, or a distortion that interferes with the insertion of the neutron driver occur over time. A more comprehensive description of the manufacturing techniques and final dimensions for the neutron multiplier will be available after detailed design.

The fast fusion neutrons that collide with the [Proprietary Information] can cause several high energy reactions to occur. The most common reactions in the multiplier include [Proprietary Information]. The resulting spectrum of [Proprietary Information] neutrons then enter the TSV.

The aluminum cladding contains [Proprietary Information]. In the event of a cladding failure, there are no consequences that would affect the safe operation and shutdown of the irradiation system. There is potential that the [Proprietary Information], and could lead to contamination in the light water pool. Sampling the LWPS detects such a breach via the increased radioactive contamination present in the water.

Radiation damage and burn-up are not expected to impact operation of the multiplier for the entire lifetime of the plant. The maximum fluence of the multiplier over a 30 year period of continuous operation is calculated to be less than [Proprietary Information] neutrons per square centimeter (n/cm^2). [Proprietary Information]

[Proprietary Information] Heat deposited in the multiplier is removed by the LWPS. The temperature profiles through the multiplier will be determined during detailed design, with appropriate design features included to address the limited thermal expansion and contraction of the neutron multiplier expected to occur.

4a2.2.6.1 Technical Specifications

There are no potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the neutron multiplier.

Table 4a2.2-1 Target Solution Chemical and Physical Properties

Property	Nominal Values
Chemical Composition	[Proprietary Information]
Temperature	68 to 176°F (20 to 80°C)
Density	[Proprietary Information]
Heat Capacity	[Proprietary Information]
Average Power Density	Up to [Proprietary Information]
Mass	[Proprietary Information]
Volume	[Proprietary Information]
Uranium Content	[Proprietary Information] uranyl sulfate ([Proprietary Information] uranium)
Fissile Uranium Density During Irradiation Conditions	[Proprietary Information]
pH	[Proprietary Information]
Initial Enrichment	19.75 ± 0.2% U-235
Precipitates	None expected
Uranium Solubility	13.8 lb/gal (1650 g/L) at 135°F (57.2°C)

Table 4a2.2-2 Irradiated Target Solution Activity for Select Radionuclides [Proprietary Information] Following Shutdown[(a)] (Sheet 1 of 2)

Radionuclide	Activity (Curies)
Kr-85	[Proprietary Information]
Kr-85m	[Proprietary Information]
Kr-87	[Proprietary Information]
Kr-88	[Proprietary Information]
Rb-86	[Proprietary Information]
Sr-89	[Proprietary Information]
Sr-90	[Proprietary Information]
Sr-91	[Proprietary Information]
Sr-92	[Proprietary Information]
Y-90	[Proprietary Information]
Y-91	[Proprietary Information]
Y-92	[Proprietary Information]
Y-93	[Proprietary Information]
Zr-95	[Proprietary Information]
Zr-97	[Proprietary Information]
Nb-95	[Proprietary Information]
Mo-99	[Proprietary Information]
Tc-99m	[Proprietary Information]
Ru-103	[Proprietary Information]
Ru-105	[Proprietary Information]
Ru-106	[Proprietary Information]
Rh-105	[Proprietary Information]
Sb-127	[Proprietary Information]
Sb-129	[Proprietary Information]
Te-127	[Proprietary Information]
Te-127m	[Proprietary Information]
Te-129	[Proprietary Information]
Te-129m	[Proprietary Information]
Te-131m	[Proprietary Information]
Te-132	[Proprietary Information]
I-131	[Proprietary Information]
I-132	[Proprietary Information]
I-133	[Proprietary Information]
I-134	[Proprietary Information]
I-135	[Proprietary Information]
Xe-131m	[Proprietary Information]
Xe-133	[Proprietary Information]
Xe-133m	[Proprietary Information]

Table 4a2.2-2 Irradiated Target Solution Activity for Select Radionuclides [Proprietary Information] Following Shutdown[(a)] (Sheet 2 of 2)

Radionuclide	Activity (Curies)
Xe-135	[Proprietary Information]
Xe-135m	[Proprietary Information]
Xe-138	[Proprietary Information]
Cs-134	[Proprietary Information]
Cs-136	[Proprietary Information]
Cs-137	[Proprietary Information]
Ba-139	[Proprietary Information]
Ba-140	[Proprietary Information]
La-140	[Proprietary Information]
La-141	[Proprietary Information]
La-142	[Proprietary Information]
Ce-141	[Proprietary Information]
Ce-143	[Proprietary Information]
Ce-144	[Proprietary Information]
Pr-143	[Proprietary Information]
Nd-147	[Proprietary Information]
Np-239	[Proprietary Information]
Pu-238	[Proprietary Information]
Pu-239	[Proprietary Information]
Pu-240	[Proprietary Information]
Pu-241	[Proprietary Information]
Am-241	[Proprietary Information]
Cm-242	[Proprietary Information]
Cm-244	[Proprietary Information]
Rb-88	[Proprietary Information]
Y-91m	[Proprietary Information]
Nb-97m	[Proprietary Information]
Nb-97	[Proprietary Information]
Rh-103m	[Proprietary Information]
Rh-105m	[Proprietary Information]
Rh-106	[Proprietary Information]
Ba-136m	[Proprietary Information]
Ba-137m	[Proprietary Information]
Pr-144	[Proprietary Information]
Pr-144m	[Proprietary Information]

a) Table is for target solution that has been irradiated for [Proprietary Information] cycles

Table 4a2.2-3 Target Solution Operating Limits

Operating Limit	Range/Value
Temperature	50 to 194°F (10 to 90°C)
Uranium Concentration	[0.95 to 1.3 lbU/gal (114 to 154 gU/L)][Proprietary Information]
pH	[Proprietary Information]

4a2.3 NEUTRON DRIVER

Within the SHINE IF, there are eight NDAS. The NDAS is a central part of the SHINE production process. The NDAS is the source of neutrons used to generate the neutron fluxes required to create medical isotopes in the TSV which holds the target solution. In the IF, the eight NDASs operate independent of each other.

The NDAS primary interfaces to the other IF structures, systems, and components (SSCs) are:

- The SCAS surrounds the NDAS target chamber. The neutrons produced by the NDAS in the target chamber irradiate the solution held by the SCAS.
- The TPS processes mixed tritium and deuterium gas from the NDAS target chamber and returns purified tritium to the NDAS.
- The IF houses the NDAS. The IF provides the physical interfaces for the NDAS such as: cooling water, power, ventilation, and shielding.
- The instrument & control systems provide the functional interface to the overall production facility process for the NDAS. The TPCS serves as the functional interface for normal operational steps.
- When necessary the TRPS can actuate its trip mechanisms, which de-energizes the NDAS.

The NDAS produces neutrons by colliding a deuterium (D) ion beam with a tritium (T) gas. D-T reactions have a lower energy threshold than any other neutron-producing, accelerator-driven reaction. D-T reactions also have a large cross section and therefore yield neutrons efficiently. Table 4a2.3-1 lists nominal design parameters for the NDAS.

The NDAS is part of the primary tritium boundary. In order to mitigate tritium leaks, the NDAS operates below atmospheric pressure. Additionally, the components and assembly methods used are designed for vacuum service which inherently requires low leak rates. The tritium supply and return are designed to have a flush, seal, and break mechanism to make the tritium release as low as possible for any procedure requiring the tritium lines to be disconnected.

The anticipated lifetime of an NDAS is [Proprietary Information]. This lifetime is driven by activation of NDAS components and the required ability to perform maintenance of the NDAS subcomponents. Neutron fluence on all portions of the NDAS except the target chamber assembly is predicted to be less than [Proprietary Information] n/cm^2. Because of this relatively low fluence, most materials will not have radiation damage concerns. However, appropriate plastics and low activation materials have been chosen. Complex electronic components, such as control microprocessors, operate outside of the IU cell. The NDAS is designed to allow for in-cell maintenance after both TSV solution and tritium gas have been drained and sufficient decay time has passed for any activated components. It has also been designed to be lifted via crane from the IU cell for installation and removal for major maintenance and replacement.

The bulk of the NDAS resides above the subcritical assembly and will, at times, be suspended via crane above it for maintenance activities. In order to mitigate risk from all, or portions of, the NDAS falling onto the subcritical assembly, an IU cell floor plate separates the bulk of the two systems.

The NDAS is a typical accelerator system with the exception of the gas target. The subsystems and components that make up the NDAS are described below and include (Figure 4a2.3-1): HVPS, control system, ion source, accelerator, focus element, differential pumping, and the target chamber. Additionally, material selections for major subsystems in each of these components are listed in Table 4a2.3-2.

4a2.3.1 HIGH VOLTAGE POWER SUPPLY

The HVPS is a 300 kilovolt (kV) power supply specifically designed to provide voltage to the accelerator. The HVPS has an internal control system made up of electronics and firmware that interfaces with the NDAS control system. The HVPS has inherent and active design protective features such as overload detection, low stored energy, and fast shutdown. The HVPS shuts down on an overvoltage of greater than 320 kV. This is a protective function and is not a safety limit or function.

4a2.3.2 CONTROL SYSTEM

The NDAS control system is the hardware and software required to energize, change, and monitor the different parts of the NDAS. Some examples are:

- Programmable power supplies that drive each turbo pump in the differential pumping subsystem.
- Electronics controlling the radio frequency required for the microwave ion source.
- Control electronics (microprocessors, field programmable gate arrays [FPGAs]) and software for setting and reading various electrical signals, for interfacing with the instrumentation and control, and for an operator (engineer, service) to run, diagnose, and test the NDAS.

The TPCS serves as the functional interface to the overall facility process for the NDAS. The NDAS control system receives commands and sends data to the TPCS during operational steps allowing the TPCS to control and monitor the behavior and status of the NDAS.

4a2.3.3 ION SOURCE

The ion source is comprised of an electron cyclotron resonance (ECR) ion source and an ion extractor. The ECR source operates by focusing 2.45 gigahertz (GHz) microwaves into a plasma chamber with axially varying magnetic fields inside it. A small amount ([Proprietary Information]) of deuterium gas is injected into the plasma chamber where it is ionized by the combination of microwave and magnetic fields. The ion extractor, which is biased negatively with respect to the plasma chamber, pulls deuterium ions from the ion source and injects them into the accelerator.

4a2.3.4 ACCELERATOR

The accelerator (Figure 4a2.3-2) is a 300 kV column that operates by converting the external voltage from the HVPS to a voltage gradient through a resistor network and transferring the voltage gradient to a series of internal lenses. This is an efficient and proven means of accelerating the ions generated by the ion source to the target chamber. Because the accelerator operates at high voltage, there exists the potential for electromagnetic interference

(EMI) generation. NDAS components exposed directly to this EMI are shielded or otherwise electrically isolated to avoid damage or interference. Other components in the vicinity of the NDAS are also shielded to ensure they are protected from EMI-related interference.

4a2.3.5 FOCUS ELEMENT

As the ion beam is accelerated, its initial divergence and any space charge forces cause it to grow in radius. The focus element focuses the expanding ion beam through the differential pumping stages into the aperture of the target chamber. This focusing force is generated by a solenoid electromagnet that produces an on-axis field of [Proprietary Information]. If the focus element failed to operate, the ion beam size change would cause the neutron yield to drop. If the focus element failed and caused the ion-beam to diverge or misalign from the target chamber, the ion-beam divergence would cause the neutron yield to drop.

4a2.3.6 DIFFERENTIAL PUMPING

Differential pumping maintains the pressure differential between the accelerator [Proprietary Information] and the tritium target chamber (about [Proprietary Information]. This large pressure differential is accomplished using stages of turbo-molecular pumps and roots blowers in a differential pumping configuration (Figure 4a2.3-3). In differential pumping configuration each stage's pump exhaust returns to the adjacent higher pressure pump stage. Each stage, in this configuration, is separated by a small [Proprietary Information] aperture that allows the ion beam to pass but minimizes the gas flow from the adjacent higher pressure stage.

4a2.3.7 TARGET CHAMBER

The aluminum target chamber is filled with low-pressure ([Proprietary Information]) tritium gas. When the accelerated deuterium ions interact with the tritium gas in the target chamber, many atomic and nuclear interactions occur in the target chamber. One of the interactions is the fusion nuclear reaction, which generates 14 MeV neutrons. The estimated neutron yield of the NDAS from the D-T fusion reaction is [Proprietary Information] neutrons per second (n/s) for a target chamber consisting of only tritium gas. The number of neutrons generated per second (n/s) is correlated to subcritical assembly fission power.

Atomic reactions between the beam and the gas cause the beam to be stopped before reaching the bottom of the target chamber. These reactions generate up to [Proprietary Information] (300 kV x [Proprietary Information]) of heat in the target chamber that must be removed by water cooling. A very small percentage of the beam may also be scattered into the target chamber walls, and the water cooling removes this heat as well. Because the beam fully stops in the gas during normal operation, inadvertent increases in target chamber pressure have no effect on neutron yield. Inadvertent decreases in target chamber pressure results in lower neutron yield and may cause some fraction of the beam power to strike the bottom of the target chamber. Therefore, this surface is water-cooled during operation.

The neutron yield is controlled using tritium gas purity in the target chamber. As the accelerated deuterium ions interact with the gas in the target chamber, the tritium becomes mixed with deuterium. As the gas becomes mixed with deuterium, the production of neutrons becomes less efficient and, therefore, the n/s generated naturally decreases. During NDAS operation, the TPS processes the mixed tritium and deuterium gas from the NDAS target chamber and returns

purified tritium gas. The flow rate of the purified tritium gas into the target chamber controls the percent of tritium in the target chamber, which also controls the neutron output.

The target chamber is surrounded by the subcritical assembly and is exposed to high neutron flux levels. The materials and design for the target chamber were selected to handle corrosion and damage due to the high neutron flux levels, compatibility with the cooling water, and compatibility with the subcritical assembly materials.

4a2.3.8 PROCESS CONTROL REQUIREMENTS

The TPCS and TRPS have control interfaces with the NDAS. In the event of an abnormal condition in the TSV, the HVPS shuts down via safety-related trip circuitry that isolates power.

4a2.3.9 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the target solution are provided in Chapter 14.

Table 4a2.3-1 Neutron Driver Nominal Design Parameters

Condition	Operating Range
Neutron Output	[Proprietary Information] +/- 3% of set neutron output for a given tritium target purity.
Neutron Driver Trip/Fault	<10 seconds; NDAS automatically recovers and restarts for any NDAS interrupt lasting less than 10 seconds.
Neutron Driver Trip/Fault	>10 seconds; NDAS automatically shuts down for any NDAS interrupt lasting more than 10 seconds.
HVPS Overvoltage	>320 kV; The NDAS turns off on a HVPS over voltage of greater than 320 kV
Shutdown Command	<1 second; NDAS stops neutron production in less than one second in response to a “shutdown command”.

Table 4a2.3-2 Materials Used in NDAS Subsystems

Component	Primary Materials
HVPS	
- Tank	Steel
- Cabinet	Steel
- Transformer	Acrylic, PVC, copper
- Internal components	Aluminum, Steel, Acrylic
- Dielectric	Sulfur Hexafluoride
Ion Source	
- Plasma Chamber	[Proprietary Information]
- Extraction Lens	[Proprietary Information]
- Waveguide	[Proprietary Information]
- Solenoid Magnets	[Proprietary Information]
Accelerator	
- Lens Elements	[Proprietary Information]
- Internal Structure	[Proprietary Information]
- Insulators	[Proprietary Information]
- Vacuum Components	[Proprietary Information]
Focus Element	
- Solenoid Magnet	Copper, Steel
- Structure	Steel, Aluminum
- Vacuum Components	Aluminum, Stainless Steel
Differential Pumping	
- Pumps	Aluminum, Stainless Steel, Steel
- Vacuum Components	Aluminum, Stainless Steel
- Apertures	[Proprietary Information]
Target Chamber	
- Vacuum Components	[Proprietary Information]
- Apertures	[Proprietary Information]

4a2.4 TARGET SOLUTION VESSEL AND LIGHT WATER POOL

This section presents information about the TSV and light water pool necessary to demonstrate their integrity. The TSV is part of the PSB, which consists of the TSV, TSV dump tank, and TOGS.

4a2.4.1 TARGET SOLUTION VESSEL

The following subsections provide an overview of the TSV design characteristics, key functions, interfaces, and environment to which the TSV is exposed during operation (see additional details including figures in Section 4a2.1 and Section 4a2.2).

4a2.4.1.1 Design Considerations

The TSV is one of the three main components of the subcritical assembly (along with the neutron multiplier and SASS [see Figure 4a2.1-2]). The TSV is designed and fabricated following the intent of the ASME Boiler and Pressure Vessel Code (BPVC), Section III (ASME, 2011).

The TSV provides structural integrity for maintaining the correct target geometry during operation. The TSV is also capable of withstanding the pressure excursions encountered during a credible unplanned deflagration of the radiolytic gases.

The subcritical assembly, of which the TSV is an internal component, is designed to allow access to the TSV for component replacement. Remote-disconnect fittings are incorporated into the top and bottom sections of the subcritical assembly to allow entry of special tools to be used in replacing the TSV components.

TSV dump valves are provided to drain the TSV, as part of the process prior to transferring the target solution downstream for processing and as a safety-related feature utilized as part of a planned response in the event of an IU upset condition. The valves are located in redundant flow paths and fail to a safe (open) position. Both the TRPS and TPCS can open both dump valves.

A TSV dump tank provides for the gravity transfer from the TSV to a passively-cooled, criticality safe storage geometry for the target solution in the event of an upset in the IU operating conditions or an accident scenario.

During normal operation, there is approximately 3.9 to 11.8 in. (10.0 to 30.0 cm) of gas space above the uranium solution in the TSV. The gas space is connected to a radiolytic gas recombination system that continuously sweeps the cover in order to reduce the potential for the development of explosive concentrations of radiolytic gases by limiting the hydrogen gas (H_2) concentration to less than the LFL (see Section 4a2.8 for a design description of the TOGS).

The operating temperature in the TSV is expected to remain below 176°F (80°C). The TSV gas space and off-gas system are also held at a slight negative pressure (a few inches of water column). Operating specification limits of the target solution are discussed in Section 4a2.2.

There is no mechanical mixing in the TSV. The target solution is mixed using natural convection during irradiation due to internally-produced fission heat and radiolytic gas bubble formation.

Work performed by ANL has shown that uranium [Proprietary Information]. No significant pH changes are expected during irradiation due to the stability of sulfuric acid under irradiation.

The TSV has several connections to the TSV protection system and the control room. The subcritical assembly instrumentation provides information to the TSV protection system (such as target solution temperature, TSV pressure, and solution height), which are used to de-energize (open) the TSV dump valves when certain parameters are exceeded.

Pressure in the TSV is monitored to assess the status of the recombination system, and to indicate possible pressure fluctuations due to power oscillations.

4a2.4.1.2 Design and Dimensions

The TSV has an internal height of approximately [Security-Related Information] and a target solution thickness of about [Security-Related Information]. The TSV is filled to a height of approximately [Security-Related Information] at cold shutdown conditions.

4a2.4.1.3 Design Description of Materials and Supporting Structures

The TSV is constructed of zircaloy-4, an alloy of zirconium that offers exceptional corrosion resistance under irradiation and offers a very low neutron absorption cross section. Zircaloy-4 is widely used throughout the nuclear industry where corrosion resistance and neutron economy are important.

The TSV is supported in the light water pool by the SASS. The neutron driver is mounted directly overhead, extending down into the tritium chamber which is located in the center of the subcritical assembly. Between the tritium chamber and the SASS inner wall is the neutron multiplier as shown in Figure 4a2.1-2. Subsection 4a2.2.5 provides a detailed description of the SASS. The SASS is designed to withstand the environment of the irradiation process and the design basis accidents:

- The SASS and supported components are designed to withstand the design basis loads, including thermal, seismic, and hydrodynamic loads imposed by the light water pool during a seismic event. In addition, the SASS and TSV are designed to withstand credible pressurized loadings due to hydrogen detonation or deflagration in the event of a loss of hydrogen recombination capacity without failure of the integrity of the PSB.
- The materials of construction for the SASS and associated fixtures used to locate the neutron flux monitors is 316 stainless steel. Properties and behavior of this material under neutron exposure and in contact with deionized water have been extensively analyzed.
- In the event of a breach of the TSV, the SASS and PCLS provide a boundary between the target solution and the light water pool.

4a2.4.1.4 Location of Penetrations

The pool surface is nominally 6.0 ft. (1.8 m) above the top of the TSV.

The TSV is cooled through both side walls, [Proprietary Information] the vessel. [Proprietary Information]. [Proprietary Information] the external walls of the TSV are convectively cooled via

the PCLS. Forced convection is utilized during normal operation to maximize heat removal from the TSV and maintain the target solution below the operating condition limits specified in Subsection 4a2.2.1. Upon loss of forced convection, the neutron driver HVPS is de-energized by the TPCS and TRPS, essentially terminating the fission process. Adequate removal of decay heat from the TSV is accomplished after loss of forced convection through simple natural convection and conduction to the light water pool.

In addition to the [Proprietary Information], the TSV contains two approximately 1.6 in. (4.0 cm) inner diameter drain ports for interconnection to the TSV dump tank (1-SCAS-01T). The TSV also has two connections to solution overflow lines and two TOGS connections. Solution addition is accomplished through a small diameter fill line connected to the TSV drain lines upstream of the TSV dump valves. The small size of the fill port limits the maximum solution fill rate for criticality-safety purposes.

The [Proprietary Information]. The drain ports are welded to the bottom of the TSV. The overflow lines are welded to the bottom of the TSV where they penetrate the vessel. Internal to the TSV, the overflow lines consist of a standpipe approximately [Security-Related Information][Proprietary Information] tall, which prevents inadvertent overfilling of the TSV.

The PCLS is an active coolant system designed to remove excess heat from the TSV during IU operation. The PCLS includes filters to remove activation products, plate-type heat exchangers for rejection of heat to the secondary coolant system, and deionizer units. The system also functions as a fission product boundary between the TSV and the light water pool.

4a2.4.1.5 Chemical Interactions and Neutron Damage

The TSV is exposed to total neutron fluxes in the range of [Proprietary Information] during irradiation, and in direct contact with the uranyl sulfate target solution during operation and credible operational accidents. However, based on the use of zircaloy-4 for the TSV construction, as noted in Subsection 4a2.4.1.3:

- No significant chemical damage to the inner surface of the TSV is expected; and a coating on that surface is not required.
- Evaluation of the effects of neutron damage will be provided in the FSAR.

The PCLS and SASS provide an additional barrier to fission product release in the event of a TSV breach, as shown in Figure 4a2.1-2. The closed loop design of the PCLS prevents commingling of the PCLS coolant with the water in the light water pool. The PCLS is operated at a higher pressure than the TSV, and normally prevents leakage of the target solution to the PCLS in the event of a breach. In the event of pressurization of the PSB due to an accident condition (such as hydrogen deflagration), leakage from the TSV would be contained within the PCLS. The PCLS is designed to withstand credible pressurized loadings due to a failure of the TSV during an accident. A detailed description of the PCLS is provided in Section 5a2.2.

