ENCLOSURE 3
ATTACHMENT 1

SHINE MEDICAL TECHNOLOGIES, INC.

SHINE MEDICAL TECHNOLOGIES, INC. PRELIMINARY SAFETY ANALYSIS REPORT REVISIONS RESULTING FROM REQUEST FOR ADDITIONAL INFORMATION RESPONSES

SUMMARY OF PRELIMINARY SAFETY ANALYSIS REPORT CHANGES
PUBLIC VERSION

PRELIMINARY SAFETY ANALYSIS REPORT CHANGES
(MARK-UP)

95 pages follow

The process vessel vent system (PVVS) collects and processes acidic and noble gases from the vents of process vessels that handle the main process fluids. This system is briefly discussed in Subsection 4b.1.3.5 and discussed in detail in Section 9b.6.

The molybdenum isotope product packaging system (MIPS) receives the Mo-99 from MEPS and packages it for shipment to the customers. This system is addressed briefly in Section 4b.1.3.4 and in detail in 9b.7.1.

Other systems located in the RPF are briefly addressed in Section 4b.1 and are discussed in more detail in the following chapters of this report.

Refer to Table 3.1-1 for the system safety classifications.

The legend for process flow diagrams provided in the PSAR is found in Figure 1.3-6.

1.3.4 ENGINEERED SAFETY FEATURES

Engineered safety features (ESFs) are SSCs of the facility that mitigate design basis events or accidents.

ESFs for the IF are addressed in Section 6a2.2 and Table 6a2.2-~~1~~2. ESFs in the IF are related to confinement of radiological material.

Confinement is the term used to describe the low-leakage boundary that surrounds radioactive materials released during an accident and the associated RCA ventilation system (RV). Confinement systems are designed to localize release of radioactive material to controlled areas in normal operational states and mitigate the consequences of DBAs. Radiation protection control features such as adequate shielding and the RV minimize hazards normally associated with radioactive materials. The principal design and safety objective of the confinement systems is to protect on-site personnel, the public, and the environment. The second design objective is to minimize reliance on administrative or complex active engineering controls to provide a confinement system that is as simple and fail-safe as reasonably possible.

The TSV, TSV dump tank, TOGS, and associated components act as the primary pressure boundary and are safety-related SSCs. These ESFs act as the primary fission product boundary and are referred to as the primary system boundary (PSB). The confinement boundary of the IU cell and TOGS shielded cell encloses the PSB.

Confinement of the IU cells is achieved through the RV, the engineered safety feature actuation system (ESFAS), and the biological shielding provided by the steel and concrete structures comprising the walls, roofs, and penetrations of the IU cell and TOGS shielded cell. Shielding of the IU cells is discussed in detail in Section 4a2.5.

ESFs outside the IF are addressed in Section 6b.2 and Table 6b.2-~~1~~2. The ESFs are related to confinement of radiological material and hazardous material. The RPF confinement areas include hot cell enclosures and gloveboxes for process operations and trench and vault enclosures for process tanks and piping.

Figure 1.3-3 – Production Building Sections Preliminary Arrangement

Security-Related Information – Withhold Under 10 CFR 2.390

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2.2.2.4 Approach and Holding Patterns near the SHINE Facility

Three airports have holding patterns near the SHINE facility. Table 2.2-7 provides a list of approach and holding patterns in the vicinity of the SHINE facility. The distance from the edge of each holding pattern to the SHINE facility is greater than two statute miles and they are therefore, in accordance with SRP Subsection 3.5.1.6 screened out and no further evaluation is performed on the holding patterns.

2.2.2.5 Evaluation of the Aircraft Hazard

2.2.2.5.1 Evaluation of Airways

The U.S. Department of Energy (DOE) provides a method for estimating the probability per year of an aircraft crashing into the facility. The methodology is outlined in DOE Standard DOE-STD-3014-96 (DOE, 2006) and utilizes crash rates for non-airport operations.

The non airport crash impact frequency evaluation is determined from using the following "four factor formula" (DOE, 2006):

$$F_j = N_j P_j f_j(x,y) A_j \qquad \text{(Equation 2.2-1)}$$

Where:

F_j	=	crash impact frequency
j	=	each type of aircraft suggested in the DOE Standard
N_jP_j	=	expected number of in-flight crashes per year
$f_j(x,y)$	=	probability, given a crash, that the crash occurs in a 1-square-mile area surrounding the facility
A_j	=	effective plant area

Tables B-14 and B-15 of DOE-STD-3014-96 (DOE, 2006) provide $N_jP_jf_j(x,y)$ values for general aviation aircraft, air carriers, air taxis, and small military aircraft applicable for specific DOE sites. ~~In addition,~~Tables B-14 and B-15 of DOE-STD-3014-96 (DOE, 2006) also ~~includes~~provide crash probabilities for unspecified locations in the continental United States (CONUS)~~ in Tables B-14 and B-15 of that document~~, and Table B-43 of DOE-STD-3014-96 (DOE, 2006) provides a generic crash frequency for helicopters. Therefore, CONUS average values and generic helicopter values are used for the ~~new plant~~SHINE facility and are ~~listed~~provided in Table 2.2-8~~ (DOE, 2006)~~.

The effective plant area (A_j) for the safety-related structures of the SHINE facility depends on the length, width, and height of the facility, as well as the aircraft's wingspan, skid distance, and impact angle as explained below (DOE, 2006):

$$A_j = A_f + A_s \qquad \text{(Equation 2.2-2)}$$

Where:

$$A_f = (WS + R) \cdot H \cdot \cot\Phi + (2 \cdot L \cdot W \cdot WS) / R + L \cdot W \qquad \text{(Equation 2.2-3)}$$

And:

$$A_s = (WS + R) \cdot S \qquad \text{(Equation 2.2-4)}$$

Where:

A_f = effective fly-in area

A_s = effective skid area

WS = aircraft wingspan (Table 2.2-8)

R = length of the diagonal of the facility = $(L^2 + W^2)^{0.5}$

H = facility height, facility-specific

$\cot\Phi$ = mean of the cotangent of the aircraft impact angle (Table 2.2-8)

L = length of facility, facility-specific

W = width of facility, facility-specific

S = aircraft skid distance (mean value) (Table 2.2-8)

The total effective area (A_j) for the safety-related structures and the two stacks of the SHINE facility ~~(in the case of the SHINE facility, the only safety-related structures are the production building and the vent stack)~~ were calculated. ~~Bounding dimensions of 316 ft. (96.3 m) by 316 ft. (96.3 m) by 80 ft. (24 m) tall for the production facility were assumed. The bounding dimensions for the vent stack were assumed to be 105 ft. (32 m) tall and 10 ft. (3.1 m) in diameter.~~ Dimensions of the production facility used in the analysis include a width of 200 ft. 4 in, a length of 194 ft. 4 in., and a height of 57 ft. 8 in., as shown in Figures 1.3-3 and 1.3-5. The dimensions for the off-gas stack were assumed to be 24 ft. 6 in. tall and 4 ft. 8 in. in diameter, and the dimensions of the boiler stack were assumed to be 49 ft. 3 in. tall and 1 ft. 8 in. in diameter.

The calculated effective area for the five aircraft types is provided in Table 2.2-9.

~~The crash impact probabilities from airways for the five aircraft types are added to determine the overall probability for small and large aircraft. Small aircraft consist of air taxis, general aviation, and small military aircraft. Large aircraft consist of air carriers and large military aircraft.~~ The crash impact probabilities for small non-military aircraft (i.e., general aviation and air taxi), ~~and~~ large non-military aircraft (i.e., air carriers), and military aircraft (i.e., small aircraft and helicopter) from airways are provided in Table 2.2-15.

2.2.2.5.2 Evaluation of Airports

Only the Southern Wisconsin Regional Airport (SWRA) and the Mercy Hospital Heliport are within 5 mi. (8 km) of the SHINE facility. No airport between 5 mi. (8 km) and 10 mi. (16 km) from the SHINE facility has greater than $200d^2$ (where d is the distance to the SHINE facility in kilometers) aircraft operations per year. Based on this screening criteria (from NUREG-1537,

Section 2.2.2), only the SWRA and the Mercy Hospital Heliport need to be evaluated for the potential hazard posed by aircraft using these facilities. The Mercy Hospital Heliport is only used sporadically. The greater size of aircraft using the SWRA, greater number of operations at the SWRA, and the closer distance from the SHINE facility to the SWRA renders a separate analysis of the Mercy Hospital unnecessary because the probability contribution from Mercy Hospital is negligible.

SRP Section 3.5.1.6, provides a method for estimating the probability of an aircraft crashing into the site from the operations at nearby airports. The probability per year of an aircraft crashing into the site due to airport operations at nearby airports is:

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} (C_j N_{ij} A_j) \qquad \text{(Equation 2.2-5)}$$

Where:

P_A	=	probability of crash per year
M	=	number of different types of aircraft
L	=	number of flight trajectories affecting the site, in this case, the runways 14-32, 4-22, and 18-36
C_j	=	probability per square mile of a crash per aircraft movement, for the jth aircraft
N_{ij}	=	number (per year) of movements by the jth aircraft along the ith flight path
A_j	=	effective plant area (in square miles) for the jth aircraft

For the Southern Wisconsin Regional Airport, ~~three effective areas consisting of those for air carrier, general aviation, and large military are used. There is not much difference in area for large and small military aircraft; using large aircraft is conservative. At the SWRA, there are air taxi and commuter flights in itinerant operations and there are civil flights in local operations. Air taxis are small aircraft (DOE, 2006). Per discussion with the SWRA Director, all local civil operations are general aviation which is considered to be small aircraft (Burdick, 2012b)~~the same calculated effective areas used for crash impact probabilities from airways are used to calculate the crash impact probabilities from airports.

The calculated effective ~~plant~~ area for ~~each~~the five aircraft types is provided in Table 2.2-~~10~~9.

The total operations at the SWRA used in the evaluation of the airport are based on the Federal Aviation Administration (FAA) Office of Aviation Policy and Plans (APO) Terminal Area Forecast Detail Report issued ~~January 2012 (APO, 2012) for the years 2010 through 2040~~ February 2014 (APO, 2014). The number of operations at the airport for each type of aircraft are listed in Table 2.2-13. The operations include both itinerant and local operations. ~~The maximum~~

~~number of operations, for each type of aircraft in the years 2010 through 2040, is listed in Table 2.2-13.~~The values provided in Table 2.2-13 are obtained from the maximum forecasted number of operations from 2014 through 2040 for non-military aircraft. A historical average of military operations from 1990 through 2013 is used to calculate crash impact probabilities from military operations. This method is used for military aircraft because the Terminal Area Forecast Detail Report did not forecast any growth from 2014 through 2040. The use of historical averages for military operations is more conservative than use of forecasted values, since the forecasted values are lower.

Based on communication with the Southern Wisconsin Regional Airport and the Janesville Air Traffic Control Tower, the following information was obtained:

- 100 percent of the civil operations are general aviation which is considered to be small aircraft. In addition, air taxi operations are treated as general aviation ~~(Burdick, 2012b)~~ (DOE, 2006).
- The percent of total operations on each runway (Burdick, 2012c):
  - Runway 14–32: 35 percent.
  - Runway 4–22: 50 percent.
  - Runway 18–36: 15 percent
- The breakdown of operations on each runway are as follows (Burdick, 2012c):
  - Runway 14–32: 60 percent operations use Runway (RW) 32 and 40 percent operations use RW 14.
  - Runway 4–22: 60 percent operations use RW 4 and 40 percent operations use RW 22.
  - Runway 18–36: 50 percent operations use RW 18 and 50 percent operations use RW 36.
- ~~Since there is no information on the breakdown of military operations at the airport between small and large aircraft, all military operations are considered to be large aircraft~~Military operations are 1 percent small aircraft and 99 percent helicopters (SHINE, 2015).

Based on this information, the operations on each runway, by type of aircraft, are provided in Table 2.2-14. The distance from the end of each runway to the SHINE facility center point is provided in Table 2.2-11. The probability of a fatal crash per square mile per aircraft movement is provided in Table 2.2-12, where the probabilities for general aviation crashes are applied to general aviation and local civil operations, and probabilities for air carriers are applied to air carrier and air taxi operations.

The crash impact probabilities for ~~small and large aircraft~~small non-military aircraft (i.e., general aviation itinerant operations, local civil operations, and air taxi itinerant operations), large non-military aircraft (i.e., air carriers), and military aircraft (i.e., small aircraft and helicopter) from airports are provided in Table 2.2-15.

2.2.2.5.3 Results of Evaluation of Airways and Airports

NUREG-1537 does not provide acceptance criteria to be used to evaluate the aircraft accident probability posed by nearby airports and airways. ~~The IAEA-TECDOC-1347, "Consideration of external events in the design of nuclear facilities other than nuclear power plants, with emphasis on earthquakes," Section 4.3 (International Atomic Energy Agency [IAEA], 1987), does provide acceptance criteria for aircraft accident probability. The risk of an aircraft accident is considered~~

~~acceptable if the occurrence is less than 1E-05 per year. The evaluation determined that large aircraft meet this criterion (4.520E-06). The calculated crash probability for small aircraft does not meet this criterion (5.424E-04).~~ DOE-STD-3014-96 (DOE, 2006) provides a screening value of 1E-06 per year, where the risk of an aircraft accident is considered acceptable if the frequency of occurrence is less than 1E-06 per year. The calculated crash probability for small non-military aircraft does not meet this criterion (2.6E-04). The safety-related structures of the SHINE facility are designed to withstand the impact of a small non-military aircraft (see Section 3.4). The combined probability of all other aircraft crashes does meet this criterion (6.1E-07).

2.2.3 ANALYSIS OF POTENTIAL ACCIDENTS AT FACILITIES

On the basis of the information provided in Subsection 2.2.1 and Subsection 2.2.2, the potential accidents to be considered as design-basis events and the potential effects of those accidents on the facility, in terms of design parameters (e.g., overpressure, missile energies) or physical phenomena (e.g., impact, flammable or toxic clouds) were identified in accordance with 10 CFR 20, 10 CFR 50.34, Regulatory Guide 1.78, Regulatory Guide 1.91, Regulatory Guide 1.206, Regulatory Guide 4.7, and NUREG-1537. The events are discussed in the following subsections.

2.2.3.1 Determination of Design-Basis Events

Design-basis events, internal and external to the SHINE facility, are defined as those accidents that have a probability of radiological release to the public on the order of magnitude of 1E-07 per year, or greater, with the potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR 50.34 could be exceeded. The following accident categories were considered in selecting design-basis events: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, and fires. The postulated accidents that would result in a chemical release were analyzed at the following locations:

- Nearby transportation routes such as Highway 51 and Interstate-90 (I-90), the Union Pacific Railway, and nearby natural gas pipelines.
- Nearby chemical and fuel storage facilities (industry in the towns of Janesville and Beloit, Wisconsin).
- Chemicals stored or used at the SHINE facility.

2.2.3.1.1 Explosions

Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels were considered for facilities and activities in the vicinity of the plant or on-site where such materials are processed, stored, used, or transported in quantity. The effects of explosions are a concern in analyzing structural response to blast pressures. The effects of blast pressure from explosions from nearby railways, highways, or facilities to critical plant structures were evaluated to determine if the explosion would have an adverse effect on plant operation or would prevent a safe shutdown.

The allowable (i.e. standoff) and actual distances of hazardous chemicals transported or stored were determined in accordance with Regulatory Guide 1.91, Revision 1, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants. Regulatory Guide 1.91 cites 1 pound per square inch (psi) (6.9 kilopascal [kPa]) as a conservative value of peak positive incident overpressure, below which no significant damage

The 133,946 lb. (60,756 kg) tank of gasoline at Janesville Jet Center has a standoff distance to where the concentration falls below the LEL of 628 yd. (574 m), 0.36 mi. (0.57 km). The Janesville Jet Center is 0.9 mi. (1.45 km) from the SHINE facility.

A 440,000 lb. (199,580 kg) tank of ethylene oxide has a standoff distance to where the concentration falls below the LEL of 947 yd. (866 m), 0.54 mi. (0.87 km). The nearest instance of a large tank of ethylene oxide is the Union Pacific Railway, 1.6 mi. (2.6 km) from the SHINE facility.

A 320,000 lb. (145,149 kg) tank of methyl chloride has a standoff distance to where the concentration falls below the LEL of 425 yd. (388 m), 0.24 mi. (0.39 km). The nearest instance of a large tank of methyl chloride is the Union Pacific Railway, 1.6 mi. (2.6 km) from the SHINE facility.

The ALOHA model shows that the vapor pressure of n butyl alcohol at the analysis temperature of 81°F (27°C) is less than the LEL. Therefore, n-butyl alcohol cannot support a vapor cloud explosion.

The results of flammable vapor cloud ignition and explosion analyses are summarized in Table 2.2-18.

2.2.3.1.2.6 Flammable Vapor Cloud (Delayed Ignition) Related Impacts Affecting the Design

A facility is acceptable when the calculated rate of occurrence of severe consequences from any external accident is less than 1 x 10^{-6} occurrences per year and reasonable qualitative arguments can demonstrate that the realistic probability is lower. Regulatory Guide 1.91 cites 1 psi (6.9 kPa) as a conservative value of peak positive incident overpressure, below which no significant damage would be expected. SHINE safety-related areas are designed to withstand a peak positive overpressure of at least 1 psi (6.9 kPa) without loss of function.

The analyses presented in this subsection demonstrate that a 1 psi (6.9 kPa) peak positive overpressure is not exceeded at a safety-related structure for any of the postulated flammable vapor cloud, delayed ignition event scenarios.

~~2.2.3.1.3 Toxic Chemicals~~

~~The control room is not safety-related. The control room operators are not required to operate safety-related equipment to ensure the safety of the public. Therefore, a toxic gas release is not a hazard to the facility.~~

~~2.2.3.1.3.1 Toxic Chemical Related Impacts Affecting the Design~~

~~Because the control room is not safety-related, toxic chemical release to the control room does not have the potential to cause a radiological release to the public.~~

2.2.3.1.3 Toxic Chemicals

Accidents involving the release of chemicals in the vicinity of the plant or on-site were considered for their potential toxicity and ability to affect personnel in the SHINE Control Room.

On-site chemical releases are evaluated using the methodology in Subsection 13b.3.2. Off-site chemical releases are evaluated in this subsection.

The potential for an off-site toxic gas release was evaluated within 5 mi. (8 km) of the site.

SHINE considered stationary sources and mobile sources expected to be transported on US 51, I-90/39, or on local railroads. The effects of a chemical release from a pipeline were considered bounded by the delayed ignition explosion of a pipeline.

Chemicals are screened in several ways. Only chemicals with vapor pressures greater than 10 Torr at 100°F were considered for further evaluation. Mobile sources were not considered if their shipment was not frequent (i.e., less than 10 shipments per year for truck traffic or 30 shipments per year for rail traffic).

In some cases, chemicals are screened as being bounded by other chemicals. A chemical determined to not present a toxic hazard to the SHINE site can be considered bounding to other chemicals that meet these four criteria: (1) have the same or lower vapor pressure; (2) have similar or lower toxicity; (3) are located the same or a farther distance away; and (4) are present in a similar or lower quantity. Additionally, in order to bound some chemicals, it was assumed that given identical meteorological conditions, initial chemical inventories, and travel distances:

a. A chemical that exists as a gas or vapor will result in higher downwind concentrations than one that exists as a liquid.
b. Volatile liquids, liquids with higher vapor pressures, or liquids with low boiling points near ambient temperatures will result in higher downwind concentrations than non-volatile liquids, liquids with lower vapor pressures, and liquids with high boiling points.
c. A spill or leak of a solid chemical will not result in significant atmospheric concentrations capable of incapacitating an operator at the SHINE site, regardless of the chemical. This is because solids typically have very low vapor pressures, and solid particulates are heavier than vapor or gas molecules, and are therefore much less widely dispersed in air.

Only those chemicals exceeding the above screening criteria are included in the list of bounding toxic chemicals provided in Table 2.2-21.

For these chemicals, airborne dispersion was evaluated deterministically, using worst-case wind directions, and a temperature and wind speed with an annual exceedance probability of five percent. Only maximum concentration accidents were evaluated based on releases of the maximum expected amounts of chemicals. Maximum concentration-duration accidents were not evaluated because after shutting down the facility the operators do not need to take any other actions to assure facility safety. These deterministic evaluations were performed using ALOHA (Areal Locations of Hazardous Atmospheres), Version 5.4.4 (ALOHA, 2013).

The SHINE Control Room in this evaluation was assumed to have an air-exchange rate of 1.2 exchanges per hour.

Regulatory Guide 1.78 states that an air exchange rate of:

1. 0.015 per hour (i.e., 0.015 of the control room air by volume is replaced by atmospheric ambient air in one hour) is considered representative of a “tight” control room that has very low-leakage construction features and automatic isolation capabilities.
2. 0.06 per hour is considered representative of a control room that has normal leakage construction features and automatic isolation capabilities.
3. 1.2 per hour is considered representative of a control room with construction features that are not as efficient for leakage control and without automatic isolation capabilities.

A two-minute exposure to National Institute for Occupational Safety and Health (NIOSH) Immediately Dangerous to Life and Health (IDLH) concentrations was used as the threshold for uninhabitability. For chemicals with no defined IDLH limit, Protective Action Criteria (PAC) Level 2 limits were used, or the chemical was screened against qualitative toxicity information if no quantitative limits were available.

2.2.3.1.3.1 Pipelines

As discussed in Subsection 2.2.3.1.2.1, there are three bounding natural gas pipelines within 5 mi. (8 km) of the SHINE facility. Natural gas is predominantly methane. The toxicity hazard from methane is that of a simple asphyxiant, and there are no defined IDLH or Emergency Response Planning Guideline (ERPG) levels for methane. A cloud of methane would reach potentially explosive concentrations before displacing enough oxygen to cause asphyxiation. Therefore, the bounding hazard from natural gas is a potential explosion or fire, which was addressed in Subsection 2.2.3.1.2.1 and determined to not be a threat to the SHINE facility.

2.2.3.1.3.2 Waterway Traffic

There is no navigable waterway within 5 mi. (8 km) of the SHINE facility.

2.2.3.1.3.3 Highways

Table 2.2-21 provides toxic materials potentially transported on US 51 and I-90/39.

The closest SHINE safety-related area is located approximately 0.22 mi. (0.35 km) from US 51, and approximately 2.1 mi. (3.4 km) from I-90/39. For this analysis, these distances were also used as the distance from US 51 and I-90/39, respectively, to the SHINE Control Room.

The hazardous chemicals evaluated were primarily based on those chemicals identified in 2010 Tier II reports in Rock County, Wisconsin (Wisconsin Emergency Management, 2011). The selection of mobile sources for an analysis of potential impact to the SHINE Control Room was based on: (1) the mobile sources of hazardous chemicals described in Table 2.2-4; (2) stationary sources within 5 mi. where deliveries or shipments could be transported on local roads; (3) large quantities of stationary sources elsewhere in the county where deliveries or

shipments could be transported on major roads or rail lines; and (4) direct communication with facilities regarding their types, quantities, and frequencies of shipments.

If a chemical is known to be in a tank (i.e., chemicals transported by rail or on US 51 or I-90/39), the dispersion is modeled in ALOHA as a tank source, with the tank volume set to accommodate the entire mass of the chemical. A hole in the bottom of the tank is sized so that the entire tank inventory is released in one minute (minimum release time for ALOHA), and if the chemical is a liquid, forms a puddle that spreads to a maximum area that can be modeled, as determined by ALOHA. Ground type is the ALOHA default soil and ground temperatures are set to ambient conditions.

Chemicals transported by truck were modeled as release of 50,000 lbs of the chemical except for sodium bisulfite and chlorine.

Chlorine is shipped in 150-lb cylinders, one-ton containers, cargo tankers (15-22 tons), and up to 90-ton rail cars. The only users of chlorine within 5 mi. (8 km) of the SHINE site are the City of Janesville and the City of Beloit water utilities. The chlorine used is obtained in standard 150-lb cylinders (City of Beloit, 2015 and City of Janesville, 2015a). The maximum amount of chlorine at any one site is 900 lbs. Therefore, a release of chlorine on US 51 is considered only for the case of the failure of one 150-lb cylinder. Chlorine releases on I-90/39 were considered for standard-size shipment containers (one ton containers (2,000 lbs) and 22 ton cargo tankers (44,000 lbs)).

Sodium bisulfite (which could generate sulfur dioxide) was modeled as a 15,000 lb release from US 51, since 15,000 lbs is the maximum inventory size of any current stationary location of sodium bisulfite (City of Janesville, 2015b).

Of the releases analyzed deterministically, only the following were found to be a potential hazard to the SHINE Control Room:

- Ammonia (50,000 lbs) from US 51
- Chlorine (44,000 lbs) from I-90/39
- Propylene oxide (50,000 lbs) from I-90/39
- Sodium bisulfite (15,000 lbs) from US 51

These mobile sources of chemicals were evaluated using a simple probabilistic model, based on shipment or inventory information from local users of those chemicals. The acceptance criteria for releases evaluated in this manner is 10^{-6} releases per year because the resultant low levels of radiological risk are considered acceptable.

The following equation was used to determine the maximum number of shipments past the SHINE facility before the probability of a release exceeded 1×10^{-6} per year.

$$R_{haz} = P_{spill} \times R_{accident} \times P_{weather} \times D_{trip}$$

Where:

R_{haz} = Rate of hazards per vehicle trip near the site (hazardous spills/trip)
P_{spill} = Probability of the spill size (spills/accident)
$R_{accident}$ = Rate of accidents (accidents/vehicle mile)
$P_{weather}$ = Adverse wind direction probability (hazardous weather conditions at the site)
D_{trip} = Hazardous trip length, the total number of miles that a vehicle travels past the SHINE site each trip where an accident could result in a hazardous condition (vehicle miles/trip)

Chlorine, propylene oxide and sodium bisulfite were eliminated as a hazard to the SHINE Control Room using this probabilistic method.

For chlorine, 53 cargo tanker shipments per year on I-90/39 past the SHINE site are required to exceed a release frequency of 1 x 10^{-6}. Without large producers or users of chlorine in the county, there are expected to be fewer than 53 cargo tanker shipments per year, and this release scenario therefore is not considered a hazard to the SHINE Control Room.

For propylene oxide, 58 truck shipments per year on US 51 are required to exceed a release frequency of 1 x 10^{-6}. Since the only user of propylene oxide within 5 mi. (Abitec Corporation) that receives shipments via truck has 6 shipments per year (Abitec Corporation, 2015), propylene oxide is not considered a hazard to the SHINE Control Room.

For sodium bisulfite, 553 truck shipments per year on US-51 are required to exceed a release frequency of 1 x 10^{-6}. Since the only current user of sodium bisulfite within 5 mi. has a reported storage quantity of 15,000 lbs, it is very unlikely they send or receive 553 shipments per year. Sodium bisulfite is therefore not considered a hazard to the SHINE Control Room.

A simple probabilistic analysis is not sufficient to eliminate ammonia from consideration as a hazard to the SHINE site. However, in the most limiting case of the closest, maximum inventory release, worst case wind directions, and five percent annual exceedance maximum wind speeds and atmospheric stability classes, the indoor toxicity limit is approached approximately one minute after the release, and outdoor concentrations begin to rise about 20 seconds after the release. Although there are only approximately 40 seconds between potential detection on-site and reaching the IDLH limit in the SHINE Control Room, the IDLH limit can be tolerated for 2 minutes without physical incapacitation. Therefore, the operators will be able to place the facility in a safe condition prior to the need to use personal protective equipment.

2.2.3.1.3.4 On-Site Chemicals

On-site chemical hazards are evaluated in Subsection 13b.3.2. This evaluation included exposure concentrations for workers located 328 ft. (100 m) downwind of a potential spill. The worker exposure calculations are considered representative of exposure to Control Room personnel. The results of this evaluation are presented in Table 13b.3-2.

These concentrations are calculated for a release of the largest container of each chemical on-site and are conservative, since they do not consider the mitigating effects of isolation dampers or the SHINE Control Room air exchange rate, with the exception of nitric acid and n-dodecane, which are only calculated for chemical amounts associated with licensed material. A release of 6,229 lbs of nitric acid (from the Acids Room, as indicated in Table 13b.3-1) or 1,033 lbs of n-dodecane (from the Caustics Room, as indicated in Table 13b.3-1), not associated with licensed materials, results in worker exposure concentrations of 15 ppm or 0.023 ppm, respectively. These concentrations are below the PAC-2 limits of 24 ppm for nitric acid and 0.031 ppm for n-dodecane, and below the IDLH limit of 25 ppm for nitric acid. Since the worker concentrations are below the PAC-2 levels for all chemicals considered, on-site chemical releases are not a hazard to the SHINE Control Room.