Leakage of very small quantities of target solution is detected by periodic sampling of PCLS and LWPS water.

Significant leakage of PSB components outside of the SASS would result in contamination within the light water pool and atmosphere above the pool. Contamination would be contained by the confinement barrier of the IU cell.

In the event a significant breach of the TSV, which would be indicated by significantly lower flux levels, an uncharacteristic drop in the PCLS head tank level, or detection of contamination in the PCLS, the target solution is transferred to the criticality safe TSV dump tank (1-SCAS-01T) to minimize the activity lost to the systems surrounding the TSV.

A material surveillance and inspection program for the TSV and other PSB components will be described in the FSAR. This surveillance and inspection program will ensure the integrity of the PSB components is not degraded below acceptable limits due to radiation damage, chemical damage, erosion, pressure pulses, or other deterioration during the 30-year lifetime.

4a2.4.2 LIGHT WATER POOL

4a2.4.2.1 Design of Light Water Pool

The light water pool is a concrete structure, lined with stainless steel, which is designed to withstand design basis events without the loss of liner integrity that could compromise the water retention capability of the liner. The pool is approximately [Proprietary Information] deep, which equates to approximately [Proprietary Information]. The IU components displace approximately 2000 gal. (7571 L); therefore the water volume is [Proprietary Information] of water.

If a breach that leaks target solution into the pool were to occur, the stainless steel liner of the pool is designed to withstand the chemical environment of the target solution. The UNCS (see Subsection 4b.4.1) may then be used to process the contents of the light water pool by separating out the uranium and passing the contaminated water on for downstream processing.

The minimum acceptable water level in the light water pool is the minimum level that is assumed for safety analysis accident scenarios. This level will be specified in the FSAR. The minimum acceptable water level in the pool allows for adequate heat removal from the target solution during normal operation and after the loss of forced cooling. Piping penetrations through the light water pool liner are either above the minimum acceptable water level in the pool or a specific evaluation is performed to determine the potential for loss of pool water through the penetration. Piping penetrations into the light water pool with the potential for siphoning below the minimum acceptable water level contain anti-siphon devices or other means to prevent inadvertent loss of pool water. Should a leak in the liner develop, a leak chase system is incorporated into the design to provide indication that a leak is occurring, to capture the lost liquid to prevent release to the environs, and to ensure that the leak is less than the allowable leakage rate for the liner.

The primary functions of the light water pool are to: 1) remove heat, and 2) provide radiological shielding. These functions are described in the following subsections along with the related instrumentation.

4a2.4.2.2 Heat Removal

During normal operation, the light water pool is used to remove heat from the neutron driver target chamber and neutron multiplier via forced convection, as well as a small portion of fission power from the TSV due to gamma shielding and neutron slowing-down energy in the pool. The LWPS cooling loop circulates the pool water through a heat exchanger to maintain the normal operating temperature of the light water pool. See Sections 4a2.1 and 5a2.2, for more information. Approximately [Proprietary Information] of heat is removed by the LWPS cooling loop during TSV operation at the licensed power limit of [Proprietary Information].

In the event of a failure of the PCLS forced cooling function, the thermal mass provided by the light water pool provides sufficient decay heat removal capacity to provide cooling for the TSV when the neutron driver is shutdown. The temperature rise in the light water pool is 11.8°F after a decay heat period of 90 days (see Figure 4a2.4-1). The light water pool is also used to remove up to [Proprietary Information] of decay heat from the uranyl sulfate solution in the dump tank after the target solution has been irradiated, but prior to transfer to the Mo extraction cell (see Section 4a2.5). Subsection 13a2.1.3 discusses loss of cooling events and concludes that the fission product boundaries are not challenged. The integrity of the light water pool is not challenged under any postulated accident conditions.

4a2.4.2.3 Radiological Shielding

The light water pool surrounding the TSV provides significant neutron and gamma shielding, thereby reducing the shielding requirements of the IU biological shield and reducing neutron activation of equipment located above the light water pool.

As noted in Section 4a2.5, neither the neutron or gamma flux levels are high enough to threaten the integrity of the concrete structure surrounding the light water pool. Significant degradation of the shielding performance of the water only occurs if void volume results from boiling. Based on the heat generation rates for the various operational conditions, the heat capacity of the pool water and the heat transfer rate to adjacent materials, boiling of the pool water is not credible. Pool water temperature is monitored to ensure it is within allowable limits (see Subsection 4a2.4.2.4 below). The relatively minor loss of shielding resulting from a decrease in the pool water density as the water temperature increases will be accounted for.

The design requirements of the light water pool and associated piping and valving prohibit configurations that could accumulate significant volumes of radiolytic gases within the bulk volume of pool water, which could result in voids that do not provide equivalent shielding.

The principal objective of the light water pool shield design (together with the concrete biological shield) is to ensure the projected radiation dose rates and accumulated doses in occupied areas do not exceed the limits of 10 CFR 20 and the guidelines of the facility's as low as reasonably achievable (ALARA) program discussed in Subsection 11.1.3.

4a2.4.2.4 Instrumentation

The light water pool includes instrumentation to monitor temperature and liquid level along with a leak chase system to provide indication that a leak is occurring.

Temperature instrumentation monitors temperature within the light water pool. If the LWPS cooling loop is not removing heat adequately, the temperature in the light water pool rises, and when the temperature exceeds the high-level setpoint, an alarm alerts operators to take remedial action. This includes shutting down the IU, investigating the cause of the cooling loss from the LWPS cooling loop, and taking corrective action. The normal operating temperature range of the light water pool is 68 to 75°F (20 to 24°C).

Liquid level instrumentation monitors volume changes in the light water pool. If the liquid level of the pool decreases to the low liquid level, there may be a leak in the LWPS cooling loop. The normal operating liquid level range of the light water pool will be determined during final design.

Repair procedures are conducted in accordance with the radiation protection, monitoring, and response guidelines of Chapter 11. Further design of the light water pool instrumentation is performed during final design.

4a2.4.3 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the light water pool are provided in Chapter 14.

4a2.5 IRRADIATION FACILITY BIOLOGICAL SHIELD

4a2.5.1 INTRODUCTION

The IU biological shield provides a barrier to protect SHINE facility personnel, members of the public, and various components and equipment of the SHINE facility by reducing radiation exposure. It is comprised of the concrete vault around the IU components, TOGS shielded cell, and the biological shielding for the primary cooling system loops.

4a2.5.2 BIOLOGICAL SHIELD DESIGN BASIS

4a2.5.2.1 Materials

The design bases for the materials to be included in the biological shield design are:

a. The dose reduction by the biological shielding is sufficient to meet the radiation exposure goals defined in Chapter 11.

b. All materials used for biological shielding meet or exceed the requirements of ANSI/ ANS-6.4.2-2006, Specification for Radiation Shielding Materials.

c. The design and construction of the concrete portions of the biological shield conform to NRC Regulatory Guide 1.69, Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants.

4a2.5.2.2 Geometry and Configuration

The side wall of the IU cell biological shield is approximately [Security-Related Information] of standard density (0.08 pounds per cubic inch [lb/in^3]) (2.3 grams per cubic centimeter [g/cm^3]) concrete, resulting in dose rates on the external surface of the shielding of less than 1.0 millirem per hour (mrem/hr). See Figure 4a2.2-1 for a general depiction of the IU cell biological shielding.

4a2.5.2.3 Loss of Shield Integrity

The biological shield walls and supporting structures are designed and constructed to remain intact during normal operations as well as during and following design basis accidents. A loss of shield integrity is not credible given the seismic design and robust nature of the IU and TOGS cells.

4a2.5.2.4 Unrestricted Environment

Design and construction of the shield is such that neutron irradiation and activation of ground water or soils surrounding the biological shield in the unrestricted environment is expected to be insignificant. The neutron flux to soils surrounding the biological shield is estimated to be less than 100 n/cm^2-s. Since the neutron flux to the soils surrounding the biological shield is low, the neutron activation to these soils is negligible.

4a2.5.3 SHIELD MATERIALS

The IU cell uses two distinct materials in different configurations to assemble the biological shield and meet the radiation exposure goals defined in Chapter 11. The materials that make up the biological shielding use an engineered concrete mix with reinforcing bars. Preliminary analysis did not credit the reinforcing bars.

4a2.5.3.1 Shielding Calculations

Calculations are performed with the software package named MCNP (Monte-Carlo N-Particle Transport Code). MCNP is developed and validated by Los Alamos National Laboratory (LANL) and distributed by the Radiation Safety Information Computational Center (RSICC) at ORNL. MCNP uses a Monte-Carlo based particle (neutrons and photons) transport method to generate a set of particle tracks through a model of the facility geometry (LANL, 2004). The Monte Carlo method generates a statistical set of results for individual particles transported through the geometry. Enough particles are simulated to obtain statistically-significant results. The MCNP calculation methodology does not require use of shielding coefficients. The shielding coefficients will be listed in the FSAR.

4a2.5.3.2 Radiation Damage

4a2.5.3.2.1 Concrete

According to the American National Standard Institute (ANSI) ANSI/ANS-6.4-2006 (ANSI/ANS, 2006a), Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, nuclear heating in concrete can be neglected if the incident energy fluxes are less than 10^{10} MeV per square centimeter per second (MeV/cm^2-sec).

During irradiation, significant neutron and gamma fluxes are created by the irradiation process in the subcritical assembly. The light water pool serves to significantly reduce the magnitude of the fluxes that reach the irradiation facility biological shield. There is also a neutron population born above the light water pool in the neutron driver estimated at less than 1 percent of the neutron driver source strength. This neutron population is not attenuated by a water shield before striking the biological shield. The cumulative effects of the neutron and gamma fluxes in the IF will be analyzed using MCNP software in final design. The results of this analysis will be provided in the FSAR.

With regards to degradation, ANSI/ANS-6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants states that compressive strength and modulus of elasticity of concrete are degraded if the concrete is exposed to a neutron dose greater than 10^{19} n/cm^2 or to an integrated dose of gamma radiation exceeding 10^{10} rads (ANSI/ANS, 2006a). Using MCNP, analysis of the maximum neutron and gamma doses to the concrete will be performed in final design. The results of this analysis will be provided in the FSAR.

Analysis performed will include activation of structural components within the biological shield, and the results of this analysis will be included in the FSAR.

4a2.5.3.3 Radiation Streaming

The biological shield requires a number of penetrations, inserts, and other features where the bulk shielding materials are reduced in thickness, or where the materials used in the penetration are less dense than the surrounding bulk material. Each such penetration is designed with well-demonstrated techniques of non-linear paths, supplemental shielding, location in areas of low incident radiation, and other methods to reduce streaming and leakage to ensure 10 CFR 20 limits are met.

4a2.5.4 ANALYSIS

Analysis is performed to:

- Give detailed results of both neutron and gamma-ray dose rates at locations that could be occupied as well as to the unrestricted environment.
- Include shield penetrations and voids, such as beamports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.
- Determine extent of radiation effect on shielding materials (i.e. heating and activation).

4a2.5.4.1 Methodology

4a2.5.4.1.1 Concrete Radiation Shielding Minimum Thickness – Radiation Shielding Requirements

The minimum thickness of concrete radiation shields, based on radiation shielding requirements, is determined using the following approach:

a. Use ANSI/ANS-6.4-2006, Chapters 6, 7, and 8, as an overview of the historic calculation methodology for concrete radiation shields (ANSI/ANS, 2006a).
b. Use Monte Carlo techniques for radiation shielding calculations.
c. Use the latest version of the software for radiation shielding calculations (i.e., MCNP5). The concrete composition input parameters for the MCNP5 calculations correspond to the concrete used for the radiation shields.

4a2.5.4.1.2 Concrete Radiation Shielding Minimum Thickness – Structural Requirements and Other Structural Dimensions and Reinforcement Requirements

The minimum thickness of concrete radiation shields, based on structural requirements, and other structural dimensions and reinforcement requirements is determined in accordance with the provisions of ACI 349-06 and ACI 349.1R-07 (ACI, 2007a; ACI, 2007b) for applicable normal loads, severe and extreme environmental loads, and abnormal loads, as defined in Section 9.1 of ACI 349-06.

4a2.5.4.1.3 Concrete Radiation Shielding – Final Minimum Thickness

The final minimum thickness of the concrete shield structure will be based on radiation shielding requirements and will not be driven by the structural requirements.

4a2.5.4.1.4 Load and Strength Reduction Factors

Load and strength reduction factors for the structural design of concrete shield structures and related members will be based on those prescribed in ACI 349-06 (ACI, 2007a), Sections 9.2 and 9.3, respectively.

4a2.5.4.1.5 Design of Concrete for Shielding Structures

The design of the concrete for shielding structures, including materials selection, durability requirements, quality control, mixing, placement, formwork, embedded pipes, construction joints, reinforcement, analysis, and design will conform to provisions outlined in Chapters 3 through 8 of ACI 349-06 (ACI, 2007a).

4a2.5.4.2 Exceptions for Use of ACI 349-06, and ACI 349.1R-07

ACI 349-06, Section 1.2.2, states that input and output data shall be retained as documentation when software is used for the calculation (ACI, 2007a). The software itself and other related documentation shall be retained as well. It is not required that the software be updated regularly.

SHINE does not utilize the following sections of ACI 349-06:

- Section 3.3.1: The exception portion of the section is not endorsed.
- Section 3.3.2: “These limitations may be waived if, in the judgment of the engineer, workability and methods of consolidation are such that concrete can be placed without honeycombs or voids.”
- Section 5.4.1: “If data required by 5.3 are not available, concrete proportions shall be based on other experience or information, if approved by the engineer. The required average compressive strength f'_{cr} of concrete produced with materials similar to those proposed for use shall be at least 1200 psi greater than f'_c. This alternative shall not be used if f'_c is greater than 5000 psi.”
- Section 5.6.2.3: “When total quantity of a given class of concrete is less than 50 yd^3, strength tests may be waived by the engineer if the engineer has been provided adequate evidence of satisfactory strength.” Instead, the provisions of Regulatory Position 5 of Regulatory Guide 1.142 for strength testing are utilized.
- Section 7.10.3: “It shall be permitted to waive the lateral reinforcement requirements of 7.10, 10.16, and 18.11 where tests and structural analysis show adequate strength and feasibility of construction.”

4a2.5.5 TEST PROGRAM

ANSI/ANS-6.3.1 1987; R2007, “Program for Testing Radiation Shields in Light Water Reactors (LWR)” is used as a guide in the development of a test program to be used in evaluating biological radiation shielding in the SHINE facility under normal operating conditions, including anticipated operational occurrences (ANSI/ANS, 2007).

4a2.5.6 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the biological shield are provided in Chapter 14.

4a2.6 NUCLEAR DESIGN

The IU for the SHINE facility employs an aqueous homogeneous target solution of uranyl sulfate which is irradiated by an external neutron source for production of medical isotopes. The irradiation facility consists of eight independent IU each consisting of the NDAS, the SCAS, PCLS, TOGS, LWPS, and supporting systems. Two TPSs are shared between the eight IUs. These systems operate in conjunction to achieve conditions and neutron fluxes sufficient to reach desired fission (Mo-99 production) rates. The subsections that follow outline the nuclear parameters and characteristics of the subcritical assembly throughout its life cycle. These analyses show that the system is inherently stable during both steady-state and transient operations.

4a2.6.1 NORMAL OPERATING CONDITIONS

The normal operating conditions for the subcritical assembly are most significantly affected by four factors:

a. Uranium concentrations in the target solution.
b. Fill height of the TSV.
c. Target solution temperature.
d. Neutron driver neutron generation rate.

The subcritical assembly system is designed to remain in the subcritical operating region in all operating modes. The three modes of operation relevant to nuclear design that are used to describe the subcritical assembly status are:

Mode 1 Startup Mode: Filling the TSV
Mode 2 Irradiation Mode: Operating mode (neutron driver active)
Mode 3 Post-Irradiation Mode: TSV dump valves open

Figure 4a2.1-2 shows the configuration of the SHINE subcritical assembly, and is a useful reference in understanding this subsection and the relationship between components in the different operating modes. These modes are also described in Section 7a2.3. For transfer of target solution to the RPF, a fourth mode is also described in Chapter 7.

Mode 1: Startup Mode

Prior to entering startup mode (filling the TSV [1-SCAS-01S]), the dump tank (1-SCAS-01T) is empty, and the criticality-safe hold tank (1-TSPS-03T) filled with target solution (see Figure 4a2.2-1). Chemical and physical properties of the target solution are described in Subsection 4a2.2.1. Prior to filling of the TSV, the target solution uranium concentration and other necessary parameters are measured and adjusted as necessary to ensure each of the values monitored are within the prescribed technical specification limits.

The target solution hold tank (1-TSPS-03T) is at an elevation below that of the TSV to prevent an accidental gravity-driven transfer of target solution to the TSV. A pump (1-TSPS-03P) is used to transfer the target solution to the TSV in a controlled manner, both with respect to fill rate and total volume. Pump operation time is used to estimate TSV fill volume, and is compared and contrasted against direct measurements of the TSV fill volume.

During filling, the subcritical neutron source allows the flux monitors to determine the reactivity increase of the assembly. The fixed neutron source provides a higher degree of accuracy and reliability compared to use of the neutron driver due to the known neutron source strength.

The TSV is filled in increments. The first fill increment is below the volume required for the system to go critical at the most reactive uranium concentration. After this fill increment, neutron flux measurements are able to detect gross fissile material concentration errors in the target solution.

During the fill process, a 1/M startup methodology is employed, and the startup curve is compared to the acceptable 1/M startup band. If the calculated 1/M curve violates the acceptable band, the operators dump the solution to the TSV dump tank. The system is filled to a height that is approximately 5 percent by volume below critical. The expected k_{eff} after a normal startup is approximately [Proprietary Information].

In addition to TSV fill volumes and reactivity, the temperature of the target solution is monitored via the temperature of the PCLS water. Due to the low decay power of the target solution, its temperature is approximately equal to the cooling water temperature during startup mode. Due to the operating characteristics of the SHINE system, a decrease in the temperature of the target solution results in an increase in system reactivity. Excessive cooldown of the target solution during startup is prevented by protection system (TRPS and TPCS) trips on low PCLS temperature and high neutron flux. Protection system trips drain the TSV to the TSV dump tank, which maintains the k_{eff} below 0.95 for the most reactive uranium concentration.

If at any time during the filling process, neutron flux, TSV fill volume, or target solution temperatures are determined to be outside allowable parameters, the entire contents of the TSV may be transferred to the TSV dump tank via gravity by opening the TSV dump valves. Due to the location of the TSV dump tank in the light water pool, decay heat removal requirements from the target solution are satisfied.

Mode 2: Irradiation Mode

After filling the TSV with target solution, it is isolated from 1-TSPS-03T as shown in Figure 4a2.2-1 by closing two redundant (series) fill valves. During operation of the system, there is no capability to increase reactivity by adding target solution to the TSV. Given the aqueous target solution negative void and temperature coefficients, reactivity decreases as the irradiation process begins. Furthermore, any increase in operating power levels beyond normal operating conditions results in a temperature increase and a corresponding increase in the void fraction of the target solution itself, reducing the power level.

Testing has demonstrated that the pH of the uranyl sulfate remains stable during full power operation. The TSV, TSV dump tank, and TOGS are operated as a closed system to prevent an inadvertent addition of any material that could affect reactivity or system chemistry. The introduction of water into the system as a result of the failure of the pressure boundary is analyzed in Subsection 13a2.1.2.

During irradiation of the subcritical assembly, the TOGS is used to purge radiolytic hydrogen from the headspace in the TSV. See Section 4a2.8 for a detailed discussion of the TOGS.

The PCLS has the capability to remove approximately [Proprietary Information] of heat from each TSV during irradiation. Cooling water is supplied to the external surfaces of the TSV at approximately 68°F (20°C) and exits the TSV at a maximum temperature of approximately 80°F (27°C).

The LWPS has the capability to remove approximately [Proprietary Information] of heat from the light water pool, multiplier, and tritium chamber. The temperature of the pool is expected to range between 68 to 75°F (20 to 24°C).

In addition to monitoring reactivity during TSV filling, the flux monitors are used to monitor neutron multiplication rates and power levels in the subcritical assembly during operation. Typical neutron flux ranges are shown in Table 4a2.6-1.

During irradiation mode, the neutron driver HVPS is energized to allow the production of fusion neutrons. Initially, the neutron driver tritium chamber is filled with deuterium gas, resulting in reduced neutron output. The tritium concentration in the tritium chamber is slowly increased by the delivery of tritium from the TPS, allowing a gradual temperature and power increase in the TSV. Without the neutrons produced by the neutron driver, neutron flux and fission power levels drop to near zero with a time constant defined by the delayed neutron precursor half lives.

Mode 3: Post-Irradiation Mode

Following irradiation, the neutron driver HVPS is de-energized. The TSV dump valves are opened and target solution is drained to the criticality-safe dump tank, 1-SCAS-01T. The pool serves to remove decay heat from the target solution via conduction through the tank walls. Target solution is not cooled down to startup mode temperatures within the TSV due to the known increase in reactivity due to the small water holdup in the TOGS. A minimum temperature for target solution within the TSV in Mode 3 will be defined in the FSAR.

4a2.6.1.1 Gas Management System Effects

The primary functions of the TOGS are the management of fission product gases, the recombination of hydrogen and oxygen that come out of the target solution during irradiation, and returning the condensed water to the TSV. The radiolysis of water in the system causes an anticipated increase in reactivity during operation due to the holdup of water within the TOGS condensers and piping. This increase occurs slowly as the assembly reaches steady-state operation. The total positive reactivity inserted is small in comparison to the large negative reactivity inserted by the increase in temperature and solution void during operation.

Sweep gas flow rate from TOGS to the TSV headspace is adequate to reduce hydrogen concentrations below acceptable limits, while not excessive to reduce the potential for solution entrainment in the gas flow.

Due to the sulfate chemistry of the SHINE system, nitrogen oxide (NO_x) gas release into the TOGS is minimal.

Due to the low power operation of the SHINE system, the effect of the release of gaseous fission products into TOGS on the power and flux in the SHINE system is minimal.

Further details on reactivity effects, gas holdup times, and water holdup volumes will be provided in the FSAR. A detailed discussion of the TOGS is provided in Section 4a2.8.

4a2.6.1.2 TSV Operating Characteristics

As stated earlier, the k_{eff} in the system is approximately [Proprietary Information] at cold startup conditions. As the system moves into Mode 2, various reactivity effects occur. The increase in target solution temperature from approximately 68°F (20°C) to a nominal average operating temperature of 140°F (60°C) introduces a significant amount of negative reactivity into the system due to the negative temperature coefficient of reactivity.

Formation of radiolytic gases during operation increases the void fraction of the target solution. This also causes a decrease in reactivity as a result of the negative void coefficient. As stated in the previous subsection, the holdup of water in the TOGS due to radiolysis has a positive effect on reactivity because it increases the uranium concentration of the target solution. The TOGS is designed to ensure that the volume of water held in the system is small and does not have a significant effect on reactivity. This will be analyzed further during detailed design.