2.2.3.1.3.5 Nearby Facilities and Railways

Table 2.2-21 provides stationary sources of bounding toxic chemicals located within 5 mi. (8 km) of the SHINE site and bounding toxic chemicals potentially transported by rail near the facility.

The hazardous chemicals evaluated were primarily based on those chemicals identified in 2010 Tier II reports in Rock County, Wisconsin. Direct communication with individual facilities was used to augment the stationary source information identified in the 2010 Tier II reports.

Releases from rail lines are set at 1.6 mi., which is the distance of the nearest approach of the Union Pacific Railway to the SHINE facility. Chemicals stored or situated at distances greater than 5 mi. from the plant need not be considered because, if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that either toxic limits will never be reached or there would be sufficient time for the control room operators to take appropriate action.

The Tier II Report was reviewed for other chemicals used within in the region, not necessarily within 5 mi. of the SHINE site, in a significant quantity (i.e., over 50,000 lbs), such that they may be frequently shipped by rail near the SHINE site. Chemicals already determined to not be hazardous based on vapor pressure or toxicity were not included.

Based on this review, acrylonitrile, which is used by a chemical manufacturer located greater than 5 mi. from the SHINE site, was also considered for analysis based on the large amounts used by this manufacturer.

Assumptions for a rail line tank release were the same as used for highway tank release, as described in Subsection 2.2.3.1.3.3. The tank size for a rail line release was set to 30,000 gallons, and converted to an equivalent mass based on the estimated density of each material.

No release from a stationary source was determined to be a hazard to the SHINE Control Room, as shown in Table 2.2-21. For rail line releases, only ammonia had the potential to exceed toxicity levels in the SHINE Control Room under 5 percent annual exceedance probability worst case meteorological conditions. A probabilistic evaluation was not undertaken

for rail shipments of ammonia, since this release was bounded by a postulated tanker truck release, as discussed in Subsection 2.2.3.1.3.3.

2.2.3.1.3.6 Toxic Chemical Related Impacts Affecting the Design

Of the chemicals evaluated, only an ammonia release could have a greater than 10^{-6} per year potential to result in an uninhabitable Control Room, based on a simple probabilistic analysis. For the closest ammonia release, the evaluation shows that the Control Room Operators would be able to shut down the facility (i.e., have at least two minutes) by manually tripping the TSVs prior to needing to use personal protective equipment. This single action ensures:

- Target solution is drained to the criticality-safe dump tank(s);
- Decay heat from the target solution is removed via conduction through the dump tank(s) walls to the light water pool; and
- Hydrogen buildup in the primary system boundary is controlled via the target solution vessel off-gas system.

These actions will maintain the target solution in a safe shutdown condition. There are no radiological consequences to the workers or the public due to an off-site toxic gas release that affects the facility.

Table 2.2-4 Hazardous Chemicals Potentially Transported on Highways within 8 Km (5 Mi.) of the Project Site

Chemical	Quantity (lbs)	Highway	DIstance to SHINE (mi.)
Ammonia	50,000	US-51	0.22
Asphyxiant Model (Carbon Monoxide)	50,000	US-51	0.22
Bounding Amide (Formamide)	50,000	US-51	0.22
Chlorine[(a)]	~~900~~150	US-51	0.22
~~Ethylene Oxide~~	~~50,000~~	~~US-51~~	~~0.22~~
Diesel	50,000	US-51	0.22
Ethyl Alcohol	50,000	US-51	0.22
Gasoline	50,000	US-51	0.22
Hydrogen Peroxide	50,000	US-51	0.22
Isopropanol	50,000	US-51	0.22
n-Butyl Alcohol	50,000	US-51	0.22
Propane	50,000	US-51	0.22
Propylene Oxide	50,000	US-51	0.22
Sodium Bisulfite (Sulfur Dioxide)[(a)]	15,000	US-51	0.22
Styrene	50,000	US-51	0.22
Acetone	50,000	I-90/39	2.1
Chlorine[(a)]	~~50,000~~2,000	I-90/39	2.1
Chlorine[(a)]	44,000	I-90/39	2.1
Hydrogen	3,300	I-90/39	2.1
Methyl Acetate	50,000	I-90/39	2.1
n-Heptane	50,000	I-90/39	2.1
Nitric Acid	50,000	I-90/39	2.1
Sodium Bisulfate (Sulfur Dioxide)[(a)]	50,000	I-90/39	2.1
Sodium Hypochlorite (Chlorine)	50,000	I-90/39	2.1

a) Chlorine and sodium bisulfate were evaluated for multiple release scenarios in Subsection 2.2.3.1.3.3.

Table 2.2-5 Airports Located within 10 Mi. (16 Km) of the SHINE Site Center Point Including Airport Operations at Each Airport

Airport[(a)]	Distance from SHINE Facility Center Point in Statute Miles [(d)]	Number of Operations in 2010	Projected Number of Operations in 2040	200d2 Screening Criterion[(f)]
Southern Wisconsin Regional Airport	0.39	48,387	~~56,818~~ 52,953	(c)
Beloit Memorial Hospital Heliport	5.3	Sporadic[(b)]	N/A	14,551
Hacklander Airport	6.86	Sporadic[(b)]	N/A	24,667
Melin Farms Airport	7.92	Sporadic[(b)]	N/A	32,490
Archie's Seaplane Base[(d)]	8.17	Sporadic[(b)]	N/A	34,576
Beloit Airport	9.15	16,790[(a)]	N/A	43,368
Turtle Airport	9.85	Sporadic[(b)]	N/A	50,257

a) Based on 46 operations per day times 365 days per year.
b) Operations of private airports and those with no aircraft stationed at the airport are considered sporadic.
c) Probabilistic hazard analysis needed because the distance is less than 5 miles.
d) This private airport does not appear to be in operation since operational data for this airport dates from 1991. It is, however, listed for completeness.
e) The greater size of aircraft using the Southern Wisconsin Regional Airport, greater number of operations at the Southern Wisconsin Regional Airport, and the closer distance from the SHINE facility to the Southern Wisconsin Regional Airport provides a bounding analysis that renders separate analysis of the Mercy Hospital unnecessary.
f) Airports considered in analysis if the airport is within 5 mi. (8 km) of the SHINE site, or if, for airports located a distance of between 5 mi. (8 km) and 10 mi. (16 km) from the SHINE site, an airport has annual operations of more than $200d^2$ (where d is the distance to the SHINE facility in kilometers).

Table 2.2-8 DOE Input Values for CONUS Average

$N_jP_jf_j(x,y)$ Values

	NjPjfj(x,y) Value[a] (1/mi2)
Air Carrier	4E-7
Air Taxi	1E-6
General Aviation	2E-4
Small Military	4E-6
~~Large Military~~ Military Helicopter	~~2E-7~~ 2.5E-5

Effective Area Input Values

	WS[b]	cot Φ [c]	S[d](ft.)
Air Carrier	98	10.2	1440
Air Taxi	59	10.2	1440
General Aviation	50	8.2	60
Small Military	110	10.4	447
~~Large Military~~ Military Helicopter	~~223~~ 79[e]	~~9.7~~ 0.58	~~780~~ 0

a) Reference (DOE, 2006), Tables B-14, ~~and~~ B-15, and B-43
b) Reference (DOE, 2006), Table B-16
c) Reference (DOE, 2006), Table B-17
d) Reference (DOE, 2006), Table B-18
e) Wingspan is for Sikorsky CH-53E/MH-53E. DOE-STD-3014-96 only contained wingspan information for general aviation helicopters, which are generally smaller than military helicopters.

Table 2.2-9 Calculated Effective Areas of Safety-Related Structures (sq. mi.) by Aircraft Type Used for the Evaluation of Airways and Airports

Aircraft Type	Effective Area (sq. mi.)
Air Carrier	~~0.05032~~0.0378
Air Taxi	~~0.04616~~0.0315
General Aviation	~~0.01764~~0.0095
Small Military	~~0.03211~~0.0221
~~Large Military~~ Military Helicopter	~~0.04672~~ 0.0028

Table 2.2-10 ~~Calculated Effective Areas of Safety-Related Structures (sq. mi.) by Aircraft Type Used for the Evaluation of Airports~~This table number not used

~~Aircraft Type~~	~~Effective Area (sq. mi.)~~
~~Air Carrier~~	~~0.05032~~
~~General Aviation~~	~~0.01764~~
~~Large Military~~	~~0.04672~~

Table 2.2-13 ~~Maximum~~Number of Operations per Year at ~~the~~ Southern Wisconsin Regional Airport ~~for the Years 2010 through 2040~~

Aircraft Type	~~Maximum~~Number of Operations
Air Carrier (itinerant)	~~104~~68
Air Taxi (itinerant)	~~5,962~~6,908
General Aviation (itinerant)	~~25,007~~23,469
Military (itinerant ~~operation~~)	~~352~~390
Civil (local)	~~25,958~~22,253
Military (local ~~operation~~)	~~1,126~~617

Table 2.2-14 Aircraft Operation by Aircraft Type on Each Runway

Aircraft Type	Runway Operations					
	RW 14	**RW 32**	**RW 4**	**RW 22**	**RW 18**	**RE 36**
Air Carrier	~~15~~10	~~22~~14	~~31~~20	~~21~~14	~~8~~5	~~8~~5
~~General Aviation~~	~~7,970~~	~~11,955~~	~~17,078~~	~~11,385~~	~~4,270~~	~~4,270~~
~~Military~~	~~207~~	~~310~~	~~443~~	~~296~~	~~111~~	~~111~~
Small Non-Military	7,368	11,053	15,789	10,526	3,947	3,947
Small Military	1	2	3	2	1	1
Military Helicopter	140	209	299	199	75	75

Table 2.2-15 Total Crash Probability

	~~Small Aircraft~~Large Non-Military Aircraft	~~Large Aircraft~~Small Non-Military Aircraft	Military Aircraft
Airport	~~5.387E-04~~2.9E-07	~~4.490E-06~~2.6E-04	1.5E-07
Airways	~~3.703E-06~~1.5E-08	~~2.947E-08~~1.9E-06	1.6E-07
Total	~~5.424E-04~~3.0E-07	~~4.520E-06~~2.6E-04	3.1E-07

Table 2.2-21 Bounding Toxic Chemical Hazards within 5 Mi. (8 Km) of the SHINE Site (Sheet 1 of 3)

Chemical	Location	Distance (mi.)	Mass (lbs)	Disposition
Polymer dispersion (1, 3-butadiene)	Humane Manufacturing	1 mi.	58,800 lbs	No Hazard
Polymer dispersion (benzene)	Humane Manufacturing	1 mi.	58,800 lbs	No Hazard
Asphyxiant Model (carbon monoxide)	Linde Merchant Production	2 mi.	5,000,000 lbs	No Hazard
Bounding Amide (Formamide)	Abitec Corporation and Evonik Goldschmidt (Bounding Case)	2 mi.	640,000 lbs	No Hazard
Bounding Amine (diethylamine)	Abitec Corporation and Evonik Goldschmidt (Bounding Case)	2 mi.	640,000 lbs	No Hazard
Bounding Amine (n-Butylamine)	Abitec Corporation and Evonik Goldschmidt (Bounding Case)	2 mi.	640,000 lbs	No Hazard
Ethylene Oxide	Abitec Corporation	2 mi.	440,000 lbs	No Hazard
Isopropanol	Abitec Corporation	2 mi.	185,800 lbs	No Hazard
Oxygen	Linde Merchant Production	2 mi.	2,150,000 lbs	No Hazard
Volatile Amine (cyclohexylamine)	WI School for the Visually Handicapped	2 mi.	300 lbs	No Hazard
Benzyl acetate	Evonik Goldschmidt	3 mi.	12,321 lbs	No Hazard
Chlorine	Janesville Pump Station #12	3 mi.	900 lbs	No Hazard
Ethyl Alcohol	Evonik Goldschmidt	3 mi.	168,000 lbs	No Hazard
Hydrogen Peroxide	Evonik Goldschmidt	3 mi.	60,000 lbs	No Hazard
Methyl Chloride	Evonik Goldschmidt	3 mi.	320,000 lbs	No Hazard
n-Heptane	Evonik Goldschmidt	3 mi.	125,000 lbs	No Hazard
Sodium Bisulfite (as sulfur dioxide)	Evonik Goldschmidt	3 mi.	15,000 lbs	No Hazard
Sodium Chlorite (as chlorine dioxide)	Evonik Goldschmidt	3 mi.	14,000 lbs	No Hazard
Styrene	Monterey Mills	3 mi.	225,280 lbs	No Hazard
Volatile Amine (DMAPA)	Evonik Goldschmidt	3 mi.	12,045 lbs	No Hazard

Table 2.2-21 Bounding Toxic Chemical Hazards within 5 Mi. (8 Km) of the SHINE Site (Sheet 2 of 3)

Chemical	Location	Distance (mi.)	Mass (lbs)	Disposition
Propylene Oxide	Evonik Goldschmidt and Rail (Bounding Case)	1.6 mi.	360,000 lbs	No Hazard
Acrylonitrile	Rail	1.6 mi.	199,852 lbs	No Hazard
Ammonia	Rail	1.6 mi.	150,054 lbs	Additional Evaluation
Bounding Amide (Formamide)	Rail	1.6 mi.	282,214 lbs	No Hazard
Bounding Amine (diethylamine)	Rail	1.6 mi.	175,391 lbs	No Hazard
Bounding Amine (n-Butylamine)	Rail	1.6 mi.	183,538 lbs	No Hazard
Ethylene Oxide	Rail	1.6 mi.	216,317 lbs	No Hazard
Methyl Chloride	Rail	1.6 mi.	227,428 lbs	No Hazard
Vinylidene Chloride	Rail	1.6 mi.	300,966 lbs	No Hazard
Ammonia	Truck (US 51)	0.22 mi.	50,000 lbs	Additional Evaluation
Asphyxiant Model (Carbon Monoxide)	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Bounding Amide (Formamide)	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Chlorine	Truck (US 51)	0.22 mi.	150 lbs	No Hazard
Ethyl Alcohol	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Gasoline (as butane)	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Gasoline (as toluene)	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Hydrogen Peroxide	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Isopropanol	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
n-Butyl Alcohol	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Propane	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard
Propylene Oxide	Truck (US 51)	0.22 mi.	50,000 lbs	Additional Evaluation
Sodium Bisulfite (as sulfur dioxide)	Truck (US 51)	0.22 mi.	15,000 lbs	Additional Evaluation
Styrene	Truck (US 51)	0.22 mi.	50,000 lbs	No Hazard

Table 2.2-21 Bounding Toxic Chemical Hazards within 5 Mi. (8 Km) of the SHINE Site (Sheet 3 of 3)

Chemical	Location	Distance (mi.)	Mass (lbs)	Disposition
Acetone	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
Chlorine	Truck (I-90/39)	2.1 mi.	2,000 lbs	No Hazard
Chlorine	Truck (I-90/39)	2.1 mi.	44,000 lbs	Additional Evaluation
Hydrogen	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
Methyl Acetate	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
n-Heptane	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
Nitric Acid	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
Sodium Bisulfite (as sulfur dioxide)	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard
Sodium Hypochlorite (as chlorine)	Truck (I-90/39)	2.1 mi.	50,000 lbs	No Hazard

2.3.1.2.9 Snowpack and Probably Maximum Precipitation (PMP)

A 100-year return-period snowpack for the project site vicinity was derived by multiplying the 50-year return interval snowpack from Figure 7.1 of ASCE, 2006 by a factor which converts the 50-year return interval snowpack to a 100-year return-interval snowpack. Table C7-3 of ASCE, 2006 suggests that an appropriate factor is 1.22 (i.e., the 50-year value divided by the factor of 0.82 listed in Table C7-3).

The estimated 50-year interval snowpack for the project site from Figure 7.1 of ASCE 2006 is 25 pounds per square foot (lb/ft^2) (122.1 kilograms per square meter [kg/m^2]) ~~in. (63.5 cm)~~. The resulting estimated 100-year return interval snow pack for the project site is 30.5 ~~in.~~lb/ft^2 (30.5 ~~in.~~lb/ft^2 = 1.22 x 25 ~~in.~~lb/ft^2) (~~77.5 cm~~148.9 kg/m^2).

The weight of the 48-hour PMP for the project site vicinity was derived by multiplying the 48-hour PMP (in inches) from Figure 21 of USDOC, 1978 by the weight of one inch of water (one inch of water covering one square foot weighs 5.2 lb (2.4 kg)).

The estimated 48-hour PMP for the project site from Figure 21 of USDOC, 1978 is 34 in. (86.4 cm). The resulting estimated weight of the 48-hour PMP for the project site is 176.8 ~~pounds per square feet (~~lb/ft^2~~)~~ (863.2 ~~kilograms per square meter [~~kg/m^2~~]~~ (176.8 lb/ft^2 = 34 in. x 5.2 lb/ft^2).

2.3.1.2.10 Design Dry Bulb and Wet Bulb Temperatures

Site design basis dry bulb temperatures (DBTs) and wet bulb temperatures (WBTs) are defined for the project site and its climate area. Those include the following statistics:

a. Maximum DBT with annual exceedance probability of 0.4 percent
b. Mean coincident WBT (MCWB) at the 0.4 percent DBT
c. Maximum DBT with annual exceedance probability of 2.0 percent
d. MCWB at the 2.0 percent DBT
e. Minimum DBT with annual exceedance probability of 0.4 percent
f. Minimum DBT with annual exceedance probability of 1.0 percent
g. Maximum WBT with annual exceedance probability of 0.4 percent
h. Maximum DBT with annual exceedance probability of 5 percent
i. Minimum DBT with annual exceedance probability of 5 percent
j. 100-year return maximum annual DBT
k. MCWB at the 100-year return maximum annual DBT
l. 100-year return maximum annual WBT
m. 100-year return minimum annual DBT

Statistics for (a)-(g) are readily available from ASHRAE, 2009. Since those statistics are available from a well-known reference, no additional data analysis is required. ASHRAE, 2009 includes values for the following stations in the project site climate region: Fond du Lac, Wisconsin; Madison, Wisconsin; Rockford, Illinois; and DuPage County Airport, Illinois. These stations represent climatic conditions in the northern, central and southern portions of the climate region, respectively (Figure 2.3-16). Worst-case (bounding) values for (a)-(g) are selected from those four stations. To maintain thermodynamic consistency between DBT and coincident WBTs, DBT/MCWB pairs are retained for a single station. The resulting statistics are listed in Table 2.3-14.

$$q = k \times \frac{\Delta h}{\Delta I} \qquad \text{(Equation 2.4-4)}$$

where q is the flux, k is the hydraulic conductivity, Δh is the difference of head over ΔI distance. However, water moves only through a portion of space (between the grains of the soil); therefore q has to be divided by porosity to get water velocity:

$$v = \frac{q}{n} \qquad \text{(Equation 2.4-5)}$$

where n is porosity, v is the velocity.

By knowing the distance between the source and the discharge point, an estimate can be made. This calculation is conservative in the following ways:

- Particles were released at the groundwater table, so the unsaturated zone had not been considered in the calculations due to the limited information available.
- The model is one-dimensional, so three-dimensional development of contaminant plume had not been modeled; pathways run straight from the site to the discharge points or areas.
- Important transport processes (adsorption, dispersion, diffusion, decay, dilution) were not involved in the calculations – only advective travel times have been estimated.
- Homogeneous, high conductivity values have been assigned to the model, no parameter heterogeneity has been considered.
- No dilution is considered along the bed of the Rock River and within the groundwater system.

Based on these assumptions, the calculation travel times and concentrations bound those that would be involved in an actual event.

A summary of parameters used for advective travel time estimations in the saturated zone is presented in Table 2.4-13. The calculations have been carried out for assumed release locations west and south to the Rock River, and to water supply wells MF461 and UJ792, identified as the nearest off- site features applicable for groundwater pathways. Using Equations 2.4 4 and 2.4 5, Table 2.4 13 provides advective groundwater travel times for two conservative cases: expected hydraulic conductivity and porosity, and unfavorable hydraulic conductivity and porosity. Note that all cases use very conservative assumptions. The river release location uses channel base, and well assumes the maximum reported drawdown.

2.4.11.3 Characteristics that Affect Transport

In transport calculations, the following processes are considered:

- Advective transport
- Dispersion
- Dilution
- Sorption
- Decay
- Diffusion

Table 2.4-13 Summary of Parameters Used for Advective Travel Time Estimations

Model Version	Permeability and Porosity Assumptions	Coordinates for Source at SHINE Facility[(a)]		Coordinates of Assumed Release Location[(b)]		Distance (ft. [m])	Head at Assumed Source[(c)] (ft. [m]) NAVD-88)	Head at Assumed Release Location[(d)] (ft. [m]) NAVD-88)	Effective Transport Porosity[(e)] (%)	Hydraulic Conductivity[(f)] (ft./sec [m/s])	Adjective Travel Time[(g)] (yrs)
		Northing (ft. [m])	Easting (ft. [m])	Northing (ft. [m])	Easting (ft. [m])						
Pathway to Rock River (West)	Expected	247763.2 (75518.2)	492642.3 (150157.2)	247763.2 (75518.2)	481715.3 (146826.8)	10,927 (3,331)	766 (233.5)	738 (224.9)	30	0.0045 (0.0014)	9.0
Pathway to Rock River (West)	Conservative	247763.2 (75518.2)	492642.3 (150157.2)	247763.2 (75518.2)	481715.3 (146826.8)	10,927 (3,331)	766 (233.5)	738 (224.9)	10	0.0083 (0.0025)	1.6
Pathway to Rock River Tributary (South)	Expected	247763.2 (75518.2)	492642.3 (150157.2)	247763.2 (75518.2)	481715.3 (146826.8)	12605 (3,842)	766 (233.5)	738 (224.9)	30	0.0045 (0.0014)	26
Pathway to Rock River Tributary (South)	Conservative	247763.2 (75518.2)	492642.3 (150157.2)	247763.2 (75518.2)	481715.3 (146826.8)	12605 (3,842)	766 (233.5)	738 (224.9)	10	0.0083 (0.0025)	4.7
Pathway to Nearest Well “Receptor” (MF461)	Expected	247763 (75518)	492642 (150157)	249603 (76078.9)	490510 (149507)	2816 (858)	766 (233.5)	738 (224.9)	30	0.0045 (0.0014)	1.4
Pathway to Nearest Well “Receptor” (MF461)	Conservative	247763 (75518)	492642 (150157)	249603 (76078.9)	490510 (149507)	2816 (858)	766 (233.5)	738 (224.9)	10	0.0083 (0.0025)	0.3
Pathway to Low Head DNR Well “Receptor” (UJ792)	Expected	247763.0 (75518.2)	492642.0 (150157.3)	246533.7 (75143.5)	491107.1 (149689.5)	1,967 (600)	766 (233.5)	660 (201)	30	0.0045 (0.0014)	0.1
Pathway to Low Head DNR Well “Receptor” (UJ792)	Conservative	247763.0 (75518.2)	492642.0 (150157.3)	246533.7 (75143.5)	491107.1 (149689.5)	1,967 (600)	766 (233.5)	660 (201)	10	0.0083 (0.0025)	0.01

a) SHINE source coordinate calculated as center of site.
b) Release coordinates for Rock River (West) and South) are calculated assuming a straight line from the SHINE facility.
c) Head at SHINE facility based on maximum head measured during monitoring period (Table 2.4-8).
d) Head at Rock River (West) and Rock River Tributary (South) release locations based on channel bottom (Table 2.4-1 and 2.4-2). Head at Well MF461 calculated based on minimum head reported in WDNR, 2012b.
e) High (Expected) and Low (Conservative) Transport Porosity Values from Gaffield et al, 2002.
f) Hydraulic Conductivity based on the Average Hydraulic Conductivity from Slug Tests (Table 2.4-10). Conservative case is highest Hydraulic Conductivity from Slug Tests (Table 2.4-10).
g) Advective Travel Time calculated from Darcy's Law (Bear, 1972).

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3.4.2.6.3 Site Design Parameters

The following subsections provide the site-specific parameters for the design of the SHINE facility.

3.4.2.6.3.1 Soil Parameters

The soil parameters for the SHINE facility are provided below.

- ~~Minimum~~Net allowable static bearing capacity ~~demand~~ for 2-ft. (0.61-m) wide strip footings: 4000 pounds per foot square (psf) (191.5 kPa).
- ~~Minimum~~Net allowable static bearing capacity ~~demand~~ for 6-ft. (1.8-m) wide ~~strip~~spread footings: 6000 psf (287.3 kPa).
- ~~Minimum~~Net allowable static bearing capacity ~~demand~~ for footings 22 ft. (6.7 m) below grade: 8000 psf (383.0 kPa).
- Minimum shear wave velocity: 459 ft/sec (140 m/s).
- Poisson's Ratio: 0.4.
- Unit Weight: ~~125~~121 pounds per cubic foot (lb/ft^3) (~~2002~~1938 kilograms per cubic meters [kg/m^3]).

3.4.2.6.3.2 Maximum Ground Water Level

- 50 ft. (15.2 m) below grade level.

3.4.2.6.3.3 Maximum Flood Level

- Section 2.4 describes the PMP.
- Section 2.4 describes the probable maximum flood (PMF).

3.4.2.6.3.4 Snow Load

- Snow load: 30 psf (1.44 kPa) (50-year recurrence interval).
- A factor of 1.22 is used to account for the 100-year recurrence interval required.

3.4.2.6.3.5 Design Temperatures

- The winter dry-bulb temperature (-7 degrees Fahrenheit) is consistent with the 0.4 percent and 1 percent minimum dry-bulb temperatures (-9.1 degrees Fahrenheit and -2.9 degrees Fahrenheit respectively).
- The summer dry bulb temperature (88 degrees Fahrenheit) is consistent with the 0.4 percent and 2 percent maximum dry-bulb temperatures (91.5 degrees Fahrenheit and 85.8 degrees Fahrenheit respectively).

3.4.2.6.3.6 Seismology

- SSE Peak Ground Acceleration (PGA): 0.20 g (for both horizontal and vertical directions).
- SSE Response Spectra: Per Regulatory Guide 1.60.
- SSE Time History: Envelope SSE response spectra in accordance with SRP Section 3.7.1.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 3 of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(6) *Emergency capability*. The design must provide for emergency capability to maintain control of: (i) Licensed material and hazardous chemicals produced from licensed material; (ii) Evacuation of on-site personnel; and (iii) Onsite emergency facilities and services that facilitate the use of available offsite services.	As Applied and Means of Compliance SHINE will develop and maintain emergency procedures for each area that contains licensed material and hazardous chemicals produced from licensed material. These procedures include provisions for the evacuation of all personnel to an area of safety in the event of an alarm. The procedures also include conducting drills to familiarize personnel with the evacuation plan, designation of responsible individuals and organizations for the disposition of licensed material, evaluation of the cause of the alarm, and the placement of facilities, systems, instruments, tools and materials for use in such an emergency. The SHINE facility includes a Fire Brigade and Hazmat Response Area that provides a command center for the use of available off-site emergency services and personnel.
(7) *Utility services*. The design must provide for continued operation of essential utility services.	As Applied and Means of Compliance ~~The SHINE facility provides a standby diesel generator for asset protection of selected systems.~~An on-site electric power system (defined to be the SHINE facility provided power supply) and an off-site (defined to be the public utility transmission network supplied substations) electric power system are provided to permit functioning of structures, systems, and components important to safety. The functions for each system (assuming the other system is not functioning) are to provide sufficient capacity and capability to ensure that the RPF processes remain in a safe condition and confinement integrity and other vital functions are maintained in the event of postulated accidents. The on-site electric power supply systems include a safety-related UPSS, a commercial grade diesel generator system, and the on-site electric distribution system. The UPSS has sufficient independence, redundancy, and testability to perform its safety functions assuming a single failure. Each of the redundant UPSS circuits are designed to be available within a few seconds following a loss of all off-site electric power, to ensure that safety systems requiring electrical power (such as RAMS and CAAS) can perform their safety functions. The diesel generator system performs no safety functions but provides an additional degree of defense-in-depth to the electric power system. Portions of the on-site electric distribution, those circuits which operate in conjunction with the UPSS, are safety-related. The SHINE facility receives a single physically off-site independent power circuit. This off-site power circuit consists of two power feeds connected to two local outdoor transformers. Provisions are included to minimize the probability of losing electric power from the UPSS as a result of or coincident with, the loss of power from the transmission network. Refer to Section ~~8b~~6b and Chapter 8 for a detailed ~~information~~discussion.