The nominal operating k_{eff} is approximately [Proprietary Information]. The target solution chemistry (pH and uranium concentration) is not expected to change during irradiation other than the effects of water holdup in the TOGS. Target solution injection from the target solution hold tank during irradiation is not credible due to the TSV being located higher than the target solution hold tank and the TSV fill pump and fill valves being de-energized and interlocked during irradiation.

Following detailed design, additional TSV operating characteristics will be provided, including:

- Calculated 1/M curves, showing the approach to a cold startup k_{eff} of approximately [Proprietary Information].
- Reactivity changes during startup and operation, showing effects of temperature increase in the target solution, pressure increases in the TSV, TOGS hold-up, plutonium buildup, and fission product poisoning.
- Transients with small power oscillations expected during Mode 2 operation.
- Temperature coefficient of reactivity in Mode 1 and Mode 2.
- Void coefficient of reactivity in Mode 1 and Mode 2.

4a2.6.1.3 TSV Burn-Up and Transmutation Effects

The target solution is cleaned up periodically by the UNCS. For more information on this process see Subsections 4a2.2.1 and 4b.1.3.3. Preliminary calculations indicate there are no appreciable reactivity changes resulting from target solution burnup. Less than 0.02 percent of the U-235 in the target solution is utilized over an irradiation cycle. Effects of plutonium buildup as well as fission product poisons will be investigated during detailed design. Preliminary calculations indicate plutonium buildup and fission product poisons have a minimal effect on this system due to the low power and short irradiation cycles. No poisons are added to the target solution by design.

4a2.6.1.4 TSV Kinetic Behavior

During operation, minor power oscillations are expected due to gaseous fission product and radiolytic gas migration in the solution. The solution also mixes through natural convection due to the heat removal at the walls [Proprietary Information]. These power oscillations are small and self-limiting due to the low power density in the target solution and the negative temperature and void coefficients. In the event of a malfunction in the TOGS (recombiner, etc.), the corresponding pressure increase in the TSV could result in a partial void collapse within the target solution. This would cause an increase in reactivity; however, the k_{eff} of the system would still not reach that of cold startup conditions due to the large temperature defect present. See Subsections 13a2.1.8 and 13a2.1.2 for more information on these phenomena.

Further analysis of the TSV kinetics behavior will be conducted and described in the FSAR. The analysis will include the following:

- Effects of radiolysis on power oscillations resulting from formation and movement of voids.
- Effects of malfunctions in the recombiner and the possible resulting pressure pulses causing target solution density changes.
- Effects of temperature changes or gradients in the target solution.
- Detector responses during transients.

Work performed by ANL has shown that [Proprietary Information].

4a2.6.1.5 Interacting Effects

The main components interacting with the target solution include the neutron driver, neutron multiplier, TOGS, PCLS, and light water pool. The neutron driver is a necessary part of the assembly, as it drives the reactions in the system. Having a subcritical assembly means that there would be no appreciable power (medical isotope production) without this external source of neutrons. Operation of the neutron driver in Mode 2 decreases the k_{eff} of the system because of the increase in temperature associated with the increase in power of the system. If neutron driver operation is lost due to an abnormal event or when the neutron driver is powered down, the reactivity of the system increases due to the cooling and void collapse within the solution. As stated previously, there is a minimum temperature below which the target solution is transferred to the criticality-safe dump tank. This temperature will be set, following detailed design, with a proper safety margin to ensure that reactivity does not exceed that of cold startup conditions.

The [Proprietary Information] neutron multiplier is designed to multiply the 14 MeV fusion neutrons and lower their energy spectrum. A significant fraction of the absorptions in the multiplier result in [Proprietary Information] reactions that increase the neutron population that would otherwise enter the TSV. The multiplier is expected to last the lifetime of the facility (30 years). Reactivity effects of the neutron multiplier [Proprietary Information] and temperature variations of the multiplier will be investigated during detailed design.

Reactivity effects of the TOGS have been discussed above and will be investigated further following detailed design.

Normal operation of the PCLS serves to cool the target solution, thereby maintaining a high enough reactivity for an effective power level. If there was a loss of coolant event involving the PCLS, reactivity would decrease. During operation, this would be largely due to the negative temperature coefficient of the system. However, even at cold conditions, a decrease in water level in the PCLS results in a decrease in reactivity.

Along with its function as a heat sink and biological shield, the light water pool serves as a reflector for the subcritical assembly. Reactivity decreases significantly with a decrease in light water pool level due to the loss of reflected neutrons.

4a2.6.1.6 Solution Processing Reactivity Effects

Following each cycle (5.5 days) of irradiation the solution is processed through the MEPS to selectively remove certain elements, including molybdenum. Less than 1 percent of the overall solution is anticipated to be removed during this stage of processing. Adequate solution is included during the preparation of the target solution batch to accommodate the small losses due to isotope extraction. The uranium concentration, pH, and any other necessary parameters of the target solution are verified prior to reuse. Adjustments are made to reach the correct values prior to use in the TSV. For more information on solution processing, see Section 4b.1.

Due to the lower power of the system and low fission product inventory, selective removal of fission products due to isotope extraction is expected to have a minimal effect on system performance and reactivity. Reactivity through multiple cycles with anticipated solution processing removal rates will be provided in the FSAR.

4a2.6.1.7 Safety Considerations of Different Core Configurations

Variation of the assembly configuration occurs with changes to the target solution properties, fill height, and TSV headspace pressure. The mechanical components of the subcritical assembly are not movable during operation. The highest reactivity for the system is at cold startup conditions.

There is an operating band for the 1/M curve, allowing for small variations in solution chemistry. If the uranium concentration in the target solution is higher than anticipated, but still within the operating band, the fill operation is stopped prior to reaching the nominal fill height. This results in a slightly higher power density, which is not great enough to affect system stability (see Subsection 13a2.1.8). If the uranium concentration in the target solution is lower than anticipated, but still within the operating band, the upper fill limit may be reached prior to reaching the ideal operating k_{eff}. This results in a slightly lower production rate, but poses no additional safety risk.

In the event of a failure in the fill administrative safety controls, there are also redundant, automated trips on high neutron flux that ensure that in an off-normal event, the target solution is dumped prior to reaching critical. The trip set points include consideration for transient neutronics behavior, detector uncertainties, maximum solution fill rate, and delay for opening of the dump valves.

4a2.6.1.8 Calculated Reactivities for TSV Configurations

Calculated reactivities for TSV configurations will be provided in the FSAR. k_{eff} for the subcritical assembly remains below 1.0 for operating modes. The highest reactivity anticipated is at cold startup conditions, immediately following the filling of the TSV (Mode 1). The k_{eff} for this configuration is approximately [Proprietary Information]. During irradiation (Mode 2, Operation), k_{eff} decreases (a more subcritical TSV) due to the large temperature and void coefficients, and is approximately [Proprietary Information] at nominal operating conditions. Accordingly, the anticipated range for k_{eff} in the subcritical assembly for TSV configurations is [Proprietary Information].

4a2.6.1.9 Means to Prevent Addition of Positive Reactivity

Reactivity control in the TSV includes the following mechanisms:

- Passive control: Target solution dump tank and hold tank are placed physically below the TSV, requiring motive force to move the solution into the TSV.
- Passive control: Valve, pipe sizing, and pump size inherently limit the flow rate of solution from the target solution hold tank into the TSV.
- Passive control: Strong neutronics feedback reduces reactivity during operation due to highly negative temperature and void coefficients.
- Active engineered control: Fill pump and fill valves are closed and interlocked when system is not in fill mode.
- Active engineered control: Neutron detectors initiate trip of TSV and closure of fill valves if neutron flux exceeds predetermined flux level trip setpoint in both startup and irradiation modes.
- Administrative control: TSV is drained if PCLS or LWPS temperatures exceed acceptable limits.
- Administrative control: Measurement and independent verification of uranium concentration occurs prior to being transferred to TSV.
- Administrative control: Operators verify correct flux levels during startup process versus procedure-based acceptance criteria.

See Subsection 13a2.1.2 for discussion on accident events involving insertion of positive reactivity.

4a2.6.1.10 Technical Specifications

Potential parameters, conditions, or other items that will be probable subjects of a technical specification associated with the operating limits of the target solution are provided in Chapter 14.

4a2.6.2 TARGET SOLUTION PHYSICS PARAMETERS

This subsection discusses the TSV physics parameters and addresses the methods and analyses used to determine them. While preliminary calculations have been done to characterize the majority of these parameters, final values are not available at this preliminary

design stage. Accordingly, detailed analysis methods and their results will be provided in the FSAR following design finalization.

4a2.6.2.1 Analysis Methods and Code Validation

MCNP5-1.60, the LANL MCNP radiation transport code is used with ENDF/B-VII cross sections libraries to calculate various nuclear physics parameters for the TSV and IU. MCNP5-1.60 is developed and maintained under a configuration management plan by LANL. The code is distributed by the RSICC (LANL, 2004). Installation of the code is conducted in accordance with the software quality assurance program at SHINE and its respective contractors. The nuclear physics parameters calculated include neutron flux, reactivity (k_{eff}), dose rates, neutron lifetime, and reaction rates. These values are utilized to calculate parameters such as reactivity coefficients, flux-dependent cross sections, and fission yields for the system.

COUPLE, a module of the larger SCALE (Standardized Computer Analyses for Licensing Evaluations)-6.1.2 computational system from ORNL, is used to generate flux-dependent cross sections and fission yields for the SHINE subcritical assembly using the flux profiles calculated by MCNP5. As part of the larger SCALE computational system, the COUPLE code is developed and maintained under a configuration management plan by ORNL. The code is distributed by the RSICC.

ORIGEN (Oak Ridge Isotope Generation)-S, a module of the larger SCALE-6.1 computational system from ORNL, is used to generate source term concentrations and activities following various irradiation and decay intervals for the subcritical assembly components, target solution, and accelerator components. As part of the larger SCALE computational system, the ORIGEN-S code is developed and maintained under a configuration management plan by ORNL. The code is distributed by the RSICC.

Preliminary validation work against historic solution reactor data has been performed using MCNP5. Comparison to benchmark experiments is used to validate the acceptable modeling capability of MCNP5 for aqueous uranium solution systems. Historical data for uranyl sulfate solution systems is limited; therefore, historical data on uranyl nitrate solution systems has been used to augment the depth of the validation process. Further validation work will be performed during final design to determine estimated accuracy of calculated parameters.

4a2.6.2.2 Neutron Lifetime and Effective Delayed Neutron Fraction

The effective delayed neutron fraction for the system will be calculated with MCNP5. This is done by calculating the prompt neutron fraction and subtracting it from 1. The average neutron lifetime for the system will be calculated following detailed design, but is expected to be similar to that of normal U-235 fission systems. Comparison of calculated neutron lifetime and effective neutron fraction to similar nuclear facilities will also be conducted.

4a2.6.2.3 Axial and Radial Distributions of Neutron Flux Densities

The axial and radial distributions of neutron flux densities will be calculated using MCNP5 for expected core configurations following detailed design. Changes in these flux distributions with power level and target solution burnup will also be provided. The information on neutron flux density will include peak-to-average values for thermal hydraulic analyses (to be provided in

Section 4a2.7). Consideration will be given to hot spots as dictated by peak-to-average values for power density, temperature distribution, and void profile.

Preliminary analysis of the radial flux distributions within the TSV have been performed using MCNP simulations to understand the neutron flux density and fission density. These initial calculations show that the neutron energy spectrum for the subcritical assembly has spatial variance from the inner to outer radius, with a predominantly fast spectrum in the tritium target chamber and neutron multiplier shifting to a predominantly thermal spectrum in the TSV target solution. Neutron spectrum is predominantly thermal in the light water pool.

4a2.6.2.4 Coefficients of Reactivity

Specific values for coefficients of reactivity will be provided following detailed design. These will include temperature coefficients of reactivity for the main components of the subcritical assembly (target solution, multiplier, PCLS, and LWPS), void coefficients of reactivity, power coefficients of reactivity, and water inventory (due to off-gas hold-up) coefficients of reactivity. MCNP5 will be used for these calculations, including an estimate of the uncertainties associated with the calculational method.

This subsection will also describe the effects of burnup on these subcritical assembly operating characteristics. Due to the low operating power of the subcritical assembly and limited reuse of the solution between cleanup cycles, burnup effects are expected to have minor effects on the reactivity coefficients. Material compositions with burnup will be calculated using ORIGEN-S. The cross section libraries for the burnup calculations will be created using COUPLE and the flux distributions computed by MCNP5. The resulting material compositions from ORIGEN-S will then be used in MCNP5 to calculate reactivity coefficients after the limiting burnup interval.

Preliminary calculations show that large negative values are expected for the target solution void, target solution temperature, and power coefficients of reactivity. While the temperature coefficients of reactivity for the PCLS and LWPS may not be negative for all operating conditions (depending on final design and estimated uncertainties), the combined effects of reactivity coefficients are expected to result in strong negative feedback. Due to the nature of the aqueous system, the majority of heat is directly deposited in the target solution, leading to the target solution coefficients of reactivity being of primary importance in the overall system response. The combined system response will be analyzed using a transient analysis model in final design.

Preliminary calculations show a temperature coefficient of reactivity at least as negative as [Proprietary Information] percent millirho per degree Celsius (pcm/°C), a void coefficient at least as negative as [Proprietary Information] percent millirho per percent void (pcm/percent void), and a water inventory coefficient of reactivity of approximately [Proprietary Information] percent millirho per liter of water (pcm/[liter of water]).

This analysis, along with the accident analysis in Chapter 13, shows that the combined temperature, void, and power reactivity coefficients are sufficiently negative over the anticipated operating conditions to prevent damage to the PSB and risk to the public from subcritical assembly transients.

4a2.6.2.5 Transient Analysis

Following detailed design, a transient analysis will be performed on the behavior of the system due to bubble formation and movement within the subcritical assembly. Radiolytic gas bubbles are formed within the target solution due to the decomposition of water into hydrogen and oxygen. These bubbles migrate to the surface of the solution, and the gases are recombined by the TOGS. The hydrogen and oxygen gas production rate in the TSV is provided in Section 4a2.8.

Historical analysis has indicated that radiolytic gas bubbles affect reactivity by introducing an equivalent void volume, which moves fissile solution from regions of higher reactivity worth to regions of lower reactivity worth and causes solution expansion, which increases neutron leakage. The result is that the k_{eff} and total power of the target solution is decreased. The formation of gas molecules is proportional to the power distribution and, accordingly, is spatially-dependent.

Transient analysis will analyze the formation of bubbles in various locations in the subcritical assembly and determine the neutronics impacts. The detailed modeling and results are discussed in Subsection 4a2.6.1.4. No adverse effect on safe operations is expected.

4a2.6.2.6 Technical Specifications

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the target solution parameters are provided in Chapter 14.

4a2.6.3 OPERATING LIMITS

This subsection addresses the nuclear physics operating limits of the subcritical assembly. However, specific values are unknown at this preliminary design stage. Accordingly, specific values of the subcritical assembly operating limits will be provided following detailed design.

4a2.6.3.1 Excess Reactivity

While typical reactors load excess reactivity into the core to accommodate power defect and fuel burnup, the SCAS operates without traditional excess reactivity. The assembly is loaded in the cold state only with an amount of fissile material that brings the device to a sufficiently high subcritical multiplication factor for the production of medical isotopes. Power defect due to heat up, void production, and fission product poisons during operation decreases reactivity with no manual reactivity insertion to compensate for this decrease. Target solution burnup is minimal in the subcritical assembly.

The equivalent to excess reactivity for the SCAS is the available reactivity insertion, which is defined as the reactivity change from hot, full power conditions to cold, zero power conditions. While the neutron driver is operating, a reactivity increase (despite the subcritical state of the TSV) leads to a power increase.

The available reactivity insertion is determined based on the worth of subcritical assembly temperature coefficients, poisons, and worth associated with the radiolytic gas formation and

changes in void fraction. The upper limit on the available reactivity insertion to ensure safe subcritical assembly operation and shutdown will be determined and described in the FSAR.

The large negative temperature coefficients in the system lead to a significant reactivity defect at hot, full power conditions. However, thermal transients within the system have long time constants in comparison to other reactivity insertion events, such as void collapse. Low temperature limits are part of the TRPS to ensure safe margin to criticality.

The only poisons anticipated in the system are fission product poisons. The two fission products that have the largest effect on the system are xenon-135 (Xe-135) and samarium (Sm-149). The worth of these fission product poisons are small in comparison to the temperature and void defects.

During irradiation, radiolytic gas formation creates voids in the solution. As the gas migrates out of the target solution, it is swept into the TOGS. The total amount of gas contained within the solution leads to a reactivity defect in the solution during operation. This defect is part of the available reactivity insertion.

No experiments are performed with the SCAS; therefore, there are no reactivity effects from experiments.

Operational and safety considerations for excess reactivity are discussed. From Subsection 13a2.1.2, only three initiating events are considered credible:

- Target solution temperature reduction (e.g., excessive cooldown).
- Increase in the target solution density (e.g., due to pressurization) during operations.
- Additional target solution injection during fill/startup and irradiation operations.

A temperature range for irradiation will be set following detailed design. In the event that the target solution temperature falls below this range, the target solution is transferred to the TSV dump tank. An increase in target solution density due to pressurization and void collapse during irradiation (Mode 2) causes a positive reactivity insertion; however, the reactivity does not exceed that of initial startup. Multiple administrative controls, passive safety features, and active safety features are in place to prevent additional target solution injection at TSV fill (Mode 1). Target solution injection during irradiation (Mode 2) is not considered credible. There are several controls that prevent or provide mitigation for the consequences of an excess reactivity insertion event. These controls are discussed in Subsection 13a2.2.2 for the three above initiating events. Credible excess reactivity insertion accidents do not result in damage to the PSB.

4a2.6.3.2 Negative Reactivity

For the final design, the amount of negative reactivity present at startup (to ensure safe margin from critical) will be determined. Expected reactivities at the completion of the startup process are approximately [Proprietary Information]. The magnitude of the negative temperature and void coefficients of reactivity will be evaluated.

Adequate shutdown margin is achieved under credible conditions by transferring the target solution to the TSV dump tank. In the TSV dump tank, the target solution is maintained below a k_{eff} of 0.95 at cold conditions for the most reactive uranium concentrations for normal, abnormal,

and accident conditions. Two fully-redundant dump valves provide shutdown capability for the TSV. The TSV can be safely shut down to a k_{eff} less than 0.95 with the failure of one dump valve with no operator intervention.

Discussion of how the margin from critical and reactivity coefficients will be verified and will be provided in the FSAR. Validation is expected to involve physics tests during facility startup.

The analyses will include relevant uncertainties and error limits.

4a2.6.3.3 Neutron Flux Density, Power Density, and Stability Criteria

Following detailed design, stability criteria will be defined for acceptable performance. Methods for maintaining power oscillations within operational or safety limits will be described in the FSAR and will include operational limits on parameters such as power density and neutron flux density. Small power oscillations occur in the TSV as a result of normal anticipated reactivity variations within the target solution and normal expected driver output variations. The initiating events or scenarios include (see Subsection 13a2.1.8):

- Radiolytic bubble formation and collapse within the target solution and at the solution surface.
- Migration of radiolytic bubbles to the target solution surface with removal by the off-gas system.
- Target solution circulation due to temperature/void non-uniform distributions.
- Variations in the neutron driver production rate.
- Excessive reactivity insertions (see Subsection 13a2.1.2).

Operating at a subcritical condition with a low power density and negative temperature and void reactivity coefficients provides TSV stability and self-limiting power oscillations. Furthermore, if a power oscillation results in an increase in neutron flux higher than anticipated, the TRPS setpoint on high neutron flux automatically de-energizes the neutron driver and opens the TSV dump tank valves. The target solution is transferred (by force of gravity) into the criticality-safe TSV dump tank (see Subsection 13a2.1.8).

Based on experiments conducted at historic aqueous homogeneous reactor (AHR) facilities (Russian ARGUS facility and French SILENE facility), steady state, stable core conditions could be sustained at power densities below approximately 1.8 thermal kilowatts/liter (kWt/L) (23,467 british thermal units per hour per gallon [Btu/hr-gal]) (BNL, 2010; IAEA, 2008; Barbry Francis, 2007). Even assuming a lower than anticipated solution volume of [Proprietary Information] and a power level of [Proprietary Information] (110 percent of maximum TSV power), the resulting maximum power density is only [Proprietary Information]. This is more than a factor of two below where previous AHR facilities have demonstrated stable operations. This analysis is discussed further in Subsection 13a2.1.8.

Following detailed design, limiting neutron flux and thermal power densities will be quantified. Normal operating conditions and credible off-normal events will be considered.

4a2.6.3.4 Limiting Core Configuration

The limiting core configuration is determined by the maximum allowable uranium enrichment combined with the maximum allowable uranium concentration in the target solution. The maximum uranium enrichment and concentration result in the smallest solution volume and highest power density. The neutron flux density and thermal power density with the limiting core configuration will be analyzed following detailed design. The resulting power density will be well below the 1.8 kWt/L (23,467 Btu/hr-gal) that has historically been shown to provide stable operation within AHRs.

4a2.6.3.5 Transient Analysis

Following detailed design, a transient analysis will be provided to demonstrate that normal power oscillations during operation will be small and self-limiting. A transient analysis of the abnormal events discussed previously will also be provided. This analysis will show that no credible events, including void collapse during operation and inadvertent target solution injection during startup, will lead to damage of the PSB. See Chapter 13 for further analysis of reactivity additions under accident conditions.

4a2.6.3.6 Redundancy and Diversity of Shutdown Methods

As indicated in Subsection 4a2.6.3.2 above, shutdown margin is achieved under credible conditions by transferring the target solution to the TSV dump tank by redundant dump valves. The target solution is maintained in a criticality-safe shutdown condition (k_{eff} less than 0.95) in locations outside the TSV by passive engineered controls. TRPS trips also lead to de-energizing the HVPS of the NDAS, which eliminates fusion neutron production, and terminates the fission process within the subcritical assembly.

4a2.6.3.7 Technical Specifications

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the subcritical assembly are provided in Chapter 14.

Table 4a2.6-1 Typical Neutron Flux Ranges

Typical Neutron Flux Ranges

Operational flux ranges in the TSV	Approximately [Proprietary Information]
Expected detector flux ranges during irradiation	Approximately $1x10^{7}$ to $1x10^{10}$ n/cm^{2}-s
Expected detector flux ranges during startup	Approximately $1x10^{0}$ to $1x10^{5}$ n/cm^{2}-s

4a2.7 THERMAL HYDRAULIC DESIGN

This section presents the information and analyses necessary to show that sufficient cooling capacity exists to prevent target solution overheating and loss of target solution barrier for anticipated system operating conditions.