Table 3.5b-1 Baseline and General Design Criteria for Radioisotope Production Facility (Sheet 3a of 5)

BASELINE DESIGN CRITERIA 10 CFR 70.64	As Applied to SHINE
(8) I*nspection, testing, and maintenance*. The design of items relied on for safety must provide for adequate inspection, testing, and maintenance, to ensure their availability and reliability to perform their function when needed.	As Applied and Means of Compliance SHINE has provided access and controls for testing, maintenance and inspection of SR SSCs. This is a general practice that is applied differently throughout the facility. Refer to Sections 4b, 6b, 7b and 9b for detailed information.
(9) *Criticality contro*l. The design must provide for criticality control including adherence to the double contingency principle.	As Applied and Means of Compliance SHINE includes criticality-safe by geometry process vessels, criticality-safe storage, as well as other passive engineering and administrative controls. Compliance with the requirements of criticality control including adherence to the double-contingency principle are described in detail in Section 6b.3.

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~~12 to 20 in. (~~10.0~~30 to ~~30.0~~50 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

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~~will 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.

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

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:

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

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~~uranium 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.

List of Tables

6a2.2 IRRADIATION FACILITY ENGINEERED SAFETY FEATURES DETAILED DESCRIPTION

The ESFs are passive or active features designed to mitigate the consequences of accidents and to keep the radiological and chemical exposures to the public, the facility staff, and the environment within acceptable values. This section provides the details of design, initiation, and operation of ESFs that are provided to mitigate the design basis accidents (DBAs) tabulated in Section 6a2.1.

According to the Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, accident-initiating events (IEs) and scenarios, the following design basis accidents are to be addressed for the irradiation facility (IF):

a. Insertion of excess reactivity/inadvertent criticality
b. Reduction in cooling
c. Mishandling or malfunction of target solution
d. Loss of off-site power
e. External events
f. Mishandling or malfunction of equipment affecting the PSB
g. Large un-damped power oscillations (fuel temperature/void-reactivity feedback)
h. Detonation and deflagration in PSB
i. Unintended exothermic chemical reactions other than detonation
j. Primary system boundary system interaction events
k. Facility-specific events
    - Inadvertent exposure to neutrons from neutron driver
    - Irradiation facility fires
    - Tritium Purification System Design Basis Accident

The ~~following~~IF design basis accidents (DBAs), the determination of which DBAs have ~~do not have~~ radiological consequences that require mitigation by ESFs~~:~~, and the bases for these determinations are listed in Table 6a2.2-1.

~~a. Insertion of excess reactivity/inadvertent criticality~~
~~b. Reduction in cooling~~
~~c. Loss of off-site power~~
~~d. External events~~
~~e. Large un-damped power oscillations (fuel temperature/void-reactivity feedback)~~
~~f. Detonation or deflagration in PSB~~
~~g. Unintended exothermic chemical reactions other than detonation~~
~~h. Primary system boundary system interaction events~~
~~i. Facility-specific events~~
    - ~~Inadvertent exposure to neutrons from neutron driver~~
    - ~~Irradiation facility fires~~

The three DBAs requiring ESFs to mitigate the consequences are identified in Table 6a2.1-1.

6a2.2.1 CONFINEMENT

6a2.2.1.1 Introduction

Confinement is a term used to describe the low-leakage boundary that surrounds radioactive materials released during an accident and the associated RCA ventilation system (RV) components. Confinement systems are designed to localize release of radioactive material to controlled areas in normal operational states and mitigate the consequences of DBAs. Radiation protection control features such as adequate shielding and RV minimize hazards normally associated with radioactive materials. The principal design and safety objective of the confinement systems is to protect on-site personnel, the public, and the environment. The second design objective is to minimize the reliance on administrative or complex active engineering controls to provide a confinement system that is as simple and fail-safe as reasonably possible.

The target solution vessel (TSV), TSV dump tank, TSV off-gas system (TOGS), and associated components act as the primary pressure boundary and are safety-related SSCs. Together they act as the primary fission product boundary. The confinement boundary of the IU cell and TOGS shielded cell encloses the PSB.

The tritium purification system (TPS) double-walled piping and TPS confinement system provides confinement of the tritium supplied to the IU. The confinement boundary of the TPS gloveboxes encloses the TPS.

Confinement of the IU cells is achieved through the RV, ESFAS, and the biological shielding provided by the steel and concrete structures comprising the walls, roofs, penetrations of the IU cell and TOGS shielded cell. Shielding of the IU cells and TOGS shielded cells is discussed in detail in Section 4a2.5.

Confinement of the TPS gloveboxes is provided by the RV, ESFAS, and biological shielding comprising the walls, roofs, and penetration seals of the TPS gloveboxes. The biological shielding is described in Section 4a2.5.

6a2.2.1.2 Confinement System and Components

The ventilation system serving the IU cells, TOGS shielded cells, and TPS gloveboxes includes components whose functions are designated as nonsafety-related and safety-related. The ductwork, the isolation dampers, and the filter trains of RVZ1 are designated as safety-related components. Refer to Table 6a2.2-~~1~~2 for a description of the system and component safety functions. Active confinement isolation components are required to operate as described below.

The IU cells, TOGS shielded cell, and TPS gloveboxes employ a combination passive-active confinement methodology. During normal operation, passive confinement is achieved through the contiguous boundary between the hazardous materials and the surrounding environment and is credited with confining the hazards generated as a result of DBAs.

This boundary includes the biological shield (created by the physical construction of the cell itself) and the extension of that boundary through the RCA Zone 1 (RVZ1) ventilation system. The extent of this passive confinement boundary extends from the upstream side of the intake

high efficiency particulate air (HEPA) filter to the final downstream HEPA filter prior to exiting the building. For the TPS the confinement also includes the double-walled piping.

In the event of a DBA that results in a release in the IU cell, TOGS shielded cell, or TPS glovebox, radioactive material would be confined by the biological shield and physical walls of the cell itself. Each line that connects directly to the IU cell, TOGS shielded cell, or TPS glovebox atmosphere and penetrates the IU cell, TOGS shielded cell, or TPS glovebox is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material. Confinement isolation valves on piping penetrating the IU cell, TOGS shielded cell, or TPS glovebox are located as close as practical to the confinement boundary and active isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

To mitigate the consequences of an uncontrolled release occurring within an IU cell, TOGS shielded cell, or TPS glovebox, as well as the off-site consequences of releasing noble fission products through the ventilation system prior to sufficient decay, the confinement barrier utilizes an active component in the form of bubble-tight isolation dampers (safety-related) on the inlet and outlet ventilation ports of each cell. This ESF effectively reduces the amount of ductwork in the confinement volume that needs to remain intact to achieve IU cell, TOGS shielded cell, or TPS glovebox confinement. These dampers close automatically (fail-closed) upon loss of power or receipt of a confinement isolation signal generated by the ESFAS. Following an initiating event, the ESFAS provides the confinement isolation signal that isolates the IU cells, TOGS shielded cell, or TPS glovebox. Refer to Subsection 7a2.5 for a description of the ESFAS.

A failure of the TPS outside the glovebox is mitigated by the TPS confinement system. The TPS confinement system uses isolation valves to stop a tritium leak outside the glovebox when a leak is detected.

Overall performance assurance of the active confinement components is achieved through factory and in-place testing. Duct and housing leak tests are performed in accordance with ASME N511 (ASME, 2008), with minimum acceptance criteria as specified in ASME AG-1 (ASME, 2009). Specific owner’s requirements with respect to acceptable leak rates are based on the safety analyses.

6a2.2.1.3 Functional Requirements

Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced.

Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence such that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions.

~~Mechanical~~ Safety-related mechanical, instrumentation, and electrical systems and components are designed to ensure that a single failure of an active component, in conjunction with an initiating event, does not result in the loss of the system’s ability to perform its intended safety functions. The single failure considered is a random failure ~~and any consequential failures in addition to the initiating event for which the system is required and any failures that are a direct or consequential result of the initiating event~~.

The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 (IEEE, 2001)and Regulatory Guide 1.53 in the application of the single-failure criterion.

6a2.2.1.4 Confinement Components

The following components are associated with the ~~secondary~~ confinement barrier of the IU cells, TOGS shielded cell, or TPS glovebox, as previously described. Their specific materials, construction, and installation and operating requirements are evaluated based on the safety analysis.

Bubble-tight isolation dampers, designed, constructed and tested in accordance with ASME AG-1, Section DA “Dampers and Louvers” (ASME, 2009):

- Maintain their functional integrity.
- Maintain their rated leak-tightness following a seismic event.
- Maintain their structural integrity under fan shut-off pressure.
- Provide bubble-tight isolation upon receipt of a control signal or, in the event of loss of actuator power, by closure of actuator.
- Provide bubble-tight isolation when using manual actuator or when locked closed with power actuator removed.
- Relay damper full-open and full-closed position for control and indication by the use of limit switches.

Dampers are butterfly type, blade and frame fabricated of heavy-gage stainless steel. Total leakage is based on bubble solution test as outlined in ASME AG-1 2009, Section DA-5141 (ASME, 2009).

Ventilation ductwork and ductwork support materials meet the requirements of ASME AG-1, Article SA-3000 “Materials”. Supports are designed and fabricated in accordance with the requirements of ASME AG-1 2009, Section SA “Ductwork” (ASME, 2009).

Details of the TPS confinement system will be developed in final design and provided in the FSAR.

Low leakage seals are provided on each penetration through the IU cell, TOGS shielded cell, and TPS glovebox. For systems open to the IU cell, TOGS shielded cell atmosphere, or TPS glovebox, redundant isolation valves are provided.

6a2.2.1.5 Engineered Safety Feature Test Requirements

Engineered safety features are periodically tested to ensure that ESF components maintain operability and can provide adequate confidence that the system performs satisfactorily in service during postulated events.

To the extent possible, the ESFAS and the confinement ESF whose operation it initiates are designed to permit testing during plant operation. Testing actuation devices and actuated equipment may be done individually or in groups to avoid negative impact to plant operations.

Table 6a2.2-1 Irradiation Facility Design Basis Accident Consequence Determination

Design Basis Accident	Identified Radiological Consequences	Subsection Containing Radiological Consequence Analysis
a. Insertion of excess reactivity/inadvertent criticality	No	13a2.2.2.6
b. Reduction in cooling	No	13a2.2.3
c. Mishandling or malfunction of target solution	Yes (Table 6a2.1-1)	13a2.2.4.6
d. Loss of off-site power	No	13a2.2.5
e. External events	No	13a2.2.6
f. Mishandling or malfunction of equipment affecting the PSB	Yes (Table 6a2.1-1)	13a2.2.7.6
g. Large un-damped power oscillations (fuel temperature/void-reactivity feedback)	No	13a2.2.8
h. Detonation and deflagration in PSB	No	13a2.2.9
i. Unintended exothermic chemical reactions other than detonation	No	13a2.2.10
j. Primary system boundary system interaction events	No	13a2.2.11
k. Facility-specific events • Inadvertent exposure to neutrons from neutron driver	No	13a2.2.12.1
• Irradiation facility fires	No	13a2.2.12.2
• Tritium Purification System Design Basis Accident	Yes (Table 6a2.1-1)	13a2.2.12.3.6

Table 6a2.2-~~1~~2 Irradiation Facility Confinement Safety Functions

System, Structure, Component	Description	Classification
RVZ1 IU cell isolation dampers	Provide confinement at IU cell and TOGS shielded cell boundaries	Safety-Related
ESFAS	Provides confinement isolation signal	Safety-Related
Isolation valves on piping systems	Provide confinement at IU cell and TOGS shielded cell boundaries, and TPS boundary	Safety-Related
IU cell and TOGS shielded cell including penetrations	Provide confinement	Safety-Related

Table 6b.1-1 Summary of RPF Design Basis ~~Events~~Accidents and ESF Provided for Mitigation

Engineered Safety Feature (ESF)	Radioisotope Production Facility Design Basis ~~Event~~ Accident Mitigated by ESF	SSCs which provide ESF	Detailed Description Section or Subsection
Confinement	• Critical equipment malfunction • Accidents with hazardous chemicals • RPF fire	• Hot cells including penetration seals • RCA ventilation system Zone 1 (including ductwork up to filters and filters) and Zone 2 • Bubble-tight isolation dampers • Tank vaults • Radiological integrated controls system (RICS) • Isolation valves on piping systems penetrating hot cells	6b.2.1

6b.2 RADIOISOTOPE PRODUCTION FACILITY ENGINEERED SAFETY FEATURES

The ESFs are passive or active features designed to mitigate the consequences of accidents and to keep the radiological and chemical exposures to the public, the facility staff, and the environment within acceptable values. This section provides the details of design, initiation, and operation of ESFs that are provided to mitigate the DBAs discussed in Section 6b.1. This includes chemical storage areas outside the RCA.

According to Chapter 13 of the Final ISG Augmenting NUREG-1537, the following DBAs are to be addressed for the RPF:

a. Critical equipment malfunction.
b. Inadvertent nuclear criticality in the RPF.
c. RPF fire.
d. Accidents with hazardous chemicals.
e. External events.

These DBAs encompass LOOP and operator errors (See Section 13b).

The ~~following~~RPF DBAs, the determination of which DBAs have~~do not have~~ consequences that require mitigation by ESFs~~:~~, and the bases for these determinations are listed in Table 6b.2-1.

~~a. Inadvertent nuclear criticality in the RPF.~~
~~b. RPF fire.~~
~~c. External events.~~

The ~~two~~three DBAs requiring ESFs to mitigate consequences are identified in Table 6b.1-1.

6b.2.1 CONFINEMENT

6b.2.1.1 Introduction

Confinement describes the low-leakage boundary surrounding radioactive or hazardous chemical materials released during an accident and parts of RVZ1 and RVZ2. Confinement systems localize releases of radioactive or hazardous materials to controlled areas and mitigate the consequences of DBAs. Personnel protection control features such as adequate shielding and RV minimize hazards normally associated with radioactive or chemical materials. The principal design and safety objective of the confinement system is to protect the on-site personnel, the public, and the environment. The second design objective is to minimize the reliance on administrative or complex active engineering controls and provide a confinement system that is as simple and fail-safe as reasonably possible.

This subsection describes the confinement systems for the RPF. The RPF confinement areas include hot cell enclosures for process operations and trench and vault enclosures for process tanks and piping.

Confinement is achieved through RV, RICS, and biological shielding provided by the steel and concrete structures comprising the walls, roofs, and penetrations of the hot cells. Shielding of the hot cells is discussed in detail in Subsection 4b.2.