4a2.7.1 HEAT REMOVAL SYSTEMS

4a2.7.1.1 Thermal-Hydraulic Design

The SHINE thermal-hydraulic design includes systems that provide heat removal from the subcritical assembly during normal irradiation and shut down operations. These systems include the PCLS and the LWPS for each IU. For detailed system descriptions of the PCLS and LWPS, see Section 5a2.2, Primary Cooling System.

4a2.7.1.1.1 PCLS Cooling Loop

The PCLS cooling loop removes approximately [Proprietary Information] of heat, via forced convection, from each Target Solution Vessel (TSV) (eight total) during full-power IU operation by circulating deionized water in the upward direction [Proprietary Information] along the outside of the TSV walls during normal operation. The heat absorbed by the PCLS water is transferred to the radioisotope process facility cooling system (RPCS) by a plate and frame heat exchanger. See Figure 4a2.7-3 for the process flow diagram of the PCLS cooling loop (the legend for the process flow diagrams is provided on Figure 1.3-6).

4a2.7.1.1.2 LWPS Cooling Loop

The LWPS cooling loop removes approximately [Proprietary Information] of heat from each neutron multiplier and tritium target chamber (eight total each) during full-power IU operation by circulating the LWPS water through the cooling loop and returning it to the SASS. The heat absorbed by the LWPS water is transferred to the RPCS by a plate and frame heat exchanger. See Figure 4a2.7-4 for the process flow diagram of the LWPS cooling loop.

The cooling loops are sized to remove heat generated during IU full-power operation. A water treatment system is included in the cooling systems to reduce corrosion and scaling to ensure efficient removal of heat, as described in Section 5a2.4.

4a2.7.1.1.3 Light Water Pool Heat Removal

Heat removal from the light water pool is provided by the LWPS. See Subsection 4a2.4.2.2 for details on LWPS heat removal.

4a2.7.1.1.4 Gas Management System

Gas management is provided by the TSV off-gas system (TOGS). TOGS provides a closed loop sweep gas over the head space of the TSV. There are two air-to-water heat exchangers in the TOGS loop for the purpose of condensing solution entrained in the sweep gas and removing heat generated by the recombination process. These heat exchangers are cooled by the RPCS. A discussion of TOGS heat loads on RPCS will be provided in the FSAR.

4a2.7.2 COOLANT HYDRAULIC CHARACTERISTICS OF THE TARGET SOLUTION VESSEL

The PCLS removes heat directly from the TSV ([Proprietary Information] around the TSV walls) during normal irradiation and shut down operations. [Proprietary Information]. The subcritical assembly support structure (SASS) provides the shell side pressure boundary to direct the coolant flow past the TSV.

The PCLS volumetric coolant flow rate is calculated to be 74 gpm (4.7 Liters per second [L/s]). The coolant supply to the lower plenum of the SASS is at approximately 5 psig and 68°F (20°C). The coolant return from the TSV is approximately 3 psig and 80°F (26.7°C). This corresponds to a pressure difference of 2 psig across the TSV within the SASS. See Table 5a2.2-1 for the PCLS specifications.

The TSV pressure is maintained slightly below atmospheric to prevent any outleakage in the event of a primary system boundary (PSB) breach. During irradiation, the target solution is maintained at a nominal temperature of 140°F (60°C).

Plating out of chemicals on the TSV surfaces is not expected in the operating temperature range of the SHINE process.

Temperature variation and wave propagation in the target solution caused by vibration will have a negligible effect on primary coolant flow conditions during normal operating modes.

Due to the pressure differential between the PCLS and the TSV volume, it is expected that a rupture of [Proprietary Information] would result in leakage of coolant into the TSV volume. This will cause a dilution of the target solution and a reduction in the system k_{eff}. Some target solution will leak out of the TSV and into the PCLS. The impact on system thermal-hydraulics in this situation is negligible. This shutdown of the PCLS would represent a change in the coolant flow from forced to natural convection.

A loss of active cooling could cause the temperature of the target solution to rise above the allowable limit. The TRPS protects against loss of PCLS flow or high temperature in the PCLS by shutting down the IU neutron driver and dumping the target solution to the TSV dump tank, as described in Subsection 7a2.4.1.1. The TSV dump tank is submerged in the light water pool.

The light water pool provides emergency cooling by passively absorbing decay heat from the TSV dump tank via conduction in accident scenarios when the PCLS and LWPS cooling loops are unavailable (e.g., cooling pumps off-line).

In the event of a loss of off-site power (LOOP), the light water pool serves as the decay heat sink for the TSV. The light water pool is sized such that it can remove decay heat from the target solution without active cooling.

The RPCS serves as the secondary cooling system for the PCLS, LWPS, and TOGS. In the event of a loss of RPCS function, the cooling capacity of these systems will be lost. In this event, the IU will be shut down and the light water pool will provide target solution decay heat removal.

4a2.7.3 TARGET SOLUTION THERMAL POWER DENSITY DISTRIBUTION

A detailed analysis of the thermal and nuclear parameters of the target solution (power density, heat flux, neutron flux, and detailed temperature gradient profile) will be determined during final design. For physical properties of the target solution (e.g., pH, uranium density) in the TSV, see Table 4a2.2.1-1.

4a2.7.4 THERMAL-HYDRAULIC METHODOLOGY

Correlation-based methodology will be used for safety calculations in the FSAR. The safety calculations will include verification of adequate heat removal in the TSV during anticipated operating and accident conditions. These calculations will use heat generation rates calculated by the MCNP5 neutronics models, and will determine the thermal profile in the cooling spaces between the target chamber, multiplier, SASS walls, and TSV. Wall heat transfer will be modeled using conduction equations, and bulk TSV temperature will be calculated using natural convection correlations from established literature. The thermal-hydraulics and neutronics coupling effects (thermal changes result in neutronic changes and neutronics changes result in thermal changes) will be included in the calculation process.

Correlation-based heat removal calculations will also be utilized in the transient model predictions in order to accurately model coupled neutronics and thermal-hydraulics behavior for anticipated system transients, including neutron driver source strength output changes and credible deflagrations.

The safety calculations will ensure that the target solution temperatures do not exceed the limits provided in Table 4a2.2-3.

For the purposes of predicting the normal operating conditions of the subcritical assembly, best-estimate methodology using computational fluid dynamics software is being utilized. Thermal-hydraulics inside the TSV are simulated using the ANSYS fluid dynamics simulation software (Fluent and CFX). The multi-physics capability allows for volumetric heat and gas generation (fission heating and radiolysis) with the Navier-Stokes solution of the natural-convection-driven closed-cavity flow. Simulations are employed for the primary purpose of estimating the steady state uranyl sulfate temperature which is determined by the effectiveness of removing heat through [Proprietary Information] inner/outer annular TSV surfaces. Preliminary simulations of target solution flow and heat transfer in the TSV have estimated normal average TSV temperature to be approximately 122°F to 158°F (50°C to 70°C) during operation at 100 kWt in the TSV.

This section will provide details on the thermal-hydraulic methodology for the transfer of heat to the PCLS. Uncertainties in thermal-hydraulic and nuclear parameters and such engineering factors as [Proprietary Information] wall thickness and the buildup of layers both inside [Proprietary Information] walls and on the outside of the [Proprietary Information] walls will be considered. The calculations will be based on measured uranyl sulfate properties. Detailed analysis methods and their results will be provided in the FSAR.

4a2.7.4.1 Code Validation

Work is being performed at LANL and University of Wisconsin-Madison to model the thermal-hydraulics using commercial CFD software (FLUENT and CFX codes) to define system temperature distributions. Validation and verification of software has not yet been performed. Validated and verified models will be used for the FSAR-provided results.

4a2.7.5 IMPACT OF OPERATING CONDITIONS ON THERMAL-HYDRAULICS

4a2.7.5.1 Target Solution Conditions

Because TOGS and the TSV are connected, target solution pressure is controlled by the TOGS. The target solution temperature is driven by system neutron flux. If PCLS flow decreased below allowable values or PCLS temperature increases above acceptable values, the IU will shut down and limit thermal damage to the IU components.

The limiting core configuration for thermal-hydraulics analysis will be provided in the FSAR. The maximum solution level is [Proprietary Information]two separate overflow lines that drain excess solution to the TSV dump tank. As discussed in 4a2.4.1.2, the minimum expected target solution level is [Proprietary Information].

The target solution is stable with respect to chemical and physical properties over the course of an irradiation cycle (see Subsection 4a2.2.1).

Void formation in the target solution is expected, and will be factored into the nuclear calculations (void coefficients of reactivity) and thermal hydraulic calculations for final design. Target solution voids are not expected to reduce cooling capability.

4a2.7.5.2 Target Solution Vessel Integrity and Target Solution Stability

Target solution vessel integrity is maintained and solution boiling is prevented through the management of several parameters:

- Target solution pressure and temperature limits.
- Target solution pH is stable, as discussed in the SHINE operating conditions (see Subsection 4a2.2.1.9).
- The solubility of uranium and fission products in the target solution are well below solubility limits (see Table 4a2.2-1).
- The TOGS recombiner is designed to have sufficient radiolytic gas recombination capacity for operating conditions, and hydrogen concentration is monitored.

4a2.7.6 COOLING SYSTEM DESIGN BASES

The thermal-hydraulic design includes systems that provide heat removal to the subcritical assembly during normal irradiation and shut down conditions. These systems in the SHINE facility include the PCLS and the LWPS for each IU. For detailed system descriptions of the PCLS and LWPS, see Section 5a2.2.

4a2.7.7 COOLING PERFORMANCE

Small amounts of radiolytic gases will be generated in the primary coolant during irradiation. Significant void formation in the SASS is not expected during normal operation. The PCLS flow past the TSV is vertical and upward, which would prevent the accumulation of gases and mitigate coolant void formation.

4a2.7.8 BULK BOILING OF THE TARGET SOLUTION

Temperature monitoring of the target solution is provided for indication for the operators. The target solution temperature is also monitored through the temperature of the PCLS loop. Temperature and flow indication on the PCLS ensure that adequate cooling is being provided to the TSV. If PCLS flow decreases below acceptable levels, the TRPS shuts down the neutron driver and opens the TSV dump valves. The light water pool will provide natural cooling to the solution contained within the dump tank.

If target solution boiling were to occur during irradiation, the target solution and its off gases will be confined to the PSB and not present a significant radiological hazard. The TSV off-gas system (TOGS) shielded cell and IU cell surround PSB components, and both contain significant radiological shielding.

4a2.7.9 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the thermal hydraulic design are provided in Chapter 14.

4a2.8 GAS MANAGEMENT SYSTEM

4a2.8.1 SYSTEM DESCRIPTION

The gas management system is the TOGS. The TOGS removes radiolysis and fission product gases from the TSV during irradiation operation to maintain safe and effective operating conditions.

4a2.8.2 SYSTEM PROCESS AND SAFETY FUNCTIONS

The process functions of the TOGS are listed below:

- Contain fission product off-gas generated during target solution irradiation from the TSV.
- Remove iodine from the off-gas to protect catalytic recombiner efficiency.
- Recombine hydrogen and oxygen in the off-gas into water vapor to reduce the amount of hydrogen in the system.
- Limit water loss in the target solution by condensing water vapor and returning it to the TSV.
- Recirculate the off-gas back to the TSV in a closed circulation loop.
- After each TSV batch is completed, purge the off-gas to the noble gas removal system (NGRS) for noble gas holdup and decay. See Subsection 9b.6.2 for more information.

The safety functions of the TOGS are listed below:

- Limit personnel exposure to radioactive and hazardous materials.
- Maintain the integrity of the primary system boundary during normal operating conditions and credible accident events.
- Maintain a hydrogen gas maximum limit less than the LFL, which is 4 percent by volume, to prevent deflagration and provide a TRPS trip signal if acceptable hydrogen concentrations are violated.
- Prevent egress of radioactive gases from the system by maintaining a negative pressure in relation to the surrounding IU cell atmosphere.
- Remove iodine from the off-gas to limit potential dose consequences during accident conditions.

4a2.8.3 TSV OFF-GAS PROCESS FLOW DIAGRAM

Figure 4a2.8-1 shows the process flow diagram of the TOGS and its major components.

4a2.8.4 TOGS MAJOR COMPONENTS AND SYSTEM INTERFACES

Table 4a2.8-1 lists the major components in the TOGS shown in Figure 4a2.8-1. The description of the components includes design specifications of the TOGS.

The physical layout of the TOGS components is found in Figure 4a2.8-2. Table 4a2.8-2 lists the TOGS system interfaces.

The specified materials of construction of the TOGS components are chemically compatible with the evolved fission product, radiolytic, and sweep gases (e.g., 316 stainless steel). All seals/ gaskets are chemically compatible with the process fluids. This ensures that possible corrosion damage to the system is reduced. The TOGS components are designed and fabricated in accordance with the codes and standards listed in Table 4a2.8-1.

The TOGS components are designed to withstand system pressures that could occur during credible TSV power fluctuations to avoid breaching the primary system boundary. The components will be designed to withstand credible hydrogen deflagration.

The TOGS shielded cells and IU cells are designed to limit the release of radioactive and hazardous materials in the event that the TOGS system leaks into the cells. See Section 9a2.1 for more information on the RCA ventilation system.

4a2.8.5 ABNORMAL CONDITIONS

The TOGS components are part of the primary system boundary for fission products in the system. The TOGS components maintain the primary system boundary during accident scenarios as well as during normal operation. The accident scenarios and responses discussed in Chapter 13 ensure that the primary system boundary limits fission products from leaving the TOGS and that the hydrogen LFL is not reached.

If the neutron driver shuts down due to power loss or an off-normal event, the blower and recombiner beds continue to operate while the hydrogen production rate subsides. The blower and recombiner beds are on facility uninterruptible electrical power supply system (UPSS) power to run for hydrogen recombination after a LOOP. The current design of the UPSS provides for a two hour duration of Class 1E power following LOOP. The effects of post-shutdown hydrogen generation have not been fully determined. Final hydrogen generation rates and associated mission time for the emergency power system will be determined as part of final design. Design studies are on-going to establish design features that could be used for maintaining stable long-term post-accident conditions assuming that the off-site power is not available. These potential design features include:

- Passive hydrogen recombiners
- On-site safety-related emergency diesel generators
- Robust piping systems
- Deflagration flame arrestors
- Other potential design features identified during the detailed design

The emergency power system will operate for the required mission time as delineated in the final analysis which will be provided in the FSAR.

If the hydrogen concentration exceeds 2.5 percent by volume, an alarm alerts the operator to take action. If the neutron driver is shut down, the blowers and recombiners remain active to circulate and recombine the hydrogen and oxygen in the off-gas. The neutron driver is automatically shut down when a 3 percent by volume hydrogen concentration is reached at the hydrogen detectors.

There is no significant NO_x gas present in the off-gas; therefore, there is no postulated accident scenario resulting in the release or accumulation of NO_x gas.

A pressure safety valve is connected to the TOGS piping to passively prevent an over-pressurization within the PSB, which may cause structural damage to the IU. The setpoint of the pressure safety valve will not exceed the design pressure of the PSB components. This setpoint value will be provided in the FSAR.

4a2.8.6 RADIATION AND HYDROGEN CONCENTRATION CONTROL/MONITORING

Purging the off-gas to the NGRS prevents a buildup of gaseous fission products. The TOGS components are designed and shielded to limit personnel exposure to radiation.

Hydrogen concentration monitoring instrumentation is included to measure the concentration of hydrogen in the TOGS piping. The TOGS system is designed to maintain hydrogen concentrations up to 2 percent during normal operation. If the limit of 2.5 percent by volume is exceeded, an alarm alerts the operator. If the hydrogen concentration reaches 3 percent by volume, the neutron driver is automatically shut down.

4a2.8.7 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the TOGS are provided in Chapter 14.

Table 4a2.8-1 TSV Off-Gas System Major Components
(Sheet 1 of 2)

Component	Description	Code/Standard
TSV Off-Gas Condenser (1-TOGS-01A-A-H)	The condenser is located in the gas discharge line from the TSV. It cools the off-gas from 140°F to 100°F. The condenser is designed with a design margin of greater than 15% for the heat transfer area.	ASME B31.3; ASME BPVC Section VIII
TSV Off-Gas Demister (1-TOGS-01F-A-H)	The demister is located downstream of the TSV off-gas condenser and allows condensed liquids to collect and flow back into the TSV to reduce water loss in the target solution during irradiation.	ASME B31.3
TSV Off-Gas Iodine Removal Beds (1-TOGS-01S-A-P)	The beds are located downstream from the demister. The beds remove iodine isotopes from the off-gas. The beds remove approximately 95% of the iodine in the off-gas. These beds may not be required if TOGS test-rig testing verifies that iodine amounts are not significant enough to require removal beds.	ASME BPVC Section VIII
TSV-Off Gas Blower (1-TOGS-01C-A-P)	The blower is located downstream from the iodine removal beds. It ensures that the fission product, radiolytic, and sweep gases from the TSV are circulated through the TOGS piping. During system purge, the blower transfers the off-gas to the NGRS after irradiation is complete. An air inlet valve and the NGRS purge valve are opened and the blower is run so it displaces the off-gas to the NGRS. Once the off-gas is displaced to NGRS, the valves are closed to return to normal operation.	ANSI/API 617

Table 4a2.8-1 TSV Off-Gas System Major Components
(Sheet 2 of 2)

Component	Description	Code/Standard
TSV Off-Gas Recombiner Beds (1-TOGS-02S-A-P)	The recombiner beds are located downstream of the blowers. The recombiner beds combine greater than or equal to 90% of the hydrogen with oxygen to form water vapor in a single pass. The flow velocity across the recombiner beds is maximized to achieve high conversion efficiency. The recombiner beds may require a heater to heat the beds for decay hydrogen recombination to ensure the catalyzed recombination reaction continues.	ASME BPVC Section VIII
TSV Off-Gas Recombiner Condenser (1-TOGS-02A-A-H)	The condenser is downstream of the recombiner beds. The condenser cools the gas exiting the recombiner to approximately 100°F.	ASME B31.3; ASME BPVC Section VIII

References: ANSI/API, 2009; ASME, 2011; ASME, 2012

Table 4a2.8-2 TSV Off-Gas System Interfaces

System	Interface Description
Radioisotope Process Facility Cooling System (RPCS)	Interfaces at the RPCS supply and return connections to 1-TOGS-01A and 1-TOGS-02A. The RPCS cools and condenses the off-gas water vapor passing through 1-TOGS-01A and 1-TOGS-02A.
Noble Gas Removal System (NGRS)	Interfaces at the TOGS purge piping that transfers the off-gas to the NGRS after the irradiation cycle is complete.
Subcritical Assembly System (SCAS)	Interfaces at the TOGS inlet and outlet connections with the TSV. Fission product, radiolytic, and sweep gases are drawn into the TOGS circulation loop from the TSV.
Normal Electric Power Supply System (NPSS)	The NPSS provides power to TOGS components and instrumentation.
Uninterruptible Electrical Power Supply System (UPSS)	1-TOGS-01C and 1-TOGS-02S are connected to the facility UPS system to ensure operation in the event of power loss.
TSV Reactivity Protection System (TRPS)	The TOGS interfaces with the TRPS to provide safety-related trips of the TSV and neutron driver. Hydrogen monitoring by TOGS is utilized by TRPS.
TSV Process Control System (TPCS)	The TPCS provides control signals to TOGS.

4a2.9 REFERENCES

ACI, 2007a. Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, ACI 349-06, American Concrete Institute, Farmington Hills, MI, 2007.

ACI, 2007b. Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures, ACI 349.1R-07, American Concrete Institute, Farmington Hills, MI, 2007.

ANSI/ANS, 2006a. Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, ANSI/ANS-6.4-2006, American National Standards Institute, La Grange Park, IL, 2006.

ANSI/ANS, 2006b. Specification for Radiation Shielding Materials, ANSI/ANS-6.4.2-2006, American National Standards Institute, 2006.

ANSI/ANS, 2007. Program for Testing Radiation Shields in Light Water Reactors (LWR), ANSI/ANS-6.3.1-1987 (R2007), American National Standards Institute, La Grange Park, IL, 2007.

ANSI/API, 2009. Axial and Centrifugal Compressors and Expander-compressors for Petroleum, Chemical and Gas Industry Services, Standard 617, American Petroleum Institute, January 1, 2009.

ANL, 2012. Final FY-12 Progress Report on the Cleanup of Irradiated 130gU/L Uranyl Sulfate SHINE Target Solutions, Argonne National Laboratory, 9/29/2012.

ASME, 2011. Boiler & Pressure Vessel Code - Rules for Construction of Pressure Vessels, Section VIII, American Society of Mechanical Engineers, July 1, 2011.

ASME, 2012. Code for Pressure Piping, B31.3-2012, American Society of Mechanical Engineers, January 10, 2013.

Baker, 1944. Baker, C., et al., Water Boiler, LA Report 134, Los Alamos National Laboratory, September 4, 1944.

Barbry Francis, 2007. French CEA Experience on Homogeneous Aqueous Solution Nuclear Reactors, Commissariat a l' Energie Atomique CEA/France, June 2007.

Beall, 1954. Beall, S., et al., The Homogeneous Reactor Experiment – A Chemical Engineering Pilot Plant, Chemical Engineering Progress, Vol. 50, No. 5, May 1954.

BNL, 2010. Aqueous Homogenous Reactor Technical Panel Report, BNL-94462-2010, Brookhaven National Laboratory, December 10, 2010.

Gamble, 1959. Gamble, D., A Proposed Model of Bubble Growth during Fast Transients in the KEWB Reactor, Paper 25-3, American Nuclear Society, June 15, 1959.

IAEA, 2008. Homogeneous Aqueous Solution Nuclear Reactors for the Production of Mo-99 and Other Short Lived Radioisotopes, TECDOC-1601, International Atomic Energy Agency, September 2008.

Lane, 1958. Lane, J., Fluid Fuel Reactors, Part 1, Aqueous Homogeneous Reactors, Oak Ridge National Laboratory, June 1958.

LANL, 2004. MCNP5_RSICC_1.30, LA-UR-04-5921, MCNP Monte Carlo Team, X-5, Los Alamos National Laboratory, Los Alamos, NM, 2004.

Lee, 1952. Lee, J., et al., The Density of Uranyl Sulfate Solutions and the Determination of Uranium Concentration by Density Measurements, ORNL-1332, Oak Ridge National Laboratory, June 18, 1952.

NCSU, 1955. NCSCR-1, Raleigh Research Reactor Critical on September 1953, North Carolina State University, Shutdown June 1955.

Olsher, 2006. A Practical Look at Monte Carlo Variance Reduction Methods in Radiation Shielding, *Nuclear Engineering and Technology*, Volume 38, No. 3, pp. 225–230, R.H. Olsher, April 2006.

ORNL, 1952. Physical Properties of Uranyl Sulfate at Atmospheric Pressure, CF52-3-253, Oak Ridge National Laboratory, March 10, 1052.

Parkins, W., 1958. Aqueous Homogeneous Type Research Reactors, Second International Conference on Peaceful Uses of Atomic Energy, March 19, 1958.