6b.2.1.2 Confinement System and Components

The RV serving the RCA, outside of the IF, includes components whose functions are designated as nonsafety-related and safety-related. The ductwork, the isolation dampers, and the filter trains of RVZ1 are designated as safety-related. Refer to Table 6b.2-~~1~~2 for a description of the system and component safety functions. Active confinement isolation components are required to operate as described below.

The hot cells employ a combination passive-active confinement methodology. During normal operation, passive confinement is achieved through the contiguous boundary between the hazardous materials and the surrounding environment and is credited with confining the hazards generated as a result of DBAs.

This boundary includes the biological shield (created by the physical construction of the cell itself) and the extension of that boundary through the RVZ1. The intent of the passive boundary is to confine hazardous materials while also preventing the introduction of external energy sources that could disturb the hazardous materials from their steady-state condition. The extent of this passive confinement boundary extends from the upstream side of the intake HEPA filter to the final downstream HEPA filter prior to exiting the building.

In the event of a DBA that results in a release in the hot cells, radioactive material would be confined by the biological shield and physical walls of the cell itself. Each line that connects directly to the hot cell atmosphere and penetrates the hot cell is provided with redundant isolation valves to prevent releases of gaseous or other airborne radioactive material. Confinement isolation valves on piping penetrating the hot cell are located as close as practical to the confinement boundary and active isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

To mitigate the consequences of an uncontrolled release occurring within a hot cell, as well as the off-site consequences of releasing fission products through the ventilation system, the confinement barrier utilizes an active component in the form of bubble-tight isolation dampers (safety-related) on the inlet and outlet ventilation ports of each hot cell. This ESF effectively reduces the amount of ductwork in the confinement volume that needs to remain intact to achieve hot cell confinement. These dampers close automatically (fail-closed) upon loss of power or receipt of a confinement isolation signal generated by the RICS. Following an initiating event, the RICS isolates the hot cells. Refer to Section 7b for a description of the RICS.

Overall performance assurance of the active confinement components is achieved through factory testing and in-place testing. Duct and housing leak tests are performed in accordance with ASME N511, with minimum acceptance criteria as specified in ASME AG-1 (ASME, 2009). Specific owner's requirements with respect to acceptable leak rates are based on the safety analyses.

6b.2.1.3 Functional Requirements

Active confinement components are designed to fail into a safe state if conditions such as loss of signal, loss of power, or adverse environments are experienced.

Mechanical, instrumentation, and electrical systems and components required to perform their intended safety function in the event of a single failure are designed to include sufficient redundancy and independence such that a single failure of any active component does not result in a loss of the capability of the system to perform its safety functions.

~~Mechanical~~ Safety-Related mechanical, instrumentation, and electrical systems and components are designed to ensure that a single failure of an active component, in conjunction with an initiating event, does not result in the loss of the system's ability to perform its intended safety functions. The single failure considered is a random failure ~~and any consequential failures in addition to the initiating event for which the system is required and any failures that are a direct or consequential result of the initiating event~~.

The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-2000 and Regulatory Guide 1.53 in the application of the single-failure criterion.

6b.2.1.4 Confinement Components

The following components are associated with the confinement barriers of the hot cells, tank vaults and pipe trenches, as previously described. Their specific materials, construction, and installation and operating requirements are evaluated based on the safety analysis.

Bubble-tight isolation dampers, designed, constructed and tested in accordance with ASME AG-1, Section DA "Dampers and Louvers" (ASME, 2009):

- Maintain their functional integrity.
- Maintain their rated leak-tightness following a seismic event.
- Maintain their structural integrity under fan shut-off pressure.
- Provide bubble-tight isolation upon receipt of a control signal or, in the event of loss of actuator power, by closure of actuator.

- Provide bubble-tight isolation when using manual actuator or when locked closed with power actuator removed.
- Relay damper full-open and full-closed position for control and indication by the use of limit switches.

Dampers are butterfly type, blade and frame fabricated of heavy-gage stainless steel. Total leakage based on bubble solution test as outlined in ASME AG-1 2009, Section DA-5141 (ASME, 2009).

Ventilation ductwork and ductwork support materials meet the requirements of ASME AG-1, Article SA-3000 “Materials”. Supports are designed and fabricated in accordance with the requirements of ASME AG-1, Section SA “Ductwork” (ASME, 2009).

Low leakage seals are provided on each penetration through the hot cells. ~~For systems~~ Systems open to the hot cell atmosphere~~,~~ that represent a potential unacceptable source of leakage are provided with redundant isolation valves ~~are provided~~.

6b.2.1.5 Engineered Safety Feature Test Requirements

Engineered safety features are tested to ensure that ESF components maintain operability and can provide adequate confidence that the system performs satisfactorily in service during postulated events.

To the extent possible, the RICS and the confinement ESF whose operation it initiates are designed to permit testing during plant operation. Testing actuation devices and actuated equipment may be done individually or in groups to avoid negative impact to plant operations.

6b.2.1.6 Design Bases

For general discussion of the codes and standards used for the SHINE facility, see Chapter 3. For the design basis of the RVZ1 and RVZ2, see Subsection 9a2.1.1. For the design basis of the hot cells, see Section 4b.2. For the ESF-related design basis of the RICS, see Section 7b.4.1.

Potential variables, conditions, or other items that will be probable subjects of a technical specification associated with the RPF confinement systems and components are provided in Chapter 14.

Table 6b.2-1 Radioisotope Production Facility Design Basis Accident Consequence Determination

Design Basis Accident	Identified Consequences	Subsection Containing Consequence Analysis
a. Critical equipment malfunction	Yes (Table 6b.1-1)	13b.2.4.7
b. Inadvertent nuclear criticality in the RPF	No	13b.2.5.7
c. RPF fire	Yes (Table 6b.1-1)	13b.2.6.7
d. Accidents with hazardous chemicals	Yes (Table 6b.1-1)	13b.3.2
e. External events	No	13b.2.3

Table 6b.2-~~1~~2 Radioisotope Production Facility Confinement Safety Functions

System, Structure, Component	Description	Classification
RVZ1 hot cell isolation dampers, ductwork up to filters and filters	Provide confinement isolation at hot cell boundaries	SR
RVZ2 isolation dampers, ductwork up to filters and filters	Provide confinement isolation at RCA boundary	SR
RICS	Provides confinement isolation signal	SR
Isolation valves on piping systems	Provide confinement at hot cell boundaries	SR
Hot cells, tank vaults, and pipe trenches	Provides confinement	SR

d. Simple administrative.

An accidental criticality is highly unlikely because the SHINE facility is designed with passive engineered design features, including the use of neutron absorbers.

The analysis, consequences, and safety controls of such an event have been described in Subsection 13b.2.5. Administrative controls ensure the reliability and availability of the safety controls are adequate to maintain subcriticality.

~~Heterogeneous effects are not considered applicable because the uranium enrichment is less than 20 percent.~~SHINE will consider heterogeneity effects when establishing NCS controls and limits, where such are credible and relevant.

Management of the Nuclear Criticality Safety Program

The NCS criteria are used for managing criticality safety and include adherence to the double contingency principle, as stated in ANSI/ANS-8.1-1998 (R2007) (ANSI/ANS, 2007a). The double contingency principle states “process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.” Each process that has accident scenarios that could result in an inadvertent nuclear criticality at the SHINE facility meets the double contingency principle. Process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

The NCS program establishes criteria for the administration of those operations in which there exists a potential for nuclear criticality accidents. Responsibilities of management, supervision, and the nuclear criticality safety staff are defined below. Objectives and characteristics of operating and emergency procedures are also defined. The emergency procedures will include reporting criteria and report content requirements. Reports will be issued based on whether the criticality controls credited were lost (i.e., they were unreliable or unavailable to perform their intended safety functions), irrespective of whether the safety limits of the associated parameters were actually exceeded. Training in criticality safety will be provided to individuals who handle nuclear material at the facility. The training is based upon the training program described in ANSI/ANS-8.20-1991 (R2005) (ANSI/ANS, 2005c) and ANSI/ANS-8.26 (R2012) (ANSI/ANS, 2012d).

Other aspects of the NCS program include:

- Providing distinctive NCS postings in areas, operations, work stations, and storage locations relying on administrative controls for NCS.
- Requiring personnel to perform activities in accordance with written, approved procedures when the activity may affect NCS. Unless a specific procedure deals with the situation, personnel shall take no action until the NCS staff has evaluated the situation and provided recovery procedures.
- Requiring personnel to report defective NCS conditions to the NCS program management.

- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- Nuclear criticality safety controls and controlled parameters ensure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety that is used.
- Process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded (ANSI/ANS, 2007a).
- Use of ANSI/ANS-8.7-1998 (R2007) (ANSI/ANS, 2007b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls.
- ANSI/ANS-8.10-1983 (R2005) (ANSI/ANS, 2005a), as modified by Regulatory Guide 3.71, as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible accident scenarios are used, NRC pre-approval of the administrative margins will be sought.
- Subcritical limits for k_{eff} calculations such that the margin is large compared to the uncertainty in the calculated values, and includes adequate allowance for uncertainty in the methodology, data, and bias to ensure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.
- The calculation of k_{eff} is based on a set of variables within the method's validated area of applicability.
- Trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

The NSCE procedure addresses requirements for:

- Normal case operating conditions.
- Nuclear criticality hazard identification.
- Hazard identification method.
- Hazard identification results.
- Nuclear criticality hazard evaluation.
- Nuclear criticality parameter discussions.
- Nuclear criticality safety controls (passive design features, active engineered features and administrative controls).
- Nuclear criticality safety peer review requirements.

<u>Geometry Tanks</u>

Each of the tanks within the scope of this section features criticality safety controls that meet the double-contingency principle, i.e., processes incorporate sufficient safety factors so that at least two unlikely, independent, and concurrent changes in process conditions are required before a nuclear criticality accident is possible. The first criticality safety control is that each tank, with the exception of the tanks associated with liquid waste processing, is criticality safe by geometry~~ or by the combination of geometry and a layer of neutron absorbing material integral to the tank construction~~. The second, independent criticality-safety control is ~~that the most reactive concentration of uranium in any tank results in k_{eff} ≤ 0.95, based on MCNP analyses. MCNP is a validated code for calculating reactivity and dosimetry. Small amounts of fissile plutonium-239 (Pu-239), resulting from activation of uranium-238 (U-238) and subsequent decay of~~

~~neptunium-239 (Np-239) during fuel solution irradiation, are accounted for in the MCNP calculations, even if the measured concentration of plutonium is negligible~~that solution leakage from a tank would enter a criticality-safe geometry sump or sump catch tank.

Piping

Pipe runs are single-parameter criticality-safe by geometry, with pipe diameters limited based on MCNP calculations, and with consideration of material and spacing relative to other uranium, reflecting, and moderating materials.

In the event of a leak in a double-walled pipe located in the pipe trenches, the target solution in the pipe trench would drain into a criticality-safe by geometry tank.

Tank Vaults

Constraints on the facility footprint prevent limiting potential maximum spills in tank vaults to single-parameter criticality safe depths. Therefore, each tank vault is connected, via a non-valved gravity drain into the criticality safe sump catch tank, which is criticality-safe by geometry.

Limits

Safe geometry limits for each of the following SSCs are established:

- Uranium and uranium oxide receipt and storage: storage container’s diameter and center-to-center spacing.
- Uranium dissolution: process vessels and piping diameter and center-to-center spacing.
- Target Solution Hold Tank (1-TSPS-03T): configuration (e.g., pencil or donut) and dimensions.
- TSV Dump Tank (1-SCAS-01T): configuration (e.g., pencil or donut) and dimensions.
- UREX (uranium extraction) recovery system: process vessels and piping diameter and center-to-center spacing.
- Denitration system: process vessels and piping diameter and center-to-center spacing.
- Criticality-safe sump catch tank (1-RDS-01T): configuration and dimensions.

Mass Limits

The mass of uranium present in an SSC is determined by analytical measurements and process calculations. Conservative administrative limits for each operation are specified in the operating procedures. Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits.

Mass limits, expressed as total mass of uranium or as uranium concentration, for each of the following SSCs will be established:

- Uranium and uranium oxide receipt and storage: total mass.

9b.7.5.3 System Design Inputs

9b.7.5.3.1 NDAS Waste Stream

The NDAS creates a waste stream that consists of activated vacuum hardware, target chambers, and the upper portion of the neutron driver.

9b.7.5.3.2 Solid Radioactive Waste Handling Hot Cell Waste Stream

The inputs to the solid radioactive waste handling hot cell are summarized in Figure 9b.7-7. These include spent Mo extraction columns, [Proprietary Information], scrap radioisotope purification process glassware, spent light water pool deionizer units, spent PCLS cooling water deionizer units, spent Tc removal columns, ~~[Proprietary Information]~~spent Cs columns, and scrap contaminated equipment.

9b.7.5.3.3 Miscellaneous Dry Waste Stream

Miscellaneous dry waste in the form of rags, paper, and contaminated PPE is collected in specific locations within the facility based on operational and health physics approaches.

9b.7.5.4 System Storage and Packaging Requirements

9b.7.5.4.1 Neutron Driver

The neutron driver drift tubes and target chambers are placed into a DOT IP-1 cargo container (SC-0001) for storage after removal from the light water pool. They accumulate until the cargo container is either full or the container reaches its maximum gross weight capacity of 67,200 lb. The upper portion of the neutron drivers are placed into decay storage. They are then placed into a DOT IP-1 cargo container for disposal. Refer to Table 9b.7-7 for waste classification, quantities, and final destination of the neutron driver waste.

9b.7.5.4.2 Solid Radioactive Waste Handling Hot Cell

Both the spent Mo extraction columns [Proprietary Information] are stored within the supercell Mo extraction area for a duration of no less than 2 weeks after use. After the initial decay period, the spent columns are transferred to the storage vault for an additional six months of decay time. The spent columns are then transferred to the solid radioactive waste handling hot cell for consolidation and packaging into an approved LSA container. Refer to Table 9b.7-7 for waste classification, quantities, and final destination of the Mo extraction [Proprietary Information].

The scrap radioisotope purification process glassware is removed from the supercell purification area and transferred to the solid radioactive waste handling hot cell. The glassware is deposited into a drum that has had an absorbent added and crushed, until the drum is full and then removed for shipping. Refer to Table 9b.7-7 for waste classification, quantities, and final destination of the radioisotope purification process glassware waste.