Secoy, 1948. Secoy, C., The System Uranyl Sulfate – Water, II. Temperature – Concentration Relationships above 250°C, ORNL-98, Oak Ridge National Laboratory, July 7, 1948.

Secoy, 1955. Secoy, C., Preprint65, Survey of Homogeneous Reactor Chemical Problems, Nuclear Engineering and Science Congress, American Chemical Society, December 12 – 16, 1955.

Silverman, 1961. Silverman, Louis, et al., Homogeneous Nuclear Reactor Fuel Composition, Patent No. 2,978,399, April 4, 1961.

Wilson, 1958. Wilson, R., et al., Aqueous Homogeneous Type Research Reactors, P/1543, Atomics International, June 1958.

4b RADIOISOTOPE PRODUCTION FACILITY DESCRIPTION

Section 4b describes the systems structures and components associated with the radioisotope production facility (RPF). These descriptions are associated with systems that are not covered in Section 4a2, which covers the IF.

4b.1 FACILITY AND PROCESS DESCRIPTION

4b.1.1 INTRODUCTION

This chapter describes the design of the RPF and the processes employed within it. The primary function of the facility is to extract, purify, package, and ship medical isotopes.

4b.1.2 FACILITY DESCRIPTION

The facility design includes a number of intrinsic safety features that represent good engineering practice for nuclear processing facilities.

- Radioactive material processing areas and related SSCs are located in Category I structures designed to survive design basis earthquake loadings and other external events. Additionally, SSCs co-located with safety-related SSCs are reviewed and supported in accordance with II over I criteria. This avoids any unacceptable interactions between SSCs that could degrade the confinement of radioactive or chemically hazardous materials and result in an uncontrolled release.
- Radioactive materials are contained within piping and processing systems. These systems are located within shielded hot cells and vaults. These hot cells are ventilated with systems that are independent of the occupied zone ventilations.
- Radioactive material transfers between hot cells and vaults are minimized to the extent possible.
- Tanks, piping, and equipment that contain fissile material are designed to be criticality-safe by geometry.
- Piping systems for radioactive liquid transfers between processing areas are contained within hardened, shielded pipe chases.
- Hot cells, tank vaults, and shielded pipe chases drain to sumps that include leak detection to alert operators to a breach of primary confinement.
- Tanks include overflow lines that are hard-piped to the criticality-safe collection tank in the low point of the building.
- Operating areas are monitored with the continuous air monitoring system (CAMS), radiation area monitoring system (RAMS), and a fail-safe criticality accident alarm system (CAAS).
- Hot cells are isolated from the building heating, ventilation, and air conditioning (HVAC) system upon detection of a leak, to prevent the spread of contamination.
- Radioactive materials are excluded from normally occupied areas, except for transfers within suitably shielded containers.

4b.1.3 PROCESS DESCRIPTION

The RPF has been divided into a number of systems that represent discrete areas for design development. The systems associated with processing are:

- Target solution preparation system (TSPS).
- Molybdenum extraction and purification system (MEPS).
- Uranyl nitrate conversion system (UNCS).
- Noble gas removal system (NGRS).
- Process vessel vent system (PVVS).

- Radioactive liquid waste evaporation and immobilization (RLWE).
- Aqueous radioactive liquid waste storage (RLWS).
- Organic liquid waste storage and export (OLWS).
- Molybdenum isotope product packaging system (MIPS).
- Radioactive drain system (RDS).

A description of the operational and safety functions of each system is included in the section for each system.

The bounding radionuclide inventory for one target solution batch entering the RPF is given in Chapter 11. The special nuclear material (SNM) inventory is estimated to be less than [Security-Related Information] for the entire facility, [Proprietary Information].

The processing systems are located within the RCA. This is a restricted area, with routine access limited to properly trained personnel. Radiation workers are issued a thermoluminescent dosimeter (TLD) and appropriate personal protective equipment (PPE). The RCA design includes a ventilation system with active monitoring equipment and audible and visual alarms.

The facility uses SNM within processing systems. The chemical forms and maximum quantities of SNM in each system are included in the subsequent process descriptions. Details of SNM controls are included in Subsections 4b.4.1 and 4b.4.2.

4b.1.3.1 Target Solution Preparation System (TSPS)

4b.1.3.1.1 Process Functions

- Prepare target solution (uranyl sulfate) by reacting uranium oxide and sulfuric acid in the uranyl sulfate preparation tanks.
- Dissolve uranium metal (low-enriched) in nitric acid to form uranyl nitrate in the uranium metal dissolution tank.
- Transfer the uranyl nitrate from the dissolution tank to the recycle uranyl nitrate hold tank to be processed through thermal denitration part of the UNCS.
- Receive and store uranium oxide from the thermal denitrator in storage racks to be used in preparation of future target solutions.
- Transfer and meter target solution into the TSV.

4b.1.3.1.2 Safety Functions

Prevent inadvertent criticality through inherent design of equipment.

4b.1.3.1.3 Process Description

The TSPS generates two products:

a. Uranyl nitrate in 12 M nitric acid from uranium metal.
b. Uranyl sulfate in [Proprietary Information] sulfuric acid from uranium oxide.

Uranyl nitrate is prepared by dissolving uranium metal in nitric acid, according to the reaction below:

$$U + 6HNO_3 \rightarrow UO_2(NO_3)_2 + 3NO_2 + NO + 3H_2O \qquad \text{(Equation 4b.1-1)}$$

The uranyl nitrate is then routed to the thermal denitration (TDN) system to produce uranium oxide. The uranium dissolution tank (1-TSPS-02T-A/B) is heated to between 167 to 185°F (75 to 85°C) for the dissolution step. The tank volume is 8.6 gal. (32.6 L). The tank design does not include any mechanical agitation. The uranium concentration in the uranyl nitrate product is 0.417 lbU/gal. (50 gU/L).

Target solution is prepared from two sources:

a. Target solution recycled directly from the UNCS.
b. Uranyl sulfate prepared by dissolution of uranium oxide in sulfuric acid.

Recycled target solution is staged in the target solution hold tank (1-TSPS-03T-A-H). It may require some minor chemical adjustment to ensure it meets the target solution specification before it is pumped into the TSV.

Uranium oxide for dissolution is recycled from the TDN process. As a provisional assumption, recycled uranium oxide makes up 99.75 percent of the total uranium in the target solution, with the balance being made up with uranium oxide produced from fresh uranium.

The uranium oxide is reacted with sulfuric acid (H_2SO_4) solution to convert the uranium oxide to uranyl sulfate, with enough excess H_2SO_4 such that the target solution contains 0.1 M H_2SO_4. The range of uranyl sulfate concentration is [Proprietary Information] and the pH is [Proprietary Information].

The online uranyl sulfate preparation tank (1-TSPS-01T-A/B) is heated to 158°F (70°C). The liquid outlet pipe includes an inline filter to remove any un-dissolved uranium. There are two uranyl sulfate preparation tanks, each sized to hold one batch of target solution.

From the uranyl sulfate preparation tank, the target solution is transferred to one of the target solution hold tanks (1-TSPS-03T-A-H). There is one target solution hold tank per TSV, and the hold tanks are located below the level of the TSV so the tank cannot drain into the TSV.

The SNM within the TSPS system is:

- U-235. This is present in the form of uranium metal, uranyl nitrate, uranium oxide and uranyl sulfate, depending on its location within the process. The maximum inventory of uranium (U-235) in the system is [Security-Related Information]. This is divided between a number of process tanks and storage racks. The U-235 present in the facility is derived from LEU, which is less than 20 percent enriched. The nominal enrichment is 19.75 percent.
- Pu-239. This is present in very small quantities [Security-Related Information] in the recycled target solution.

- Uranium-233 (U-233). This is present in very small quantities [Security-Related Information] in the both the recycled target solution and the recycled uranium oxide.

The equipment and piping in the TSPS that contain uranium in any form are criticality-safe by geometry.

A more detailed description of the TSPS can be found in Chapter 9.

4b.1.3.2 Molybdenum Extraction and Purification System

4b.1.3.2.1 Process Functions

- Receive irradiated target solution.
- [Proprietary Information]
- Extract Mo-99 from the irradiated target solution.
- Transfer post-extraction target solution to the recycle target solution tank or uranyl nitrate conversion tank.
- [Proprietary Information]
- Wash the Mo Extraction Column and [Proprietary Information] between steps to avoid cross-contamination and unwanted reactions.
- Purify the Mo-99.
- Transfer the purified Mo-99 solution to MIPS.

4b.1.3.2.2 Safety Functions

Prevent inadvertent criticality through inherently-safe design of equipment.

4b.1.3.2.3 Process Description

The MEPS extracts the Mo from the irradiated uranyl sulfate and prepares a concentrated form of Mo. This Mo concentrate is then purified to the final Mo product form and is sampled for product specification compliance prior to packaging and shipment.

MEPS operations are performed in two distinct phases:

a. Molybdenum extraction
b. Molybdenum purification

Each of these operations takes place in a separate, segregated enclosure within a single shielded hot cell (supercell).

4b.1.3.2.3.1 Molybdenum Extraction

The molybdenum extraction is a batch process. It is performed in a number of steps.

a. Irradiated uranyl sulfate [Security-Related Information] passes through an adsorption column (1-MEPS-01T-A/B/C) packed with [Proprietary Information]. The adsorption column is approximately [Proprietary Information] in volume. The [Proprietary Information] nominally adsorbs [Proprietary Information]. The target solution, stripped of molybdenum, is transferred to the UNCS.
b. Wash the column with 10 column volumes of 0.1 M H_2SO_4, [Proprietary Information], in upflow, to remove the residual uranyl sulfate solution. This is then transferred to the UNCS.
c. Wash the column with 10 column volumes of 2.5 percent potassium permanganate ($KMnO_4$) solution, [Proprietary Information], in upflow.
d. Elute the Mo with 30 column volumes of 0.1 M sodium hydroxide (NaOH), [Proprietary Information], in downflow.

[Proprietary Information] It is then evaporated to dryness in a rotary evaporator, then re-dissolved in 3.4 fluid ounces (fl. oz.) (100 milliliters [mL]) of 1 M HNO_3. The Mo is again evaporated until dry and re-dissolved in 2.7 fl. oz. (80 mL) of 1 M HNO_3.

The molybdenum extraction column is replaced after each TSV batch is processed. This ensures predictability and repeatability of processing operations. Spent extraction columns are disposed of as solid waste.

[Proprietary Information]

A more detailed description of the molybdenum extraction can be found in Subsection 4b.3.1.

4b.1.3.2.3.2 Molybdenum Purification

The purification process is a series of steps which purifies the 2.7 fl. oz. (80 mL) of Mo-99 solution produced by the molybdenum extraction process. This process removes impurities through small-scale additions of reagents, precipitation, filtration, and boiling.

The total additions and wastes to and from the purification process are expected to be less than 17 fl. oz. (500 mL), excluding the glassware.

The purification process is performed using laboratory glassware, mounted on a rack. The purification glassware is replaced after each batch of Mo is run through it. Spent glassware is disposed of as solid radioactive waste.

A more detailed description of the purification process can be found in Subsection 4b.3.1.

4b.1.3.3 Uranyl Nitrate Conversion System

4b.1.3.3.1 Process Functions

Refer to Subsection 4b.4.1.1.1 for process functions of UNCS.

4b.1.3.3.2 Safety Functions

Prevent inadvertent criticality through inherently-safe design of equipment.

4b.1.3.3.3 Process Description

The UNCS is comprised of three subsystems, each with a specific processing function:

a. Uranyl nitrate conversion.
b. Target solution cleanup (UREX).
c. Thermal denitration.

4b.1.3.3.3.1 Uranyl Nitrate Conversion

The target solution is recycled and irradiated [Proprietary Information] times before being cleaned up by the UNCS. This results in [Proprietary Information] TSV batches of target solution being recycled to the TSVs each [Proprietary Information] and [Proprietary Information] TSV batches per [Proprietary Information] of target solution being converted to uranyl nitrate, processed through UREX, and converted to uranium oxide. This processing strategy is enabled by the relatively low concentration of fission products in the recycled target solution.

The target solution being recycled to the TSVs is staged in one of three recycle target solution tanks (1-UNCS-09T-A/B/C). From here it is pumped to one of the target solution hold tanks (1-TSPS-03T-A-H) in the TSPS.

The spent target solution being converted to uranyl nitrate is reacted with [Proprietary Information] then [Proprietary Information]. The uranyl nitrate conversion tank (1-UNCS-01T-A/B) is mixed by a pump-around recycle. The conversion reactions are given by:

[Proprietary Information] (Equation 4b.1-2)

[Proprietary Information] (Equation 4b.1-3)

[Proprietary Information] is added to the batch solution first, as a saturated solution, and is mixed for 1 hour at 194 to 212°F (90 to 100°C). A [Proprietary Information], is then added and mixed for 1 hour at 194 to 212°F (90 to 100°C). The process temperature is achieved by recirculating the tank contents through an electrical heater. After the reaction steps are complete, the tank contents are cooled to 122°F (50°C) by recirculating them through a chilled water cooler. [Proprietary Information] both have low [Proprietary Information] Excess [Proprietary Information] remain dissolved in the uranyl nitrate solution. The reaction product slurry is pumped to a centrifuge, where the precipitated sulfates are separated from the uranyl

nitrate solution. The sulfate solids concentration at the centrifuge discharge is assumed to be 50 percent by weight. The solid product in the centrifuge is washed to achieve an overall uranium recovery of 99.75 percent. The filtrate is transferred to the UREX feed tank (1-UNCS-02T) where nitric acid is added to adjust the pH before being transferred to the target solution cleanup subsystem.

4b.1.3.3.3.2 Target Solution Cleanup

The target solution cleanup system uses a solvent extraction process to separate uranium from fission products and transuranics with a very high specificity. The solvent extraction process is known as UREX. A variation of this process, plutonium uranium extraction (PUREX), has been used for many years in reprocessing facilities. The UREX process differs in that the plutonium is not extracted with the uranium, and remains with the fission products in the raffinate.

The UREX solvent is 30 percent (by volume) tri-butyl phosphate (TBP) in an n-dodecane carrier. The solvent extraction contacting and separation are performed in centrifugal contactors, arranged in series.

The UREX system is comprised of four distinct sections:

a. Extraction, in which the uranium in the uranyl nitrate feed is transferred from the aqueous to the solvent phase.
b. Scrubbing, in which impurities in the solvent, carried over from the extraction section, are returned to the aqueous phase.
c. Stripping, in which the uranium is transferred from the solvent to the aqueous phase.
d. Diluent washing, in which the uranyl nitrate product is washed with dodecane to remove organic impurities.

Radioisotopes, except the uranium and some technetium and iodine, remain in the aqueous phase in the extraction contactors and are discharged from the system as the aqueous raffinate. They are then transferred to the RLWE system. The solvent product from the extraction section is washed with a scrub solution of 0.3 M HNO_3 and 0.1 M acetohydroxamic acid (AHA) in water. The scrub solution removes the plutonium from the solvent phase and washes out entrained aqueous impurities carried over from the extraction section. The aqueous product from the scrub contactors is combined with the aqueous feed to the extraction section. The uranium-rich solvent is then stripped of the uranium, technetium and iodine by 0.01 M nitric acid. The uranium-rich HNO_3 solution is routed to an anion exchange column, which removes the technetium and iodine from the stream. When the anion exchange resin is loaded, it is replaced and the spent column is routed to the solid radioactive waste packaging (SRWP) system. The uranyl nitrate product from the anion exchange column is contacted with n-dodecane solvent to remove solvent degradation products and TBP from the aqueous stream. The uranyl nitrate stream from the diluent wash feeds the TDN process.

The solvent within the system is replaced as necessary to eliminate the need for a solvent washing section. Solvent replacement is assumed to be very infrequent because the solvent is exposed to a relatively low radiation field.

4b.1.3.3.3.3 Thermal Denitration

The TDN system first concentrates the uranyl nitrate solution in an evaporator to 8.34 lbU/gal (1000 gU/L). The evaporator is a thin-film design. A recycle loop allows multiple passes through the evaporator to achieve the necessary concentration.

The concentrated uranyl nitrate from the evaporator is then sprayed into the thermal denitrator, which is a fluidized bed reactor. The thermal denitrator is fluidized by air at approximately 2.6 feet per second (ft/sec) (0.79 meters per second [m/s]). The air is pre-heated by an electrical heater. The reaction in the thermal denitrator is as follows:

$$UO_2(NO_3)_2 \cdot 6H_2O \rightarrow UO_x + NO_2 + NO + O_2 + 6H_2O \quad \Delta H = 595 \text{ kJ/mol U} \quad \text{(Equation 4b.1-4)}$$

The thermal denitrator operating temperature is 572°F (300°C). The pressure at the bottom of the thermal denitrator is approximately 7 pounds per square inch gage (psig) (47.6 kPa) and the pressure at the top is approximately 5 psig (34.0 kPa). The TDN reaction is highly endothermic, so the thermal denitrator is heated using clamshell heaters around the denitrator walls and/or bayonet heaters within the bed. The ursanium oxide generated in the thermal denitrator is loaded into cans to be recycled into target solution in the TSPS.

The SNM within the UNCS system consists of:

- U-235. This is in the form of uranyl sulfate, uranyl nitrate, and uranium oxide. The maximum inventory of uranium (U-235) in the UNCS at any given time is [Security-Related Information]. This is divided between a number of process units and storage tanks and represents [Proprietary Information] TSV batches. The uranium is in the form of uranyl sulfate, uranyl nitrate and uranium oxide, depending on its specific location in the process.
- Pu-239. This is present in very small quantities (maximum inventory [Security-Related Information]) in the target solution and uranyl nitrate being processed.
- U-233. This is present in very small quantities (maximum inventory [Security-Related Information][Proprietary Information]) in the target solution, uranyl nitrate and uranium oxide being processed. The uranium is in the form of uranyl sulfate, uranyl nitrate, and uranium oxide, depending on its specific location in the process.

A more detailed description of the UNCS can be found in Subsection 4b.4.1.

4b.1.3.4 Noble Gas Removal System

4b.1.3.4.1 Process Functions

Refer to Subsection 9b.6.2.1.1.

4b.1.3.4.2 Safety Functions

Refer to Subsection 9b.6.2.1.2.

4b.1.3.4.3 Process Description

A detailed description of the NGRS can be found in Subsection 9b.6.2.

4b.1.3.5 Process Vessel Vent System

4b.1.3.5.1 Process Functions

Refer to Subsection 9b.6.1.1.1.

4b.1.3.5.2 Safety Functions

Refer to Subsection 9b.6.1.1.2.

4b.1.3.5.3 Process Description

The PVVS treats off-gas from the vessels that handle the main process fluids. These include:

- Uranyl sulfate preparation tank (1-TSPS-01T-A/B).
- Uranium metal dissolution tank (1-TSPS-02T-A/B).
- Target solution hold tank (1-TSPS-03T-A-H).
- Mo extraction feed tank (1-MEPS-04T).
- Mo eluate hold tank (1-MEPS-02T-A/B/C).
- Uranyl nitrate conversion tank (1-UNCS-01T-A/B).
- UREX feed tank (1-UNCS-02T-A/B).
- Recycle target solution tank (1-UNCS-09T-A/B/C).
- Solvent hold tank (1-UNCS-04T).
- Raffinate hold tank (1-UNCS-05T-A/B).
- Recycle UN hold tank (1-UNCS-06T).
- Thermal denitrator (1-UNCS-08T).
- Liquid waste storage tank (1-RLWS-01T-A/B).
- Liquid waste condensate collection tank (1-RLWE-01T-A/B).
- Liquid radioactive waste collection tank (1-RLWE-02T-A/B).

The primary contaminants in the PVVS system are:

- Acid gases, both nitric and sulfuric
- Caustic gases
- Iodine
- Noble gases

The PVVS system design includes three primary unit operations:

- Acid gas scrubber scrubs acid gases from the PVVS with sodium hydroxide (caustic).
- Caustic scrubbing also removes some iodine species, most notably elemental iodine (I_2).
- The caustic recirculation loop includes a cooler to remove the heat of reaction for absorbed acid gases.

There is no significant quantity of SNM within the PVVS system. Therefore, no criticality controls or safeguards are required.

A more detailed description of the PVVS is provided in Subsection 9b.6.1.

4b.1.3.6 Radioactive Liquid Waste Evaporation and Immobilization

The RLWE system receives aqueous liquid wastes from a number of systems within the RCA. The aqueous wastes are concentrated by evaporation, to reduce their volume. They are then immobilized in Portland cement to comply with the relevant storage, shipping, and disposal requirements.

There is no significant quantity of SNM within the RLWE system. Therefore, no criticality controls or safeguards are required.

A more detailed description of the RLWE system is provided in Subsection 9b.7.3.

4b.1.3.7 Aqueous Radioactive Liquid Waste Storage

The RLWS provides surge capacity for aqueous liquid wastes from the RCA prior to evaporation in the RLWE. This surge capacity performs several functions:

- Allows short-lived radioisotopes to decay.
- Reduces fluctuations in the aqueous liquid waste composition.
- Reduces fluctuations in the aqueous liquid waste flowrate.

There is no significant quantity of SNM within the RLWS system. Therefore, no criticality controls or safeguards are required.

A more detailed description of the RLWS is provided in Subsection 9b.7.4.

4b.1.3.8 Organic Liquid Waste Storage and Export

The OLWS system is used to load, store, and export organic wastes from the RCA. The primary organic waste is spent solvent from the UREX process.

There is no significant quantity of SNM within the OLWS system. Therefore, no criticality controls or safeguards are required.

A more detailed description of this system is provided in Subsection 9b.7.19.

4b.1.3.9 Molybdenum Isotope Product Packaging System

4b.1.3.9.1 Process Functions

Refer to Subsection 9b.7.1.5 for process functions.

4b.1.3.9.2 Safety Functions

There are no safety functions of the MIPS.

4b.1.3.9.3 Process Description

The MIPS receives concentrated Mo-99 product solution from the MEPS.

Product packaging is accomplished in the molybdenum packaging section of the shielded supercell and accommodates packaging requirements for three different product delivery casks. Product assay and quality control may be performed in this cell. Lot number and expiration date are recorded on final product containers. Final molybdenum product containers are transferred into the shipping casks for delivery.

There is no significant quantity of SNM within the MIPS system. Therefore, no criticality controls or safeguards are required.

A more detailed description of the MIPS can is provided in Subsection 9b.7.1.

4b.1.3.10 Radioactive Drain System

4b.1.3.10.1 Process Functions

Refer to Subsection 9b.7.6.1.1 for process functions.

4b.1.3.10.2 Safety Functions

Refer to Subsection 9b.7.6.2.1 for safety functions.

4b.1.3.10.3 Process Description

The RDS routes liquids from the cell sumps and tank vaults to a collection tank in the basement of the facility. The system includes the collection tank, which is criticality-safe by geometry.

A more detailed description of the RDS is provided in Subsection 9b.7.6.