The ~~[Proprietary Information]~~spent Cs columns are transferred to the solid radioactive waste handling hot cell. ~~[Proprietary Information]~~The resins are then extracted and sent to the liquid radioactive waste handling hot cell for further processing. The resin-free spent Cs column is then processed and placed into liners for Type B shipping. Refer to Table 9b.7-7 for waste classification, quantities, and final destination of the ~~[Proprietary Information]~~spent Cs columns.

Scrap contaminated equipment is transferred to the solid radioactive waste handling hot cell, as necessary. The contaminated equipment is then packaged in a specifically designed LSA container.

9b.7.5.4.3 Miscellaneous Dry Waste

The miscellaneous dry waste is consolidated, packaged, and sent to the waste staging and shipping building. Refer to Table 9b.7-7 for waste classification, quantities, and final destination of the dry waste.

9b.7.5.5 Equipment Description

See Table 9b.7-6 for a list of major equipment associated with the solid radioactive waste packaging system.

9b.7.5.6 System Shipping Procedures

Solid radioactive waste is shipped on a regular basis by a DOT licensed carrier.

9b.7.5.7 Criticality Control Features

No criticality-safe control features are necessary in the solid radioactive waste packaging system as transuranic or fissile material is not present in significant quantities in these waste streams.

As an additional precaution, the SRWP is monitored by the CAAS.

9b.7.5.8 Instrumentation and Control

The RICS provides the process monitoring and control of the SRWP (See Subsection 7b.1).

9b.7.5.9 Technical Specifications

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

9b.7.6 RADIOACTIVE DRAIN SYSTEM

9b.7.6.1 RDS Process Description

The RDS receives radioactive liquids from various systems. Its primary purpose is to collect spills and leaks of radioactive liquids from processing areas.

Table 9b.7-7 Estimated Type and Quantity of Radioactive Wastes Associated with the SHINE Facility (Sheet 1 of 3)

Description	Matrix	Class as Generated	Contents	Volume	Volume as shipped (ft^3)	55-gallon drum equivalent as shipped	Shipment Type	Number of Shipments/yr	Destination
Neutron Generator	Solid	A	Activated metal parts	4338 ft^3/yr	4338	590	LSA	3	ES
Extraction Columns	Solid	A	Stainless resin columns						
Class A Trash	Solid	A	PPE, Mo-99 purification glassware, filters, etc						
Coolant Cleanup Ion Exchange Resin	Resin	A	Resin	48 ft^3/yr	48	7.3	LSA	1	ES
Spent Solvent	Liquid	A	n-dodecane, tributyl phosphate	22 gallons/yr	--	0.4	LSA	1	DSSI
Tc/I columns	Resin	C	Resin	16 gallons/yr	23	3.1	Type B	0.3	WCS
Zeolite Beds	Solid	GTCC	Silver coated beds	0.4 ft^3/yr	0.4	0.05	Type B	1	WCS
~~[Proprietary Information]~~ Cs/Ce Media	~~[Proprietary Information]~~ Resin	GTCC	Resin	16 gallons/yr	23	3.1	Type B	0.3	WCS
[Proprietary Information]	[Proprietary Information]	B	[Proprietary Information]	295 gallons/yr	79	11	Type B	1	WCS

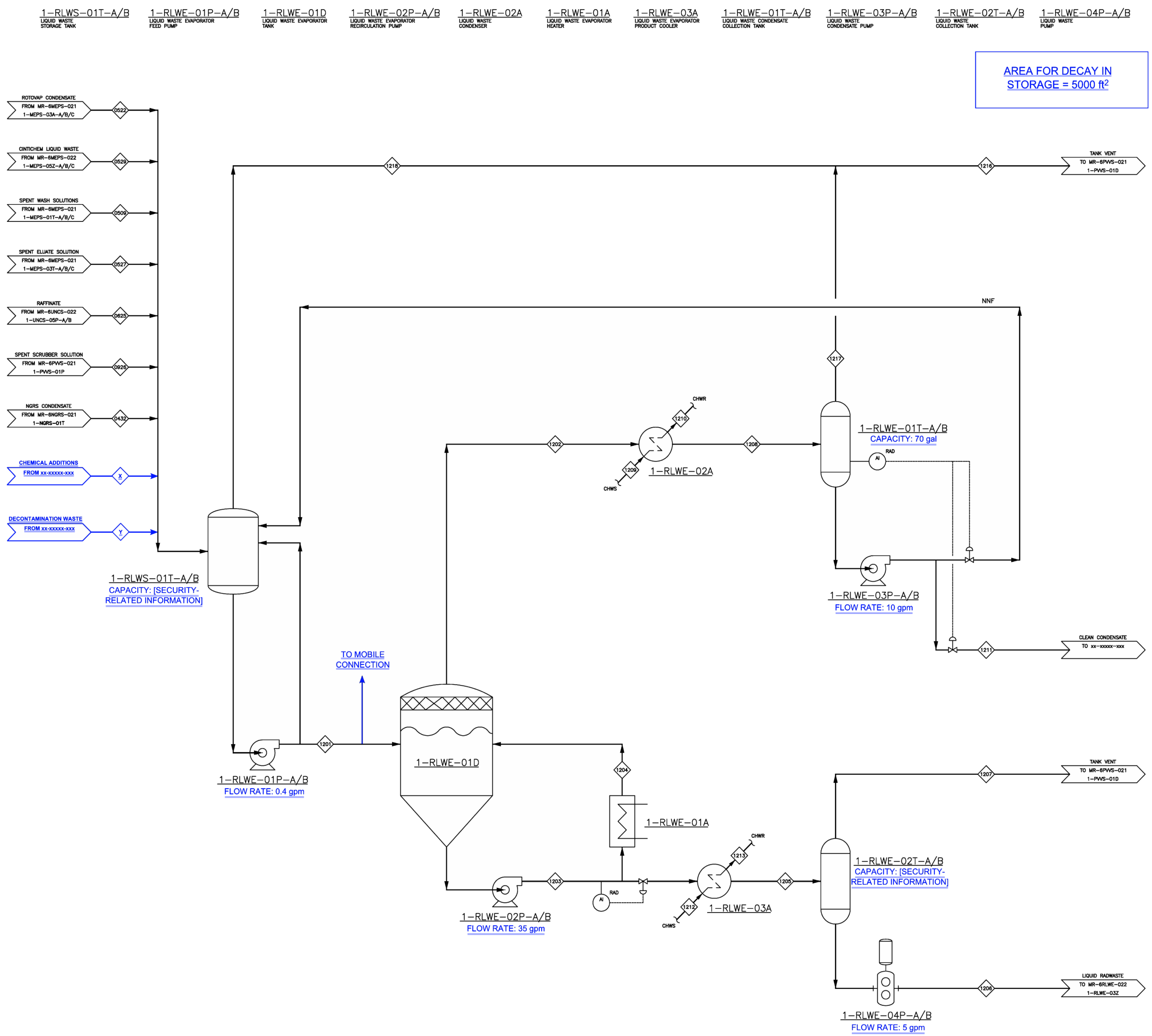
Table 9b.7-7 Estimated Type and Quantity of Radioactive Wastes Associated with the SHINE Facility (Sheet 2 of 3)

Description	Matrix	Class as Generated	Contents	Volume	Volume as shipped (ft^3)	55-gallon drum equivalent as shipped	Shipment Type	Number of Shipments/yr	Destination
Spent Washes	Liquid[a]	A	[Proprietary Information]	~~59,708 gallons/yr~~ 2100 gallons/yr					
Rotovap Condensate	Liquid[a]	A	[Proprietary Information]	200 gallons/yr					
UREX Raffinate	Liquid[a]	B	[Proprietary Information]	27,000 gallons/yr	~~9738~~ 7540[b]	~~1324~~ 1143[c]	LSA	~~18~~ 19	ES
Decontamination Waste	Liquid[a]	A	Decon fluid unknown	400 gallons/yr					
Spent Eluate Solution	Liquid[a]	A	[Proprietary Information]	2600 gallons/yr					
NO_x Scrubber Solution	Liquid[a]	A	[Proprietary Information]	20,000 gallons/yr					

Table 9b.7-7 Estimated Type and Quantity of Radioactive Wastes Associated with the SHINE Facility (Sheet 3 of 3)

a) This liquid waste discharged from the various processes at the SHINE facility is either solidified and then shipped to a waste depository or reused.
b) As shipped volume of waste is in the form of concrete. Total liquid volume of approximately 52,000 gallons/yr is reduced via evaporation to approximately 35,000 gallons/yr (volume reduction factor of 1.5). The liquid waste is then solidified by adding Portland cement (or equivalent). Concrete volume is estimated using a conservative waste to cement mass ratio of 0.5.
c) A 55-gallon drum, filled to 90 percent to account for minor voiding, has a volume of approximately 6.6 ft^3.

Figure 9b.7-5 – RLWE Liquid Waste Evaporator and Storage System Process Flow Diagram



List of Tables

Information developed from reportable occurrences is tracked in the Corrective Action Program and is used to improve radiation protection practices, precluding the recurrence of similar incidents.

11.1.2.1.1 Responsibilities of Key Program Personnel

In this subsection the Radiation Protection Program organization, shown in Figure 11.1-1 is described. The responsibilities of key personnel are also discussed. These personnel play an important role in the protection of workers, the environment and implementation of the ALARA program. Chapter 12 discusses the organization and administration and responsibilities of key management personnel in further detail.

Plant Manager

The plant manager is responsible for operation of the facility, including the protection of personnel from radiation exposure resulting from facility operations and materials, and for compliance with applicable NRC regulations and the facility license. The plant manager reports to the chief operating officer (COO).

Environment, Safety, and Health Manager

The environment, safety, and health (ES&H) manager reports to the COO and has the responsibility for directing the activities that ensure the facility maintains compliance with appropriate rules, regulations, and codes. This includes ES&H activities associated with nuclear safety, radiation protection, chemical safety, environmental protection, industrial safety and establishing and maintaining the radiological environmental monitoring program. The ES&H manager works with other managers to ensure consistent interpretations of ES&H requirements, performs independent reviews, and supports facility and operations change control reviews.

Radiation Protection ~~Manager~~Supervisor

The radiation protection ~~manager~~supervisor reports to the ES&H manager. The radiation protection ~~manager~~supervisor is responsible for implementing the Radiation Protection Program. In matters involving radiological protection, the radiation protection ~~manager~~supervisor has direct access to executive management.

The radiation protection ~~manager~~supervisor and his/her staff are responsible for:

- Establishing the radiation protection program.
- Generating and maintaining procedures associated with the program.
- Ensuring that ALARA is practiced by personnel.
- Reviewing and auditing the efficacy of the program in complying with NRC and other governmental regulations and applicable regulatory guides.
- Modifying the program based upon experience and facility history.

- Adequately staffing the radiation protection group to implement the Radiation Protection Program.
- Ensuring that the occupational radiation exposure dose limits of 10 CFR 20 are not exceeded under normal operations.
- Establishing and maintaining an ALARA program.
- Establishing and maintaining a Respiratory Protection Program.
- Monitoring worker doses, both internal and external.
- Handling of radioactive wastes when disposal is needed.
- Calibration and quality assurance of health physics associated radiological instrumentation.
- Establishing and maintaining a radiation safety training program for personnel working in the radiologically restricted areas.
- Performing audits of the Radiation Protection Program on an annual basis.
- Posting the restricted areas, and within these areas, posting radiation, airborne radioactivity, high radiation, and contaminated areas as appropriate.

Operations Manager

The operations manager is responsible for operating the facility safely and in accordance with procedures so that effluents released to the environment and exposures to the public and on-site personnel meet the limits specified in applicable regulations, procedures and guidance documents.

On-site Personnel

On-site personnel are required to work safely and to follow the rules, regulations and procedures that have been established for their protection and the protection of the public. Personnel whose duties require (1) working with radioactive material, (2) entering restricted areas, (3) controlling facility operations that could affect effluent releases, or (4) directing the activities of others, are trained such that they understand and effectively carry out their responsibilities.

11.1.2.1.2 Staffing of the Radiation Protection Program

The Radiation Protection Program staff is assigned responsibility for implementation of the Radiation Protection Program functions, therefore, only suitably trained radiation protection personnel are employed at the facility. The radiation protection staff includes a radiation protection ~~manager~~supervisor and radiation control technicians.

Staffing is consistent with the guidance provided in Regulatory Guides 8.2 and 8.10. For example, the radiation protection ~~manager~~supervisor has, as a minimum, a bachelor's degree (or equivalent) in an engineering or scientific field and three years of responsible nuclear experience

associated with implementation of a radiation protection program. Other members of the Radiation Protection Program staff are trained and qualified consistent with the guidance provided in Regulatory Guide 1.8.

Sufficient resources in terms of staffing and equipment are provided to implement an effective radiation protection program.

11.1.2.1.3 Independence of the Radiation Protection Program

The Radiation Protection Program is independent of facility operations. This independence ensures that the Radiation Protection Program maintains its objectivity and is focused only on implementing sound radiation protection principles necessary to achieve occupational doses and doses to members of the public that are ALARA.

11.1.2.1.4 Radiation Safety Committee

A radiation safety committee (RSC) is established and meets periodically (at least annually) to review the status of projects, measure performance, look for trends and to review radiation safety aspects of facility operations, in accordance with 10 CFR 20.1101(c). The radiation protection ~~manager~~supervisor chairs the RSC. The other RSC members come from quality assurance, operations, maintenance, and technical support.

The objectives of the RSC are to maintain a high standard of radiation protection in facility operations. The RSC reviews the content and implementation of the radiation protection program at a working level and strives to improve the program by reviewing exposure trends, the results of audits, regulatory inspections, worker suggestions, survey results, reportable occurrences, and exposure incidents.

A written report of each RSC meeting is forwarded to all managers.

An official RSC charter will be prepared defining the purposes, functions, authority, responsibility, composition, quorum, meeting frequency, and reporting requirements of the RSC.

11.1.2.1.5 Commitment to Written Radiation Protection Procedures

Radiation protection procedures are to be prepared, reviewed and approved to carry out activities related to the Radiation Protection Program. Procedures are used to control radiation protection activities in order to ensure that the activities are carried out in a safe, effective and consistent manner. Radiation protection procedures are reviewed and revised as necessary by the radiation protection ~~manager~~supervisor to incorporate any facility or operational changes.