4b.2 RADIOISOTOPE PRODUCTION FACILITY BIOLOGICAL SHIELD

4b.2.1 INTRODUCTION

The production facility biological shield (PFBS) provides a barrier to protect SHINE facility personnel, members of the public, and various components and equipment of the SHINE facility by reducing radiation exposure. The radioisotope production facility receives the irradiated target solution from the IU cell and distributes the target solution to various downstream processes. The target solution has a fission product activity that is defined in Chapter 11. The major areas outside of the IU cell that the target solution and by-product material occupies are:

- Supercell (for Mo extraction, purification, and packaging)
- Process tanks
- Pipe chases
- Waste processing cells
- UREX cell
- TDN cell
- Pump room hot cell
- PVVS cell
- NGRS shielded cell

A description of radiation source locations and source term characterizations can be found in Chapter 11.

The ventilation for these areas is described in Section 9a2.1.1.

4b.2.2 BIOLOGICAL SHIELD DESIGN BASIS

4b.2.2.1 Materials

The design bases for the materials included in the biological shield design are:

- The dose reduction by the biological shielding is sufficient to meet the radiation exposure goals defined in Chapter 11.

- All materials used for biological shielding meet or exceed the requirements of ANSI/ ANS-6.4.2-2006, Specification for Radiation Shielding Materials (ANSI/ANS, 2006b).

- The design and construction of the concrete portions of the biological shield conforms to Regulatory Guide 1.69-2009.

4b.2.2.2 Geometry and Configuration

Bounding calculations for the biological shield in the RPF have been performed. Detailed calculations will be performed for locations throughout the RPF, and figures will be provided in the FSAR showing PFBS configurations.

4b.2.2.3 Loss of Shield Integrity

The biological shield and supporting structures are designed and constructed to remain intact during normal operations as well as during and following design basis accidents.

4b.2.2.4 Unrestricted Environment

No significant neutron flux exists in the RPF that could result in activation of groundwater or soils in the RPF.

4b.2.3 SHIELD MATERIALS

The RPF uses two primary materials in different configurations to assemble the biological shield and meet the radiation exposure goals defined in Chapter 11. The materials credited for biological shielding are steel and reinforced concrete.

4b.2.3.1 Shielding Calculations

Calculations are performed with the software package named MCNP (Monte-Carlo N-Particle Transport Code). MCNP is developed and validated by LANL and distributed by the RSICC at ORNL. MCNP uses a Monte-Carlo based particle (neutrons and photons) transport method to generate a set of particle tracks through a model of the facility geometry. The Monte Carlo method generates a statistical set of results for individual particles transported through the geometry. Enough particles are simulated to obtain statistically-significant results. The MCNP calculation methodology does not require use of shielding coefficients. The shielding coefficients will be listed in the FSAR.

4b.2.3.2 Radiation Damage

4b.2.3.2.1 Concrete

According to the ANSI/ANS-6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, nuclear heating in concrete can be neglected if the incident energy fluxes are less than 10^{10} MeV/cm^2-sec (ANSI/ANS, 2006a). No substantial neutron flux is anticipated in the RPF. The magnitude of nuclear heating effects caused by gamma fluxes present in the RPF will be presented in the FSAR.

ANSI/ANS-6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, states that compressive strength and modulus of elasticity of concrete are degraded if the concrete is exposed to a neutron fluence greater than 10^{19} n/cm^2 (ANSI/ANS, 2006a). No substantial neutron flux is anticipated outside of the IU cell and subsequently no neutron damage to concrete occurs.

ANSI/ANS-6.4-2006, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, states that compressive strength and modulus of elasticity of concrete are degraded if the concrete is exposed to an integrated dose of gamma radiation exceeding 10^{10} rads (ANSI/ANS, 2006a). Fission product and actinide decay of the target solution and its derivatives occur throughout the production facility. Using MCNP, analysis of this condition will

be performed once final target solution parameters and process geometries are confirmed following detailed design.

Due to the absence of significant neutron fluxes, no significant activation of structural components within the production facility biological shield will occur.

4b.2.3.2.2 Steel

Criticality mitigation is a design feature of the facility and thus no substantial neutron flux is anticipated outside of the IU cell. Subsequently, no neutron damage to steel in the production facility occurs.

With regard to gamma radiation, no substantial effect on steel integrity is expected.

4b.2.3.3 Radiation Streaming

The biological shield requires a number of penetrations, inserts, and other features where the bulk shielding materials are reduced in thickness, or where the materials used in the penetration are less dense than the surrounding bulk material. Each such penetration is designed with well-demonstrated techniques of non-linear paths, supplemental shielding, location in areas of low incident radiation, and other methods to reduce streaming and leakage to ensure 10 CFR 20 limits are met.

4b.2.4 ANALYSIS

Analysis is performed to:

- Give detailed results of both neutron (when applicable) and gamma-ray dose rates at locations that could be occupied as well as to the unrestricted environment.
- Include shield penetrations and voids, such as beamports, thermal columns, and irradiation rooms or vaults, as well as the shielding of piping and other components that could contain radioactive materials or allow radiation streaming.
- Determine extent of radiation effect on shielding materials (i.e., heating and activation).

4b.2.4.1 Methodology

4b.2.4.1.1 Concrete Radiation Shielding Minimum Thickness – Radiation Shielding Requirements

The minimum thickness of concrete radiation shields, based on radiation shielding requirements, is determined using the following approach:

a. Use ANSI/ANS-6.4-2006, Chapters 6, 7, and 8, as an overview of the historic calculation methodology for concrete radiation shields (ANSI/ANS, 2006a).
b. Use Monte Carlo techniques for radiation shielding calculations.
c. Use the latest version of the software for radiation shielding calculations (i.e., MCNP5). The concrete composition input parameters for the MCNP5 calculations correspond to the concrete used for the radiation shields.

4b.2.4.1.2 Concrete Radiation Shielding Minimum Thickness – Structural Requirements and Other Structural Dimensions and Reinforcement Requirements

The minimum thickness of concrete radiation shields, based on structural requirements, other structural dimensions, and reinforcement requirements, is determined in accordance with the provisions of ACI 349-06, and ACI 349.1R-07 (ACI, 2007a; ACI, 2007b), for applicable normal loads, severe and extreme environmental loads, and abnormal loads, as defined in Section 9.1 of ACI 349-06.

4b.2.4.1.3 Concrete Radiation Shielding – Final Minimum Thickness

The final minimum thickness of the concrete shield structure will be based on radiation shielding requirements and will not be driven by the structural requirements.

4b.2.4.1.4 Load and Strength Reduction Factors

Load and strength reduction factors for the structural design of concrete shield structures and related members shall be based on those prescribed in ACI 349-06, Sections 9.2 and 9.3, respectively (ACI, 2007a).

4b.2.4.1.5 Design of Concrete for Shielding Structures

The design of the concrete for shielding structures, including materials selection, durability requirements, quality control, mixing, placement, formwork, embedded pipes, construction joints, reinforcement, analysis, and design, shall conform to provisions outlined in Chapters 3 through 8 of ACI 349-06, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary (ACI, 2007a).

4b.2.4.2 Exceptions for Use of ACI 349-06 and ACI 349.1R-2007

ACI 349-06, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, Section 1.2.2, states that input and output data shall be retained as documentation when software is used for the calculation. The software itself and other related documentation shall be retained as well. It is not required that the software be updated regularly.

SHINE does not utilize the following sections of ACI 349-06:

- Section 3.3.1: The exception portion of the section is not endorsed.
- Section 3.3.2: “These limitations may be waived if, in the judgment of the engineer, workability and methods of consolidation are such that concrete can be placed without honeycombs or voids.”
- Section 5.4.1: “If data required by 5.3 are not available, concrete proportions shall be based on other experience or information, if approved by the engineer. The required average compressive strength f'_{cr} of concrete produced with materials similar to those proposed for use shall be at least 1200 psi greater than f'_c. This alternative shall not be used if f'_c is greater than 5000 psi.”

- Section 5.6.2.3: “When total quantity of a given class of concrete is less than 50 yd^3, strength tests may be waived by the engineer if the engineer has been provided adequate evidence of satisfactory strength.” Instead, the provisions of Regulatory Position 5 of Regulatory Guide 1.142 for strength testing are utilized.

- Section 7.10.3: “It shall be permitted to waive the lateral reinforcement requirements of 7.10, 10.16, and 18.11 where tests and structural analysis show adequate strength and feasibility of construction.”

4b.2.5 TEST PROGRAM

ANSI/ANS-6.3.1-1987 (R2007) (ANSI/ANS, 2007), Program for Testing Radiation Shields in Light Water Reactors (LWR), is used as a guide in the development of a test program to be used in evaluating biological radiation shielding in the SHINE facility under normal operating conditions, including anticipated operational occurrences.

4b.2.6 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the RPF biological shield are provided in Chapter 14.

4b.3 RADIOISOTOPE EXTRACTION SYSTEM

4b.3.1 RADIOISOTOPE EXTRACTION PROCESS DESCRIPTION

4b.3.1.1 Process Functions

The following are the process functions of the MEPS:

- Receive irradiated target solution.
- [Proprietary Information].
- Extract Mo-99 from the irradiated target solution.
- Transfer post-extraction target solution to the recycle target solution tank or uranyl nitrate conversion tank.
- [Proprietary Information]
- Wash the Mo extraction column and [Proprietary Information] between steps to avoid cross-contamination and unwanted reactions.
- Purify the Mo-99.
- Transfer the purified Mo-99 solution to the MIPS.

4b.3.1.2 Safety Functions

The safety function of the MEPS is to prevent inadvertent criticality through inherently-safe design of equipment.

4b.3.1.3 Primary System Interfaces

For a list of system interfaces, refer to Table 4b.3-1.

4b.3.1.4 Process Sequence

This process includes the extraction, concentration, and purification steps of the MEPS. See Figure 4b.3-1 for the process flow diagram (PFD) related to the extraction and concentration steps. See Figure 4b.3-2 for the PFD related to the purification step.

4b.3.1.4.1 Radioisotope Extraction and Concentration Process Sequence

The following steps are for the purpose of extracting and concentrating the radioisotope (Mo-99):

a. One TSV volume of irradiated target solution is transferred through the MEPS in each processing cycle. The target solution is pumped from the TSV dump tank (1-SCAS-01T) to the MEPS hot cell by pump(s) in the IU cell.

b. The irradiated target solution is transferred through the molybdenum extraction [Proprietary Information] (1-MEPS-01A), and through the molybdenum extraction column (1-MEPS-01T) in the up-flow direction. The [Proprietary Information] irradiated target solution and subsequent column washes to a pre-determined [Proprietary Information]. The molybdenum extraction column contains a fixed bed of [Proprietary Information]. The molybdenum extraction column is replaced for each batch of target solution. The fresh [Proprietary Information] adsorbs [Proprietary Information] percent

of the Mo-99. Extraction of the molybdenum from the irradiated target solution takes approximately [Proprietary Information]. In addition to the molybdenum extracted, the [Proprietary Information] adsorbs a portion of the iodine.

c. After the irradiated target solution passes through the extraction column, it is transferred to the UNCS.

d. All of the column washes (described in the subsequent steps) are prepared and metered outside the hot cell. The column washes pass through a manifold exterior to the hot cell, through a single hot cell penetration, [Proprietary Information].

e. The column is washed with sulfuric acid in the up-flow direction to remove the residual target solution. The sulfuric acid is transferred to the uranyl nitrate conversion tank (1-UNCS-01T) to minimize uranium loss.

f. The column is then washed with potassium permanganate in the up-flow direction to wash out the sulfuric acid. The potassium permanganate wash is drained to the liquid waste storage system.

g. The inlet and outlet valves on the extraction column switch position to allow the next solution to flow in the down-flow direction. The extraction column is eluted in the down-flow direction with sodium hydroxide to extract the Mo-99, and is transferred to the molybdenum eluate hold tank (1-MEPS-02T).

h. Finally, the extraction column is washed with deionized water in the up-flow direction. The water is drained to the liquid waste storage system.

i. The collected molybdenum-bearing solution is re-acidified with concentrated nitric acid.

j. The molybdenum-bearing solution [Proprietary Information]

k. [Proprietary Information]

l. The molybdenum-rich solution is dried completely in the rotary evaporator (1-MEPS-02A). The molybdenum is re-dissolved in nitric acid and dried again in the same rotary evaporator. The overhead streams from the rotary evaporator are condensed, cooled, and drained to the RLWS.

m. The Mo-99 is dissolved again in nitric acid and drained into a container. The rotary evaporator is rinsed with nitric acid and drained again into the container. This container is transferred to the molybdenum purification portion of the supercell.

n. The Mo extraction column (1-MEPS-01T) and the [Proprietary Information] are disconnected and removed from service. The extraction column and [Proprietary Information] are placed on a decay storage rack for 2 weeks. After 2 weeks, the extraction column and [Proprietary Information] are placed in a double door for leaktight transfer (DPTE) waste transfer system to be transferred to the SRWP.

o. A new Mo extraction column and [Proprietary Information] are installed to replace 1-MEPS-01T and [Proprietary Information].

4b.3.1.4.2 Radioisotope Purification Process Sequence

The following steps are for the purpose of purifying the radioisotope (Mo-99). The purification process is anticipated to take 3 to 4 hours to complete in a hot cell.

The purification process purifies the Mo-99 through additions of small quantities of chemicals, filtration, and other laboratory techniques. The following steps describe the current purification process. The details below are intended to provide an overview of the techniques that are employed. The details are not intended to preclude process enhancements that may occur.

a. The molybdenum concentrate is received from the extraction area of the hot cell and loaded into a flask (1-MEPS-02Z). In this flask, sodium iodide carrier is added. The next addition is silver nitrate, which is agitated and left to rest. Then hydrochloric acid is added.

b. This solution is filtered through a filter assembly (1-MEPS-01F), and drained into another flask (1-MEPS-03Z). The filter assembly (1-MEPS-01F) is rinsed with nitric acid, which is collected in the next flask (1-MEPS-03Z).

c. In the second flask (1-MEPS-03Z), potassium permanganate is added until a deep pink color holds. The next addition is rhodium carrier, followed by ruthenium carrier. Alpha-benzoin oxime in sodium hydroxide is added and manually agitated. This is followed by a rest while a precipitate forms.

d. This solution and precipitate are filtered through a fritted glass column (1-MEPS-04Z). The fritted glass column should be prefilled with pyrogen-free water and glass beads prior to filtration. The solution is collected in a waste bottle. If any precipitate passes into this waste bottle, the waste bottle should be re-filtered over the fritted glass column. The second flask (1-MEPS-03Z) and the fritted glass column are washed with nitric acid, which is also collected in the waste bottle. The waste bottle is changed, and the nitric acid wash is repeated.

e. The fritted glass column is washed with nitric acid. The column is filled with the nitric acid, mixed well, and then drained to an acid waste bottle each time.

f. The fritted glass column is filled with sodium hydroxide with hydrogen peroxide. The column is then heated with forced hot air until the solution boils, in order to dissolve the precipitate. The column is removed from the heat and allowed to cool, then drained into a new bottle (1-MEPS-06Z). This step is repeated with sodium hydroxide and hydrogen peroxide. The fritted glass column is rinsed with sodium hydroxide.

g. A small sample is taken from the bottle (1-MEPS-03Z). This sample is transported to the FDA lab, and its activity is read to verify the Mo-99 is present.

h. The solution is purified by filtration through a column packed with silver-coated charcoal. The column is washed with sodium hydroxide, which is collected in a bottle with the filtrate. A small sample is taken from this bottle. The sample is transported to the FDA lab, and the activity level is monitored to verify the Mo-99 did not become trapped on the column.

i. Sodium iodide carrier is added, followed by silver nitrate in nitric acid. This product is filtered through a column packed with a combination silver-coated charcoal and hydrous zirconium-oxide, and activated charcoal. This column is washed with sodium hydroxide.

j. This solution may be filtered through a filter into a final product bottle. A small sample is taken from the final product bottle. The sample is transported to the FDA lab, and the activity is read and recorded. The final product bottle is transferred to the MIPS area of the supercell.

4b.3.1.4.3 Process Equipment

The following is a list of process equipment associated with the Mo-99 extraction system.

a. [Proprietary Information]

    1. [Proprietary Information]
    2. [Proprietary Information]
    3. [Proprietary Information]
    4. [Proprietary Information]
    5. [Proprietary Information]
    6. Design/fabrication code: ASME B31.3 and ASME BPVC Section VIII (ASME, 2012; ASME, 2011)

b. 1-MEPS-01P – Mo eluate pump

    1. Type: positive-displacement
    2. Power: 0.25 horsepower (hp) (186 W)
    3. Flow rate: <10 gal/hr (38 L/h)
    4. Nominal fluid temperature: [Proprietary Information]
    5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

c. 1-MEPS-02P – Cintichem vacuum pump

    1. Details to be provided in the FSAR.

d. 1-MEPS-01T – Molybdenum extraction column

    1. Size: [Proprietary Information]
    2. Normal operating pressure: atmospheric
    3. Packing: [Proprietary Information]
    4. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME, 2011)

e. 1-MEPS-02T – Mo eluate hold tank

1. Size: 10 gal. (38 L)
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME, 2011)

f. [Proprietary Information]

1. [Proprietary Information]
2. [Proprietary Information]
3. [Proprietary Information]
4. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME, 2011)

g. 1-MEPS-01Z – Mo eluate rotovap

1. Size: 1.3 gal. (5 L) flask size
2. Heat duty: <3000 W (10,236 Btu/hr)
3. Condenser type: surface condenser
4. Condenser inlet temperature: 60 to 212°F (16 to 100°C)
5. Condenser outlet temperature: 60 to 100°F (16 to 38°C)
6 Cooling medium: chilled water

4b.3.2 PHYSICAL PROPERTIES

4b.3.2.1 Target Solution Properties

This subsection describes the approximate physical, chemical, and radiological properties of the irradiated target solution, which is received from the TSV dump tank (1-SCAS-01T).

4b.3.2.1.1 Physical and Chemical Properties

Table 4a2.2-1 shows the chemical properties of the target solution.

The physical and chemical form and inventory of SNM in the RPF is provided in Table 4b.4-1.

4b.3.2.1.2 Radioisotope Properties

The radioisotope extraction system receives target solution after 5.5 days of irradiation and approximately [Proprietary Information] hours of decay. The estimated total radioactive inventory in one TSV batch is less than [Security-Related Information][Proprietary Information] when it is transferred to the MEPS (see Subsection 11.1.1). Note that the total radioactive inventory of irradiated target solution within a MEPS portion of the supercell is significantly less, since the supercell never contains an entire TSV batch.

4b.3.2.2 Hazardous Chemicals

Refer to Table 4b.3-2 and Table 4b.3-3 for the chemical inventories associated with the MEPS. In addition to the reagents used, the MEPS receives target solution. For the target solution preparation process, refer to Section 9b.2. Any potential off-gases formed during MEPS processing are evolved within the supercell, isolated from workers.

4b.3.3 CRITICALITY CONTROL FEATURES

The MEPS prevents inadvertent criticality through inherently-safe geometrical design of equipment.

4b.3.4 SHIELDING AND RADIOLOGICAL PROTECTION

The MEPS processes described above are performed in shielded hot cells, which keeps worker exposure to radiation within the regulatory limits of 10 CFR 20.1201 and 10 CFR 20.1301. Refer to Chapter 11 for the radiation exposure goals, and Subsection 4b.2 for further detail of shielding requirements.

The processes are remotely, manually controlled, and performed with tele-manipulators, with minimal automated sequences. Radiation monitors and alarms are used to monitor release of radiological materials, monitor high background gamma dose levels, and to detect criticality events.

Piping that contains potentially-radiological material is routed through shielded pipe chases to limit the worker exposure to radiation. Tanks within the MEPS are inside shielded hot cells, so additional tank shielding is not required.

4b.3.5 TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with radioisotope extraction are provided in Chapter 14.

Table 4b.3-1 Mo Extraction and Purification System Interfaces

Interfacing System	Interface Description
Subcritical Assembly System (SCAS)	The MEPS interfaces with the SCAS at one location. The irradiated target solution is pumped from the SCAS to the supercell.
Process Vessel Vent System (PVVS)	The MEPS interfaces with the PVVS at two locations, the vent lines from 1-MEPS-02T and 1-MEPS-04T.
Radioisotope Process Facility Cooling System (RPCS)	The MEPS interfaces with the RPCS. The RPCS provides cooling water to Mo eluate rotary evaporator (1-MEPS-01Z).
Normal Electrical Power Supply System (NPSS)	Electrical power is provided to the extraction column [Proprietary Information], the Mo eluate rotary evaporator (1-MEPS-01Z), and ancillary equipment.
Aqueous Radioactive Liquid Waste Storage System (RLWS)	The MEPS interfaces with the RLWS. A hot cell drain line is provided from MEPS to RLWS.
Uranyl Nitrate Conversion System (UNCS)	The MEPS interfaces with the UNCS. The post-extraction target solution is routed to UNCS from MEPS.
Solid Waste Storage System	The MEPS interfaces with the solid waste storage system at one location: the solid waste shielded 55-gallon drum interface point.
Molybdenum Isotope Product Packaging System (MIPS)	The MEPS interfaces with the MIPS by transferring a container with the purified Mo-99 from the MEPS area of the supercell to the MIPS area of the supercell.

Table 4b.3-2 Molybdenum Extraction Chemical Inventory (Approximate)

Chemical Name (CAS Number)	Chemical Concentration	Chemical Inventory in Mo Extraction Portion of Supercell
Ammonium hydroxide (1336-21-6)	1.0 M	2.5 L
Nitric acid (7697-37-2)	10 M	5 L
Potassium permanganate (7722-64-7)	2.5%	12 L
Sodium hydroxide (1310-73-2)	0.1 M	35 L
Sulfuric acid (7664-93-9)	0.1 M	12 L

Table 4b.3-3 Purification Chemical Inventory (Approximate)

Chemical Name (CAS Number)	Chemical Concentration	Chemical Inventory in Mo Purification Portion of Supercell
Alpha-benzoin oxime (441-38-3)	2 wt%	50 mL
Hydrochloric acid (7647-01-0)	1.0 M	5 mL
Hydrogen peroxide (7722-84-1)	1 wt%	20 mL
Molybdenum trioxide (1313-27-5)	10 mg/mL	2 mL
Nitric acid (7697-37-2)	0.1 M 1.0 M 4.0 M	300 mL 10 mL 20 mL
Potassium permanganate (7722-64-7)	2.5 wt%	100 mL
Potassium hexachlororuthenate (23013-82-3)	5 mg/mL	2 mL
Rhodium chloride (10049-07-7)	8 mg/mL	4 mL
Silver nitrate (7761-88-8)	10 wt%	4 mL
Sodium hydroxide (1310-73-2)	0.2 M 0.4 M	200 mL 70 mL
Sodium iodide (7681-82-5)	1 mg/mL	20 mL

4b.4 SPECIAL NUCLEAR MATERIAL PROCESSING AND STORAGE

The processing and storage of SNM is conducted in the production facility building and the waste staging and shipping building. SNM is used throughout the RCA in both unirradiated and irradiated forms for the production of medical isotopes.

Mo-99 is extracted from the irradiated SNM in the MEPS. Following Mo-99 extraction, the processing of irradiated SNM is performed in the UNCS. When cleanup of solution is required, the UNCS converts spent target solution in the form of uranyl sulfate into uranyl nitrate, separates the uranium from fission products and transuranic isotopes in the UREX process, and recovers uranium in the form of uranium oxide in the TDN process. The uranium oxide is loaded into cans and then returned to the TSPS. Irradiated SNM is stored in criticality-safe tanks between irradiation cycles. Subsection 4b.4.1 discusses the processing of irradiated SNM in greater detail.

The following are the main SNM processing steps:

- Extract Mo-99 from irradiated target solution.
- Store and transport irradiated target solution.
- Separate fission products and transuranic isotopes from spent target solution.
- Produce uranium oxide from clean uranyl nitrate.
- Dissolve uranium metal in nitric acid to form uranyl nitrate.
- Dissolve uranium oxide in sulfuric acid to form target solution.

The facility receives and stores new shipments of uranium metal, and stores uranium oxide, which is used to prepare unirradiated target solution. Uranium oxide is stored in uranium oxide storage cans and is transported from the TDN area (subsystem of the UNCS) to the TSPS area. Section 9b.2 discusses the preparation and storage of the target solution.

Shipments of SNM are received at the facility in solid form from a U.S. Department of Energy (DOE) supplier. The shipments consist of uranium metal enriched to 19.75 ± 0.2 percent U-235 (LEU). The SNM is transported in approved transport containers (a general-purpose Type B fissile material shipping container). The SNM is removed from the transport containers and stored in uranium metal storage cans in criticality-safe configuration within the uranium metal storage rack. The uranium metal is dissolved in nitric acid to form uranyl nitrate. The uranyl nitrate is processed through TDN to form uranium oxide, which is then transported to TSPS and used to prepare fresh target solution by dissolving the uranium oxide in sulfuric acid. Subsection 4b.4.2 provides more detail on the processing of unirradiated SNM.

The RPF contains SNM in multiple forms: uranium metal, uranium oxide, uranyl sulfate, uranyl nitrate, and aqueous and solidified plutonium. The SNM is enriched uranium and plutonium from the irradiation process. Refer to Table 4b.4-1 for the total inventory of SNM in the facility.

The SNM processing and storage systems prevent inadvertent criticality through inherently-safe design of tanks, process equipment, storage containers, and other components that may handle the SNM. A detailed description of the Criticality Safety Program is provided in Section 6b.3.

The processing and storage of irradiated SNM is performed in tank vaults (below-grade concrete enclosures providing shielding and confinement of hazardous materials) and shielded hot cells (controlled-environment work enclosures providing confinement of hazardous materials), gloveboxes, and fumehoods, which limits the exposure of personnel to radiation within the regulatory limits of 10 CFR 20.1201 and 10 CFR 20.1301.

4b.4.1 PROCESSING OF IRRADIATED SPECIAL NUCLEAR MATERIAL

4b.4.1.1 Process Description

4b.4.1.1.1 Process Functions

Once the target solution is determined to be unacceptable for reuse, UNCS performs cleanup of the solution. The UNCS converts spent target solution in the form of uranyl sulfate into uranyl nitrate, separates the uranium from fission products and transuranics in the UREX process, and recovers uranium in the form of uranium oxide in the thermal denitration process. There are three subsystems within the UNCS: uranyl nitrate preparation (UNP), UREX, and TDN. This subsection presents preliminary process functions for processing irradiated SNM. The UNCS also holds target solution prior to reuse in the irradiation facility.

The process functions of the UNP subsystem are the following:

- Receive and hold target solution (and sulfuric acid wash solution) from MEPS.
- In the event of fissile material transfer to the RDS, receive waste solution from RDS for fissile material recovery.
- Heat spent target solution to reaction temperature.
- Convert uranyl sulfate to uranyl nitrate.
- Cool reaction products.
- Separate uranyl nitrate from [Proprietary Information].
- Adjust nitric acid concentration of uranyl nitrate.
- Feed uranyl nitrate to the UREX process.
- Transfer recycled target solution to TSPS.

The process functions of the UREX subsystem are the following:

- Receive uranyl nitrate from UNP subsystem.
- Remove fission products and transuranics from the uranyl nitrate.
- Recover uranium from contactor holdup.
- Adsorb technetium and iodine isotopes from the cleaned uranyl nitrate.
- Wash the cleaned uranyl nitrate to remove excess tri-butyl phosphate.
- Transfer the raffinate (fission products and transuranics) to RLWS.
- Receive fresh uranyl nitrate from the TSPS.
- Feed clean uranyl nitrate to the evaporator in the TDN subsystem.

The process functions of the TDN subsystem are the following:

- Receive uranyl nitrate from the UREX subsystem.
- Concentrate the uranyl nitrate to 8.3 lbU/gal (1000 gU/L).
- Produce uranium oxide from the concentrated uranyl nitrate.

- Collect and package uranium oxide for transport to the TSPS.
- Cool and condense the off-gases from the evaporator and thermal denitrator.

4b.4.1.1.2 Safety Functions

Prevent inadvertent criticality through inherently-safe design of equipment.

4b.4.1.1.3 Primary System Interfaces

For a list of system interfaces, refer to Table 4b.4-2, Table 4b.4-3, and Table 4b.4-4. The stream numbers referred to in the interface descriptions can be found in Figure 4b.4-1, Figure 4b.4-2, and Figure 4b.4-3.

4b.4.1.1.4 Process Sequence

The process sequence for each UNCS subsystems are provided below. See Figure 4b.4-1 for the PFD of the UNP subsystem. See Figure 4b.4-2 for the PFD of the UREX subsystem. See Figure 4b.4-3 for the PFD of the TDN subsystem.

4b.4.1.1.4.1 Uranyl Nitrate Preparation Process Sequence

The following steps describe the UNP subsystem process sequence where uranyl sulfate in the spent target solution is converted into uranyl nitrate. The process is expected to take approximately five hours per UNCS batch.

a. Spent target solution is fed from the Mo extraction column of the MEPS hot cell to the uranyl nitrate conversion tank (1-UNCS-01T). Pump 1-SCAS-01P in the SCAS transports the irradiated target solution through the MEPS and into the uranyl nitrate conversion tank.

b. The sulfuric acid washes of the Mo extraction column [Proprietary Information] in the MEPS hot cell are also fed to the uranyl nitrate conversion tank. One spent target solution batch plus the weekly sulfuric acid washes constitutes one UNCS batch.

c. The spent target solution/sulfuric acid mixture is converted to uranyl nitrate by the additions of [Proprietary Information] followed by [Proprietary Information]. The sulfuric acid is similarly converted into nitric acid. The uranyl nitrate conversion tank is agitated by a pumped recycle loop to promote mixing and reaction completion at approximately 5 volume changes per hour. An electric heating element (1-UNCS-07A) on the recycle loop maintains the reaction temperature. The reactions are given by:

   [Proprietary Information] (Equation 4b.4-1)

d. [Proprietary Information] The reaction products are cooled to approximately 122°F (50°C) by a process chilled water cooler (1-UNCS-06A) on the recycle loop.

e. The reaction product slurry is pumped to the uranyl nitrate centrifuge (1-UNCS-01F) in a semi-continuous process to separate the precipitated sulfates from the uranyl nitrate solution. The sulfates solids concentration at the centrifuge discharge is assumed to be

50 percent by weight. The liquid products gravity drain to the UREX feed tank (1-UNCS-02T), and solid salts remain in the centrifuge.

f. After the reaction products have been processed, the solid salts in the centrifuge are washed with deionized (DI) water to minimize uranium losses. The DI water wash is also collected in the UREX feed tank. The uranium recovery for this separation process is approximately 99.75 percent. Solid salts are discharged from the centrifuge and removed to the SRWP system in 55-gallon drums.

g. The nitric acid concentration of the uranyl nitrate in the UREX feed tank is increased to 0.1 M nitric acid by adding concentrated nitric acid. The tank contents are well-mixed by operating a recycle loop on the tank with pump 1-UNCS-03P.

h. The pH-adjusted uranyl nitrate is fed to the UREX subsystem of the UNCS from the UREX feed tank. The flow rate is determined by the uranium extraction processing rates, to be defined in the UREX subsystem process sequence.

i. In addition to processing spent target solution, the UNP subsystem also processes recycled target solution. Recycled target solution from the MEPS is fed to the recycle target solution tank (1-UNCS-09T). [Proprietary Information] The recycled target solution is pumped to the TSPS target solution hold tank (1-TSPS-03T) to be irradiated again without undergoing cleanup.

j. The tanks are continuously vented to the PVVS.

4b.4.1.1.4.2 Uranium Extraction Process Sequence

The following steps describe the UREX subsystem process sequence. The UREX process is a solvent extraction process in which the uranium is isolated from fission products and transuranics by extraction into an organic solvent phase followed by stripping back into an aqueous phase. There are three sections of centrifugal contactors: extraction, scrub, and strip. The process is expected to take approximately six hours per UNCS batch.

a. A cold feed of 1 M nitric acid is fed to the UREX process to startup the contactors at the beginning of a new batch. The cold feed is a nitric acid solution similar in chemistry to uranyl nitrate that is used to startup the UREX contactors. The cold feed does not contain any SNM. Streams and contactors in the UREX process are running during this startup phase and the cold feed runs for about 30 minutes.

b. Once the contactors have achieved steady state separation, uranyl nitrate (hot feed) from 1-UNCS-02T of the UNP subsystem replaces the cold feed. A typical UNCS batch contains approximately [Security-Related Information] of uranyl nitrate, which requires approximately 4 hours to process.

c. In the extraction section, uranium plus some technetium and iodine are selectively extracted into the solvent phase. The solvent is 30 percent by volume TBP in n-dodecane. Elements except the uranium (U), technetium (Tc), and iodine (I) remain in the aqueous phase and are discharged into 1-UNCS-05T as aqueous raffinate.

d. The solvent product from the extraction section is washed in the scrub section with a scrub solution of AHA in nitric acid. The scrub solution removes any plutonium from the solvent phase and washes out entrained aqueous impurities carried over from the extraction section. The aqueous product from the scrub contactors is combined with the uranyl nitrate as the aqueous feed to the extraction section. The aqueous stream from the extraction contactors is discharged into 1-UNCS-05T as aqueous raffinate.

e. The uranium-rich solvent then passes through the strip section where the U, Tc, and I are stripped from the organic solvent into the aqueous nitric acid strip solution. The solvent returns to the solvent hold tank (1-UNCS-04T) to be recycled through the UREX contactors. The solvent is replaced as necessary to eliminate the need for a solvent washing section.

f. The uranyl nitrate aqueous stream exiting the strip section flows through an anion exchange column (1-UNCS-03T) which adsorbs the Tc and I. Once the Tc and I are removed, the uranyl nitrate solution is considered clean, with minimal activity. When the anion exchange resin becomes loaded, it is replaced and the spent column is routed to the SRWP system.

g. The uranyl nitrate effluent from the anion exchange column and passes through a diluent wash section of three stages of contactors. The wash consists of independent recycle loops of dodecane at an approximate 3:1 flow ratio to the uranyl nitrate, and it removes TBP carried over in the strip aqueous product. After the dodecane in the first recycle loop becomes loaded with TBP, it is discarded. The second tank is piped to the first, and the third tank is piped to the second; the third tank is filled with fresh dodecane. The washed uranyl nitrate collects in 1-UNCS-06T.

h. After the uranyl nitrate hot feed has been processed, a cold flush of nitric acid is fed through the UREX process to capture the uranium in the contactor holdup volume. The cold flush is a nitric acid solution similar in chemistry to uranyl nitrate that is used to shut down the UREX contactors. The cold flush does not contain any SNM. The cold flush runs for approximately 40 minutes.

i. The cleaned uranyl nitrate in 1-UNCS-06T is pumped to the TDN subsystem for further processing. In addition, 1-UNCS-06T occasionally receives fresh uranyl nitrate from the TSPS which is also pumped to the TDN subsystem for further processing.

j. The raffinate waste in 1-UNCS-05T contains approximately 0.01 percent of the uranyl nitrate, 5 percent of the Tc, 5 percent of the I, and greater than 99 percent of the Pu and other fission products, and is pumped to the RLWS system.

k. The tank vents are routed to the PVVS.

4b.4.1.1.4.3 Thermal Denitration Process Sequence

The following steps describe the TDN subsystem process sequence where cleaned uranyl nitrate is concentrated and then reacted at high temperature to form uranium oxide power. The process is expected to take approximately 12 hours per UNCS batch.

a. Uranyl nitrate feed from the UREX subsystem is concentrated in a thin-film evaporator (1-UNCS-07T). The evaporator has a recycle loop so that the uranium concentration can be increased to approximately [Security-Related Information]. It is estimated that the uranyl nitrate will make three passes through the evaporator.

b. The processing rate of the TDN subsystem is limited by the size of the thermal denitrator, which is constrained by criticality-safe geometry. The flow rate of uranyl nitrate to the evaporator is approximately [Security-Related Information] and there are approximately [Security-Related Information] per UNCS batch, for an estimated evaporator run time of 8 hours.

c. The evaporated water and nitric acid overheads is condensed in a heat exchanger (1-UNCS-02A) cooled by the RPCS.

d. The concentrated uranyl nitrate is pumped from the evaporator and sprayed into the thermal denitrator (1-UNCS-08T), a fluidized bed reactor (FBR). The reaction in the thermal denitrator is as follows (AAEC, 1974):

$$UO_2(NO_3)_2 \bullet 6H_20 \rightarrow UO_x + NO_2 + NO + O_2 + 6H_2O \quad \Delta H = 595 \text{ kJ/mol U} \quad \text{(Equation 4b.4-2)}$$

e. The thermal denitrator operates at 572°F (300°C) and is fluidized by preheated air. The thermal denitration reaction is highly endothermic, so additional heaters are used to heat the thermal denitrator.

f. The thermal denitrator produces solid uranium oxide powder which flows from the FBR bed into a hopper. After an UNCS batch is completed, the uranium oxide is loaded into cans which are staged in the thermal denitration room, and transported to the UO_3 storage racks (1-TSPS-01S) in the TSPS.

g. The overheads from the thermal denitrator are cooled in a heat exchanger (1-UNCS-04A) by the RPCS and drawn by negative pressure to the PVVS.

4b.4.1.1.4.4 Process Equipment

The following is a list of process equipment associated with the UNCS. System components are criticality-safe by design. Nominal sizes and specifications are provided below:

a. 1-UNCS-01C – Thermal denitration blower

   1. Type: positive-displacement blower
   2. Power: 4 hp (3 kW)
   3. Flow rate: 54 scfm (1529 slpm)
   4. Air temperature: 70°F (21°C)
   5. Design/fabrication code: commercial grade

b. 1-UNCS-01F – Uranyl nitrate centrifuge

   1. Type: vertical basket bottom discharge
   2. Power: 10 hp (7.5 kW)
   3. Flow rate: 10 gpm (38 lpm)
   4. Fluid temperature: 122°F (50°C)
   5. Design/fabrication code: commercial grade

c. 1-UNCS-01A – UN evaporator heater

   1. Note: this heater is integral to the UN evaporator vessel; see 1-UNCS-07T.

d. 1-UNCS-02A – Evaporator overheads condenser

   1. Type: plate and frame heat exchanger
   2. Heat duty: 740,000 Btu/hr (217 kW)
   3. Flow rate: 257 actual cubic feet per minute (acfm) (7277 lpm)
   4. Inlet temperature: 212°F (100°C)
   5. Outlet temperature: 140°F (60°C)
   6. Cooling medium: process chilled water
   7. Design/fabrication code: ASME BPVC Section VIII (ASME, 2011)

e. 1-UNCS-03A – Thermal denitrator air heater

   1. Type: electric circulation heater
   2. Power: 14 kW (19 hp)
   3. Flow rate: 54 scfm (1529 slpm)
   4. Inlet temperature: 70°F (21°C)
   5. Outlet temperature: 572°F (300°C)
   6. Design/fabrication code: commercial grade, Nationally Recognized Testing Laboratory (NRTL)

f. 1-UNCS-04A – Thermal denitrator overheads cooler

   1. Type: plate and frame heat exchanger
   2. Heat duty: 89,600 Btu/hr (26 kW)
   3. Flow rate: 88 acfm (2492 lpm)
   4. Inlet temperature: 572°F (300°C)
   5. Outlet temperature: 140°F (60°C)
   6. Cooling medium: process chilled water
   7. Design/fabrication code: ASME BPVC Section VIII (ASME, 2011)

g. 1-UNCS-05A – Recycle target solution cooler

   1. Type: plate and frame heat exchanger
   2. Heat duty: 38,500 Btu/hr (11 kW)
   3. Flow rate: 0.633 gpm (2.510 lpm)
   4. Inlet temperature: 176°F (80°C)
   5. Outlet temperature: 70°F (21°C)
   6. Cooling medium: process chilled water
   7. Design/fabrication code: ASME BPVC Section VIII (ASME, 2011)

h. 1-UNCS-06A – Uranyl nitrate conversion tank cooler

1. Type: plate and frame heat exchanger
2. Heat duty: 84,600 Btu/hr (25 kW)
3. Flow rate: 10 gpm (38 lpm)
4. Inlet temperature: 212°F (100°C)
5. Outlet temperature: 122°F (50°C)
6. Cooling medium: process chilled water
7. Design/fabrication code: ASME BPVC Section VIII (ASME, 2011)

i. 1-UNCS-07A – Uranyl nitrate conversion tank heater

1. Type: electric circulation heater
2. Power: 30 kW (102,364 Btu/hr)
3. Flow rate: 10 gpm (38 lpm)
4. Inlet temperature: 176°F (80°C)
5. Outlet temperature: 212°F (100°C)
6. Design/fabrication code: commercial grade, NRTL

j. 1-UNCS-08A – Thermal denitrator heater

1. Type: resistance heater
2. Power: To be provided in the FSAR
3. Flow rate: To be provided in the FSAR
4. Operating temperature: 572°F (300°C)
5. Design/fabrication code: commercial grade, NRTL

k. 1-UNCS-01P – UN evaporator pump

1. Type: positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 4.64 gpm (17.6 lpm)
4. Fluid temperature: 223°F (106°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

l. 1-UNCS-02P – Centrifuge feed pump

1. Type: positive-displacement
2. Power: 0.50 hp (0.37 kW)
3. Flow rate: 10 gpm (38 lpm)
4. Fluid temperature: 122 to 212°F (50 to 100°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

m. 1-UNCS-03P – UREX feed pump

1. Type: positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 0.9 to 5 gpm (3.4 to 19 lpm)
4. Fluid temperature: 122°F (50°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

n. 1-UNCS-04P – Recycle target solution pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 1.27 gpm (4.81 lpm)
4. Fluid temperature: 70°F (21°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

o. 1-UNCS-05P – Raffinate pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 4.67 gpm (17.68 lpm)
4. Fluid temperature: 70°F (21°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

p. 1-UNCS-06P – Recycle UN pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 1.16 gpm (4.39 lpm)
4. Fluid temperature: 70°F (21°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

q. 1-UNCS-07P – Solvent pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 1.52 gpm (5.75 lpm)
4. Fluid temperature: 70°F (21°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

r. 1-UNCS-08P – Evaporator condensate pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow rate: 1.15 gpm (4.35 lpm)
4. Fluid temperature: 140°F (60°C)
5. Design/fabrication code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

s. 1-UNCS-10P – Diluent pump

1. Type: Positive-displacement
2. Power: 0.25 hp (0.19 kW)
3. Flow Rate: 5.54 gpm (20.97 lpm)
4. Fluid Temperature: 70°F (21°C)
5. Design/Fabrication Code: ANSI/HI 3.1-3.5 (ANSI/HI, 2008)

t. 1-UNCS-01T – Uranyl nitrate conversion tank

1. Size: [Security-Related Information][Proprietary Information]
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME, 2011)

u. 1-UNCS-02T – UREX feed tank

1. Size: [Security-Related Information]
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

v. 1-UNCS-03T – Tc removal column

1. Size: 4.4 gal. (16.7 L)
2. Normal operating pressure: atmospheric
3. Packing: Reillex Resin
4. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

w. 1-UNCS-04T – Solvent hold tank

1. Size: 68 gal. (257 L)
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

x. 1-UNCS-05T – Raffinate hold tank

1. Size: [Security-Related Information]
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

y. 1-UNCS-06T – Recycle UN hold tank

1. Size: [Security-Related Information]
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

z. 1-UNCS-07T – UN evaporator vessel

1. Size: To be provided in the FSAR
2. Type: thin film evaporator
3. Power: 250 kW (853,035 Btu/hr)
4. Flow rate: [Security-Related Information]
5. Inlet temperature: 70°F (21°C)
6. Outlet temperature: 226°F (108°C)
7. Normal operating pressure: atmospheric (vented)
8. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

aa. 1-UNCS-08T – Thermal denitrator

1. Size: 7.2 gal. (27.3 L)
2. Type: fluidized bed reactor with atomized spray injection
3. Flow rate: [Security-Related Information]
4. Inlet temperature: 212°F (100°C)
5. Outlet temperature: 572°F (300°C)
6. Normal operating pressure: 7 psig (48 kPa)
7. External heaters: To be provided in the FSAR
8. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

bb. 1-UNCS-09T – Recycle target solution tank

1. Size: [Security-Related Information][Proprietary Information]
2. Normal operating pressure: atmospheric (vented)
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

cc. 1-UNCS-10T – Diluent wash recycle tank

1. Size: 12 gal. (45 L)
2. Normal operating pressure: To be provided in the FSAR
3. Design/fabrication code: ASME BPVC Section VIII Div. 1 (ASME/2011)

dd. 1-UNCS-01Z – Extraction contactors

1. Type: liquid-liquid centrifugal contactor
2. Size: [Security-Related Information][Proprietary Information]
3. Flow rate: [Security-Related Information]
4. Design/fabrication code: To be provided in the FSAR

ee. 1-UNCS-02Z – Scrub contactors

1. Type: liquid-liquid centrifugal contactor
2. Size: [Security-Related Information][Proprietary Information]
3. Flow rate: [Security-Related Information]
4. Design/fabrication code: To be provided in the FSAR

ff. 1-UNCS-03Z – Strip contactors

1. Type: liquid-liquid centrifugal contactor
2. Size: [Security-Related Information][Proprietary Information]
3. Flow rate: [Security-Related Information]
4. Design/fabrication code: To be provided in the FSAR

gg. 1-UNCS-04Z – Diluent contactors

1. Type: liquid-liquid centrifugal contactor
2. Size: [Security-Related Information][Proprietary Information]
3. Flow rate: [Security-Related Information]
4. Design/fabrication code: To be provided in the FSAR

4b.4.1.2 Physical Properties

4b.4.1.2.1 Process Fluid Properties

This subsection describes the approximate physical, chemical, and radiological isotope properties of the process fluids containing irradiated SNM. The SNM can be divided into six distinct stages during processing through the UNCS (see Table 4b.4-5). The process fluid properties are presented in quantities representing a single batch. The batches for the process stages consist of a UNCS batch, except for the recycled target solution which consists of a TSV batch.

4b.4.1.2.1.1 Physical and Chemical Properties

Refer to Table 4b.4-6, Table 4b.4-7, Table 4b.4-8, Table 4b.4-9, Table 4b.4-10, and Table 4b.4-11 for the physical and chemical properties of each stage in the UNCS process.

4b.4.1.2.1.2 Radioisotope Properties

The UNP subsystem begins receiving target solution from the MEPS after 5.5 days of irradiation and [Proprietary Information] hours of decay. The highest batch activity in the UNCS system is immediately following transfer from MEPS. The estimated radioactive inventory from one TSV batch is [Security-Related Information][Proprietary Information] This curie amount is calculated based on shorter processing times than those described in Subsection 4b.4.1.1.3. The actual processing times are longer; thus, the curie amount listed above is conservative. More detailed radionuclide inventories during processing steps will be provided in the FSAR.

4b.4.1.2.2 Hazardous Chemicals

Several processes in the UNCS involve chemicals that may be hazardous in high concentrations, especially strong acids. Refer to Table 4b.4-12 for a list of the chemicals and associated quantities used in the UNCS. The organics n-dodecane and TBP are recycled within the process and are replaced as needed. Chemicals are prepared within process vessels or well-ventilated areas. Any potential off-gases are evolved within the process equipment and exhausted by the PVVS, isolated from individuals.

SHINE will have chemical inventory controls including separation of chemicals based on the potential for exothermic reactions. These controls, in addition to procedures controlling the processing of irradiated SNM, will include measures to prevent accidents. These procedures and controls will be described in the FSAR.

4b.4.1.2.3 Special Nuclear Material

The special nuclear materials within the UNCS are enriched uranium and plutonium. The enriched uranium is in the form of uranyl sulfate, uranyl nitrate, and uranium oxide, depending on the specific location in the process. The plutonium complexes with AHA in the scrub solution during the UREX process. The UNCS contains a maximum of [Security-Related Information][Proprietary Information] batches (TSV batch and/or UNCS batch) within the system boundaries at one time. Refer to Table 4b.4-13 for the SNM inventory in each stage of the process, and the maximum SNM inventory within the UNCS.

4b.4.1.3 Criticality Control Features

The UNCS prevents inadvertent criticality through inherently-safe design of equipment that may handle the irradiated SNM. A detailed description of the Criticality Safety Program is provided in Section 6b.3.

4b.4.1.4 Shielding and Radiological Protection

The cleanup of irradiated SNM is performed in tank vaults and a shielded hot cell, which limits the exposure of individuals to radiation within the regulatory limits of 10 CFR 20.1201 and 10 CFR 20.1301.

The processes are remotely controlled, manually controlled, or performed with tele-manipulators, with minimal automated sequences. Radiation monitors and alarms are used to monitor release of radiological materials, monitor high background gamma dose levels, and to detect criticality events.

Piping that contains potentially-radiological material is transferred through shielded pipe chases to limit the exposure of individuals to radiation.

4b.4.1.5 Technical Specifications

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the processing of irradiated special nuclear material are provided in Chapter 14.

4b.4.2 PROCESSING OF UNIRRADIATED SPECIAL NUCLEAR MATERIAL

Unirradiated SNM is received in the form of uranium metal. Shipments of SNM are received at the facility in solid form from a DOE supplier. The shipments consist of uranium metal enriched to 19.75 ± 0.2 percent U-235. The SNM is manually transported in approved transport containers and is stored criticality-safe in those containers in accordance with packaging limitations for use. The approved transport container is an NRC licensed Type B shipping package. The SNM is manually transferred to the uranium metal receipt area and removed from the transport containers and stored in uranium metal storage cans in criticality-safe configuration within the uranium metal storage rack. During the receipt process the uranium metal receipt inspections are performed.

The uranium metal is dissolved in nitric acid within the uranium metal dissolution tank (1-TSPS-02T) to provide makeup solution for uranium losses within the process. Once dissolution is complete the uranyl nitrate solution is transferred to the recycle uranyl nitrate hold tank (1-UNCS-06T) for further processing and conversion to uranium oxide by the thermal denitrator (1-UNCS-08T). The uranium oxide produced from the dissolution of unirradiated uranium metal is stored in uranium oxide storage cans in criticality-safe configuration within the uranium oxide storage rack in the uranium oxide storage area. Unirradiated SNM in the form of uranium oxide may be received for initial startup and recharging and are stored in uranium oxide storage cans in criticality-safe configuration within the uranium oxide storage rack in the uranium oxide storage area. Uranium oxide is stored for future production of uranyl sulfate target solution. Operations with irradiated SNM are discussed in Subsection 4b.4.1 and Chapter 9.

This subsection discusses the handling and storage of unirradiated uranium metal and oxide. The TSPS general arrangement, which shows the areas of unirradiated SNM receipt, storage, and handling, is shown in Figure 4b.4-4.

4b.4.2.1 Safety Functions

Prevent inadvertent criticality through inherently-safe design of equipment.

4b.4.2.2 Uranium Metal Receipt

Shipment of uranium metal is received on pallets containing up to four approved transport containers and is transferred to the uranium metal receipt area within the RCA. Uranium metal is stored criticality-safe within the transport containers and is stored in those containers in accordance with packaging limitations for use. The uranium metal receipt area has the capacity to store up to four transport containers on a pallet. The transport containers containing three convenience containers with up to [Security-Related Information] of uranium metal each are individually removed from the transport containers and placed in the receipt ventilation hood. Within the receipt ventilation hood, the convenience container is opened, the contents verified. Up to [Security-Related Information] of uranium metal are then repackaged into the uranium metal storage can. The uranium metal storage can is designed to be criticality-safe in this configuration. The uranium metal storage can is then transferred to the uranium metal storage rack prior to further processing.

4b.4.2.3 Uranium Metal Storage

Uranium metal is stored in the uranium metal storage rack within the uranium metal receipt area. The uranium metal storage rack holds up to [Security-Related Information] uranium storage cans in a criticality-safe configuration by maintaining an edge to edge separation distance between any two cans of at least 6 in. (15.2 cm) as seen in Figures 4b.4-5 and 4b.4-7. Uranium metal in the uranium storage cans is transferred individually to the uranium metal dissolution tank (1-TSPS-02T) located within the TSPS area. Detailed descriptions of the processing of unirradiated uranium metal dissolution can be found in Section 9b.2.

4b.4.2.4 Uranium Oxide Receipt

Shipments of uranium oxide may be received in approved containers and are manually transferred to the uranium oxide receipt area within the RCA. As needed by the facility, containers containing uranium oxide are individually placed in the receipt ventilation hood. Within the receipt ventilation hood the container is opened and the contents are verified. Up to [Security-Related Information] of uranium oxide is then repackaged into the uranium oxide storage can. The uranium oxide storage can is designed to be criticality-safe in this configuration. The uranium oxide storage can is then manually transferred to the uranium oxide storage rack prior to further processing. Uranium oxide is also received as a product of TDN. This uranium oxide is also packaged in uranium oxide storage cans and stored in the uranium oxide storage rack prior to further processing. A feedstock of uranium oxide is required at plant startup to fill the fluidized bed tray(s) in the thermal denitration equipment.

4b.4.2.5 Uranium Oxide Storage

Uranium oxide is stored in the uranium oxide storage rack within the uranium oxide storage area. The uranium oxide storage rack holds up to [Security-Related Information] uranium oxide storage cans in a criticality-safe configuration, as seen in Figures 4b.4-6 and 4b.4-7. Uranium oxide in the uranium oxide storage cans is transferred to the uranyl sulfate preparation tank (1-TSPS-01T) located within the uranyl sulfate preparation area as seen in Figure 4b.4-4. Introduction of the uranium oxide to this tank is done within a glovebox interface. Detailed descriptions of uranium oxide used to prepare target solution can be found in Section 9b.2.

4b.4.2.6 Unirradiated SNM Related Equipment

The following is a list of process equipment associated with processing unirradiated SNM. System components are criticality-safe by design. Nominal sizes and specifications are provided below:

- Uranyl sulfate preparation tank (1-TSPS-01T)
  - Quantity: 2
  - Size: [Security-Related Information]
  - Normal operating pressure: atmospheric (vented)
  - Design/fabrication code: ASME BPVC Section VIII Div 1 (ASME, 2011)
- Uranium metal dissolution tank (1-TSPS-02T)
  - Quantity: 2
  - Size: [Security-Related Information]
  - Normal operating pressure: atmospheric (vented)
  - Design/fabrication code: ASME BPVC Section VIII Div 1 (ASME, 2011)
- Target solution hold tank glovebox
  - Quantity: 2
  - Design/fabrication code: AGS-G001-2007 (AGS, 2007)
- Uranium receipt ventilation hood
  - Quantity: 1
  - Design/fabrication code: AGS-G001-2007 (AGS, 2007)
- TDN interface glovebox
  - Quantity: 1
  - Design/fabrication code: AGS-G001-2007 (AGS, 2007)
- Uranium metal storage rack
  - Storage locations: [Security-Related Information] cans

- Uranium oxide storage rack
  - Storage locations: [Security-Related Information] cans
- Uranium metal storage can
  - Quantity: [Security-Related Information]
- Uranium oxide storage can
  - Quantity: [Security-Related Information]

4b.4.2.7 Criticality Control Features

Process of unirradiated SNM prevents inadvertent criticality through the use of criticality-safe equipment. A detailed description of the Criticality Safety Program is provided in Section 6b.3.

4b.4.2.8 Technical Specifications

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the processing and storage of SNM is provided in Chapter 14.

Table 4b.4-1 Estimated RPF Special Nuclear Material Inventory

SNM Chemical Form	SNM Physical Form	SNM Inventory lb. (kg)
Uranium metal	Metal	[Security-Related Information]
Uranium oxide	Powder	[Security-Related Information]
Uranyl nitrate	Aqueous	[Security-Related Information]
Uranyl sulfate	Aqueous	[Security-Related Information]
Total uranium(a)	-	[Security-Related Information]
Plutonium (nitrate, sulfate, solidified)	Aqueous, solid	[Security-Related Information]

a) Total uranium accounts only for elemental uranium, while the other chemical forms account for the elements associated with the chemical.

Table 4b.4-2 System Interfaces – Uranyl Nitrate Preparation

Interfacing System	Interface Description
Molybdenum Extraction and Purification System (MEPS)	The UNP subsystem interfaces with the MEPS. MEPS routes the post-extraction spent target solution and sulfuric acid washes to the UNCS.
Uranium Extraction (UREX) subsystem of the UNCS	The UNP subsystem interfaces with the UREX subsystem. The UNP subsystem feeds uranyl nitrate to the UREX contactors.
Target Solution Preparation System (TSPS)	The UNP subsystem interfaces with the TSPS. The recycled target solution is pumped from the UNCS (1-UNCS-09T) to the TSPS (1-TSPS-03T).
Process Vessel Vent System (PVVS)	The UNP subsystem interfaces with the PVVS. The PVVS provides vessel venting to tanks 1-UNCS-01T, 1-UNCS-02T, and 1-UNCS-09T.
Solid Radioactive Waste Packaging (SRWP)	The UNP subsystem interfaces with the SRWP system. The uranyl nitrate centrifuge, 1-UNCS-01F, transfers sulfate slurry to the SRWP system.
Radioactive Drain System (RDS)	The UNP subsystem interfaces with the RDS. The RDS provides drain lines for the tank vaults containing 1-UNCS-01T, 1-UNCS-02T, and 1-UNCS-09T, and from the centrifuge area.
Radioisotope Process Facility Cooling System (RPCS)	The UNP subsystem interfaces with the RPCS. Process chilled water is supplied to heat exchangers 1-UNCS-05A and 1-UNCS-06A.
Normal Electrical Power Supply System (NPSS)	The UNP subsystem interfaces with the NPSS. The NPSS is distributed to operate the centrifuge (1-UNCS-01F), pumps (1-UNCS-02P, 1-UNCS-03P, 1-UNCS-04P), heating element (1-UNCS-07A), and ancillary equipment.

Table 4b.4-3 System Interfaces - Uranium Extraction

Interfacing System	Interface Description
Uranyl Nitrate Preparation (UNP) subsystem of the UNCS	The UREX subsystem interfaces with the UNP subsystem. The UNP subsystem feeds the UREX contactors.
Thermal Denitration (TDN) subsystem of the UNCS	The UREX subsystem interfaces with the TDN subsystem. The UREX subsystem feeds uranyl nitrate from the recycle UN hold tank (1-UNCS-06T) to the UN evaporator vessel (1-UNCS-07T).
Target Solution Preparation System (TSPS)	The UREX subsystem interfaces with the TSPS. The TSPS feeds fresh uranyl nitrate from the uranium metal dissolution tank (1-TSPS-02T) to the recycle UN hold tank (1-UNCS-06T).
Process Vessel Vent System (PVVS)	The UREX subsystem interfaces with the PVVS. PVVS provides vessel venting for tanks 1-UNCS-04T, 1-UNCS-05T, and 1-UNCS-06T.
Aqueous Radioactive Liquid Waste Storage System (RLWS)	The UREX subsystem interfaces with the RLWS system. The UREX subsystem transports fission products from the raffinate hold tank (1-UNCS-05T) to RLWS.
Organic Liquid Waste Storage and Export (OLWS)	The UREX subsystem interfaces with the OLWS system. Organic waste from the UREX subsystem are routed to the OLWS system.
Radioactive Drain System (RDS)	The UREX subsystem interfaces with the RDS. The RDS provides drain lines for the tank vaults containing 1-UNCS-04T, 1-UNCS-05T, and 1-UNCS-06T.
Solid Radioactive Waste Packaging (SRWP)	The UREX subsystem interfaces with the SRWP system. The SRWP provides the 55-gallon drum for the Tc removal column (1-UNCS-03T).
Normal Electrical Power Supply System (NPSS)	The UREX subsystem interfaces with the NPSS. The NPSS is distributed to operate the extraction contactors (1-UNCS-01Z), scrub contactors (1-UNCS-02Z), strip contactors (1-UNCS-03Z), diluent contactors (1-UNCS-04Z), pumps (1-UNCS-05P, 1-UNCS-06P, 1-UNCS-07P, 1-UNCS-10P), and ancillary equipment.

Table 4b.4-4 System Interfaces - Thermal Denitration

Interfacing System	Interface Description
Uranium Extraction (UREX) subsystem of the UNCS	The TDN subsystem interfaces with the UREX subsystem. The UREX subsystem feeds uranyl nitrate from the recycle UN hold tank (1-UNCS-06T) to the UN evaporator vessel (1-UNCS-07T).
Process Vessel Vent System (PVVS)	The TDN subsystem interfaces with the PVVS. The PVVS provides vessel venting for the thermal denitrator (1-UNCS-08T).
Radioactive Drain System (RDS)	The TDN subsystem interfaces with the RDS. The RDS system provides a drain line for the thermal denitration area.
Radioisotope Process Facility Cooling System (RPCS)	The TDN subsystem interfaces with the RPCS. The RPCS provides process chilled water return for heat exchangers 1-UNCS-02A and 1-UNCS-04A.
Normal Electrical Power Supply System (NPSS)	The TDN subsystem interfaces with the NPSS. The NPSS is distributed to operate the pumps (1-UNCS-01P, 1-UNCS-08P), blower (1-UNCS-01C), heating elements (1-UNCS-03A, 1-UNCS-01A, and 1-UNCS-08A), and ancillary equipment.

Table 4b.4-5 UNCS Process Stages

Process Stage	SNM Physical Form	SNM Chemical Form	Process Location
Recycled Target Solution	Liquid	Uranyl Sulfate	1-UNCS-09T
Spent Target Solution	Liquid	Uranyl Sulfate	1-UNCS-01T
UREX Feed	Liquid	Uranyl Nitrate	1-UNCS-02T
TDN Feed	Liquid	Uranyl Nitrate	1-UNCS-06T
UREX Raffinate	Liquid	Transuranics	1-UNCS-05T
TDN Product	Solid	Uranium Oxide	1-UNCS-08T

Table 4b.4-6 Physical and Chemical Properties - Recycled Target Solution

Property	Value
Content	[Proprietary Information] wt% water; [Proprietary Information] wt% sulfuric acid; [Proprietary Information] wt% uranyl sulfate
Temperature	70°F (21°C)
Density	[Proprietary Information] lb/ft^3
Mass	[Security-Related Information][Proprietary Information] lb/batch
Volume	[Security-Related Information][Proprietary Information] gal/batch
Uranium Content	[Security-Related Information][Proprietary Information] lb/batch uranyl sulfate ([Security-Related Information][Proprietary Information] lb/batch uranium)
Uranium Concentration	[Security-Related Information][Proprietary Information] lbU/gal ([Security-Related Information][Proprietary Information] gU/L)
pH	[Proprietary Information]
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-7 Physical and Chemical Properties - Spent Target Solution

Property	Value
Content	[Proprietary Information] wt% water; [Proprietary Information] wt% sulfuric acid; [Proprietary Information] wt% uranyl sulfate
Temperature	176°F (80°C)
Density	[Proprietary Information] lb/ft^3
Mass	[Security-Related Information][Proprietary Information] lb/batch
Volume	[Security-Related Information][Proprietary Information] gal/batch
Uranium Content	[Security-Related Information][Proprietary Information] lb/batch uranyl sulfate ([Security-Related Information][Proprietary Information] lb/batch uranium)
Uranium Concentration	[Security-Related Information][Proprietary Information] lbU/gal ([Security-Related Information][Proprietary Information] gU/L)
pH	[Proprietary Information]
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-8 Physical and Chemical Properties - UREX Feed

Property	Value
Content	[Proprietary Information] wt% water; [Proprietary Information] wt% nitric acid; [Proprietary Information] wt% uranyl nitrate
Temperature	122°F (50°C)
Density	66.9 to 67.4 lb/ft^3
Mass	[Security-Related Information][Proprietary Information] lb/batch
Volume	[Security-Related Information] gal/batch
Uranium Content	[Security-Related Information] lb/batch uranyl nitrate ([Security-Related Information] lb/batch uranium)
Uranium Concentration	[Security-Related Information] lbU/gal ([Security-Related Information] gU/L)
pH	0.0
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-9 Physical and Chemical Properties - TDN Feed

Property	Value
Content	[Proprietary Information] wt% water; [Proprietary Information] wt% nitric acid; [Proprietary Information] wt% uranyl nitrate
Temperature	70°F (21°C)
Density	[Security-Related Information] lb/ft^3
Mass	3914 to 4776 lb/batch
Volume	[Security-Related Information] gal/batch
Uranium Content	[Security-Related Information] lb/batch uranyl nitrate ([Security-Related Information] lb/batch uranium)
Uranium Concentration	[Security-Related Information] lbU/gal ([Security-Related Information] gU/L)
pH	2.0
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-10 Physical and Chemical Properties - UREX Raffinate

Property	Value
Content	[Proprietary Information] wt% water; [Proprietary Information] wt% nitric acid; [Proprietary Information] 0.04 wt% acetohydroxamic acid; trace Pu-239
Temperature	70°F (21°C)
Density	61.5 to 62.2 lb/ft^3
Mass	[Security-Related Information]
Volume	[Security-Related Information]
Uranium Content	No significant quantities of uranium
Uranium Concentration	No significant quantities of uranium
pH	0.0
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-11 Physical and Chemical Properties - TDN Product

Property	Value
Content	100 wt% uranium oxide
Temperature	572°F (300°C)
Density	To be provided in FSAR
Mass	[Security-Related Information] lb/batch
Volume	To be provided in FSAR
Uranium Content	[Security-Related Information] lb/batch uranium oxide ([Security-Related Information] lb/batch uranium)
Uranium Concentration	To be provided in FSAR
pH	Not applicable
Enrichment	Generated from low-enriched U-235 metal (19.75 ± 0.2%)

Table 4b.4-12 UNCS Hazardous Chemicals Inventory

Chemical	Quantity Per UNCS Batch
Acetohydroxamic Acid	0.83 lb.
[Proprietary Information]	[Proprietary Information] lb.
n-Dodecane	260 lb. (recycled in process)
[Proprietary Information]	[Proprietary Information] lb.
Nitric Acid	157 lb.
Tri-Butyl Phosphate	99 lb. (recycled in process)

Table 4b.4-13 Estimated SNM Process Inventory

Process Stage	U-233 (g/batch)	U-235 (kg/batch)	Pu-239 (g/batch)
Recycled Target Solution	[Security-Related Information]	[Proprietary Information]	< 1
Spent Target Solution	[Security-Related Information]	[Proprietary Information]	[Security-Related Information]
UREX Feed	[Security-Related Information]	[Proprietary Information]	[Security-Related Information]
TDN Feed	[Security-Related Information]	[Proprietary Information]	0
UREX Raffinate	0	0	[Security-Related Information]
TDN Product	[Security-Related Information]	[Proprietary Information]	0
Maximum UNCS Inventory (Constitutes [Proprietary Information] Batches)	[Security-Related Information]	[Proprietary Information]	[Security-Related Information]

4b.5 REFERENCES

AAEC, 1974. The Thermal Denitration of Uranyl Nitrate in a Fluidised Bed Reactor, A.G. Fane, B.G. Charlton, and P.G. Alfredson, Australian Atomic Energy Commission, AAEC/E284, July 1974.

ACI, 2007a. Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, ACI 349-06, American Concrete Institute, September 2007.

ACI, 2007b. Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures, ACI 349.1R-07, American Concrete Institute, 2007.

AGS, 2007. Guideline for Gloveboxes, AGS-G001, Third Edition, American Glovebox Society, February 2007.

ANSI/ANS, 2006a. Nuclear Analysis and design of Concrete Radiation Shielding for Nuclear Power Plants, ANSI/ANS-6.4-2006, American National Standards Institute, 2006.

ANSI/ANS, 2006b. Specification for Radiation Shielding Materials, ANSI/ANS-6.4.2-2006, American National Standards Institute, 2006.

ANSI/ANS, 2007. Program for Testing Radiation Shields in Light Water Reactors (LWR), ANSI/ANS-6.3.1-1987 R2007, American National Standards Institute, 2007.

ANSI/HI, 2008. Rotary Pumps (A109), 3.1-3.5, American National Standards Institute, January 1, 2008.

ASME, 2011. Boiler & Pressure Vessel Code – Rules for Construction of Pressure Vessels, Section VIII, American Society of Mechanical Engineers, July 1, 2011.

ASME, 2012. Process Piping, Code for Pressure Piping, B31.3-2012, American Society of Mechanical Engineers, January 10, 2013.

LANL, 2004. MCNP5_RSICC_1.30, LA-UR-04-5921, MCNP Monte Carlo Team, X-5, Los Alamos National Laboratory, 2004.

Olsher, 2006. A Practical Look at Monte Carlo Variance Reduction Methods in Radiation Shielding, *Nuclear Engineering and Technology*, Volume 38 No. 3, pp. 225-230, R.H. Olsher, April 2006.