Work performed in radiologically controlled areas is performed in accordance with a radiation work permit (RWP). The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10. A RWP is required whenever the radiation protection ~~manager~~supervisor determines one is necessary. The RWP provides a description of the work to be performed. The RWP summarizes the results of recent dose rate surveys, contamination surveys, and airborne radioactivity measurements. The RWP specifies the precautions to be taken by those performing the task. The specified

- Instructed in the appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and radioactive material.

- Advised of the various notifications and reports to individuals that a worker may request in accordance with 10 CFR 19.13.

The radiation protection training program takes into consideration a worker's normally assigned work activities. Abnormal situations involving exposure to radiation and radioactive material, that can reasonably be expected to occur during the life of the facility, are also evaluated and factored into the training. The extent of these instructions is commensurate with the potential radiological health protection problems present in the work place.

Retraining of personnel previously trained is performed for radiological, chemical, industrial, and criticality safety at least annually. The retraining program also includes procedure changes and updating and changes in required skills. Changes to training are implemented, when required, due to incidents potentially compromising safety or if changes are made to the facility or processes. Records of training are maintained in accordance with a records management system. Training programs are established in accordance with Subsection 12.1.4. The radiation protection sections of the training program are evaluated at least annually. The program content is reviewed to ensure it remains current and adequate to ensure worker safety.

11.1.2.1.7 Radiation Safety Audits

Audits are conducted, at a minimum, on an annual basis for the purpose of reviewing all functional elements of the Radiation Protection Program. This function is performed to meet the requirements of 10 CFR 20.1101(c). The audit activity is led by the radiation protection ~~manager~~supervisor with the results being sent to the RSC, the COO, and the CEO for review. Deficiencies identified during the audit are addressed through the Corrective Action Program.

11.1.2.1.8 Record Keeping

Record keeping associated with the radiation protection program meets the requirements of 10 CFR 20 Subpart L.

11.1.2.1.9 Technical Specifications

There are no potential variables, conditions, or other items that are probable subjects of a technical specification associated with radiation safety audits.

11.1.3 ALARA PROGRAM

Subsection 11.1.2.1 states the facility's commitment to the implementation of an ALARA program. The objective of the program is to make every reasonable effort to maintain exposure to radiation as far below the dose limits of 10 CFR 20.1201 and 10 CFR 20.1301 as is practical. The design and implementation of the ALARA program is consistent with the guidance provided in Regulatory Guides 8.2, 8.13, and 8.29. The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10.

Annual doses to individual personnel are maintained ALARA. In addition, the annual collective dose to personnel (i.e., the sum of annual individual doses, expressed in person-sievert [Sv] or person-rem) is maintained ALARA. The dose equivalent to the embryo/fetus of a declared pregnant worker is maintained below the limits of 10 CFR 20.1208.

The Radiation Protection Program is written and implemented to ensure that it is comprehensive and effective. The written program documents policies that are implemented to ensure the ALARA goal is met. Procedures are written so that they incorporate the ALARA philosophy into the routine operations and ensure that exposures are consistent with administrative dose limits. As discussed in Subsection 11.1.5, radiological zones are established within the facility. The establishment of these zones supports the ALARA commitment by minimizing the spread of contamination and reducing exposure of personnel to radiation.

Specific goals of the ALARA program include maintaining occupational exposures and environmental releases as far below regulatory limits as is reasonably achievable. The ALARA concept is also incorporated into the design of the facility. The plant is divided into radiation zones with radiation levels that are consistent with the access requirements for those areas. Areas where on-site personnel spend significant amounts of time are designed to maintain the lowest dose rates reasonably achievable.

The radiation protection ~~manager~~supervisor is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The radiation protection ~~manager~~supervisor prepares an annual ALARA program evaluation report. The report reviews (1) radiological exposure and effluent release data for trends, (2) audits and inspections, (3) use, maintenance, and surveillance of equipment used for exposure and effluent control, and (4) other issues that may influence the effectiveness of the radiation protection/ ALARA programs. Copies of the report are submitted to the plant manager and the RSC. The RSC monitors the duties of the radiation protection staff. This is to ensure they are specifically involved during the planning and implementing of operation and maintenance activities and the management and disposition of radioactive wastes.

Programs for improving the effectiveness of equipment used for effluent and exposure control are also evaluated by the RSC. The recommendations of the committee are documented in writing. The committee's recommendations will be dispositioned through the Issues Management System.

As part of its duties, the RSC reviews the effectiveness of the ALARA program and determines if exposures, releases and contamination levels are in accordance with the ALARA concept. It also evaluates the results of assessments made by the radiation protection organization, reports of facility radiation levels, contamination levels, and employee exposures for identified categories of workers and types of operations. The committee is responsible for ensuring that the exposure limits of 10 CFR 20 are not exceeded under normal operations. The committee determines if there are any unplanned upward trends in personnel exposures, environmental releases, and facility contamination levels.

The ALARA program facilitates interaction between radiation protection and operations personnel. The RSC, comprising staff members responsible for radiation protection and operations, is particularly useful in achieving this goal. The RSC periodically reviews the goals and objectives of the ALARA program. The ALARA program goals and objectives are revised to

NRC regulation 10 CFR 20.1003 defines an unrestricted area as an area for which access is neither limited nor controlled by the licensee. The area adjacent to the SHINE site is an unrestricted area. This area can be accessed by members of the public or by facility personnel. The unrestricted area is governed by the limits in 10 CFR 20.1301. The total effective dose equivalent to individual members of the public from the licensed operation may not exceed 1 mSv (100 mrem) in a year (exclusive of background radiation). The dose in any unrestricted area from external sources may not exceed 0.02 mSv (2 mrem) in any one hour.

b. Restricted area

The NRC defines a restricted area as an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Access to and egress from a restricted area at the facility site is through a radiation protection control point. Monitoring equipment is located at these control points.

Most restricted areas are located within the physical structure of the SHINE facility. However, radioactive material may be temporarily stored outside the facility in areas such as the waste staging and shipping building. Such areas may require that a restricted area be established with the controls described in this section.

The RCA is a restricted area. Personnel who have not been trained in radiation protection procedures are not allowed to access this area without escort by trained personnel.

The additional areas defined below may exist within the restricted area. These areas may be temporary or permanent. The areas are posted to inform workers of the potential hazard in the area and to help prevent the spread of contamination. These areas are conspicuously posted in accordance with the requirements of 10 CFR 20.

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (5 mrem) in 1 hour (hr) at 30 cm (11.8 in.) from the radiation source or from any surface that the radiation penetrates is designated a "radiation area" as defined in 10 CFR 20.1003.
- An "airborne radioactivity area" means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations (1) In excess of the derived air concentrations (DACs) specified in Appendix B, to 10 CFR 20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake or 12 DAC-hours. Note that entry into this area does not automatically require the wearing of a respirator.
- A "high radiation area" is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in 1 hr at 30 cm (11.8 in.) from the radiation source or from any surface that the radiation penetrates. No areas of this type are accessible to individuals during routine operation of the facility. Such areas are radiologically shielded and isolated from access to individuals by the use of engineered physical barriers. These include structural shield blocks, and/or locked shield doors.
- A “very high radiation area” is an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads (5 grays) in 1 hr at 1 m from a radiation source or 1 m from any surface that the radiation penetrates. Note that at very high doses received at high dose rates, units of absorbed dose (e.g., rads or grays) are appropriate, rather than units of dose equivalent (e.g., rems and sieverts).

11.1.5.5 Personnel Monitoring for External Exposures

External exposures are received primarily from the fission products produced in the target solution. The nuclides of radiological significance are identified in Section 11.1.

Personnel whose duties require them to enter restricted areas wear individual external dosimetry devices that are sensitive to beta and gamma radiation. Personal dosimetry shall be worn in a manner consistent with the manufacturer's directions. External dosimetry devices are evaluated at least quarterly, or soon after participation in high-dose evolutions, to ascertain external exposures. Administrative limits on radiation exposure are listed in Table 11.1-7, Administrative Radiation Exposure Limits.

If 25 percent of the annual administrative limit is exceeded in any quarter, then an investigation is performed and documented to determine what types of activities may have contributed to the worker's external exposure. The administrative limit already reflects ALARA principles, so this action level is appropriate. This investigation may include, but is not limited to procedural reviews, efficiency studies of the ventilation system, uranium storage protocol, and work practices.

Anytime an administrative limit is exceeded, the radiation protection ~~manager~~supervisor is informed. The radiation protection ~~manager~~supervisor is responsible for determining the need for and recommending investigations or corrective actions to the responsible manager(s). Copies of the radiation protection ~~manager~~supervisor's recommendations are provided to the RSC.

11.1.5.6 Determination of Internal Exposures

a. For purposes of assessing dose used to determine compliance with occupational dose equivalent limits, SHINE shall, when required under 10 CFR 20.1502, take suitable and timely measurements of one of the following:

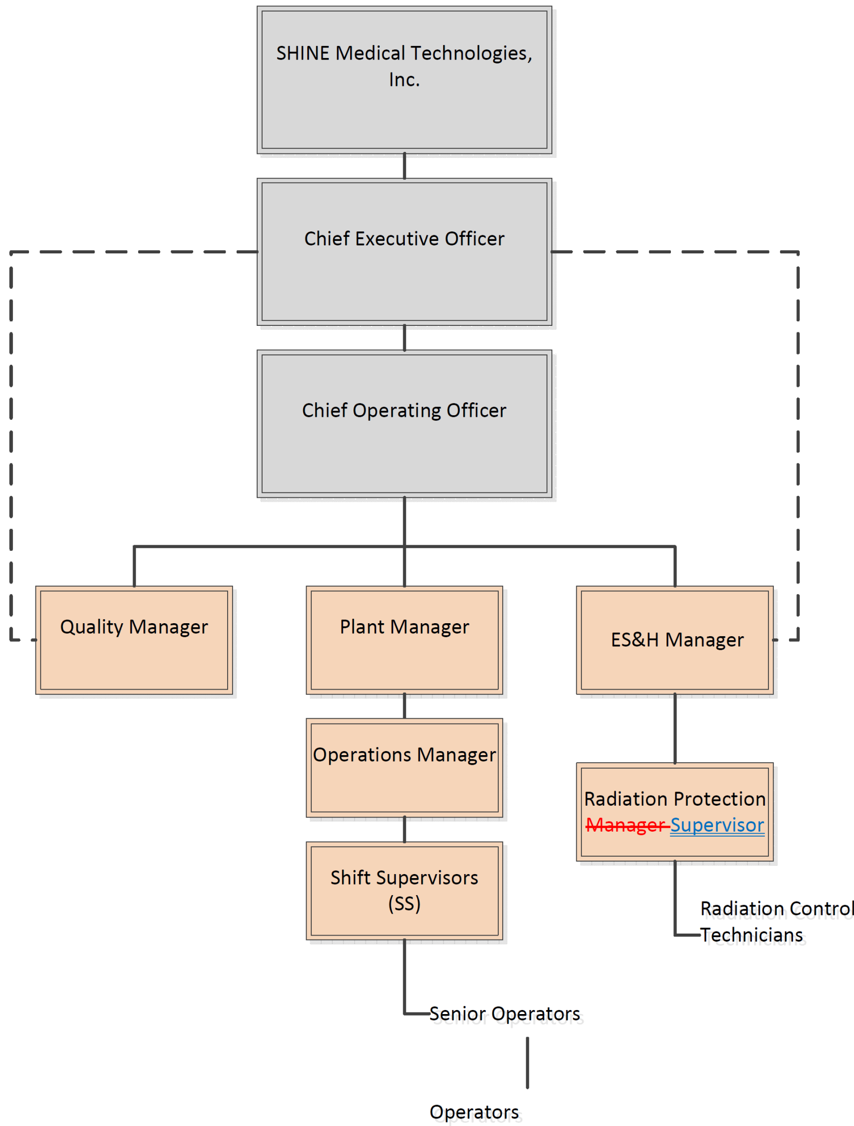
   1. Concentrations of radioactive materials in air in work areas.
   2. Quantities of radionuclides in the body.
   3. Quantities of radionuclides excreted from the body.
   4. Combinations of these measurements.

b. Unless respiratory protective equipment is used, as provided in 10 CFR 20.1703, or the assessment of intake is based on bioassays, SHINE shall assume that an individual inhales radioactive material at the airborne concentration in which the individual is present.

c. When specific information on the physical and biochemical properties of the radionuclides taken into the body or the behavior or the material in an individual is known, SHINE may:

   1. Use that information to calculate the committed effective dose equivalent, and, if used, the licensee shall document that information in the individual's record.
   2. Adjust the DAC or annual limit on intake (ALI) values to reflect the actual physical and chemical characteristics of airborne radioactive material (e.g., aerosol size distribution or density).

Figure 11.1-1 – Organization Chart Showing the Radiation Protection Organization



UREX Raffinate

While the bulk of the fission products produced in the irradiation process goes into UREX, because of the large volume and weight, this waste stream (Class B) is not expected to be GTCC. This waste stream is Class B as produced. The raffinate goes to the consolidated radioactive liquid waste tank. When the waste is concentrated through a treatment process, such as evaporation, the waste classification increases. In the case of an approach in which the waste is concentrated, waste classification is then dominated by plutonium (Pu). There is no Class B limit for Pu. Therefore, if the waste is greater than Class A, then the waste is Class C.

Thermal Denitration Evaporator Condensate

UREX is a solvent extraction process in which the aqueous phase is sent to the raffinate waste stream (Class A). The solvent is re-used and the uranium is routed for TDN. Prior to TDN, the remaining technetium (Tc) and iodine (I) are removed. Since H-3 is strongly bound as water, most of the H-3 has already left the process and entered into the aqueous raffinate waste stream. The large volume of water condensate from the TDN process is not likely to have significant contaminant buildup; therefore, the evaporator condensate is easily recycled. The UREX strip solution shares the same chemical makeup as this waste stream and as such, it is reused as strip solution.

Spent Solvent Replacement

Solvent replacement rate is a function of absorbed dose. The solvent is not a Resource Conservation and Recovery Act (RCRA) listed waste (Class A).

Spent Resin Column

The spent resin column captures the remaining Tc and I. Column change out frequency may be dictated by I-129 and Tc-99 loading as they are both sensitive radionuclides for waste classification (Class C). The column has been sized and the medium for removal of technetium has been selected; however, the column loading and change out frequency has not been determined. In order to estimate waste volume it was assumed the waste volume is approximately the same of the waste volumes associated with the ~~[Proprietary Information]~~cesium-137 (Cs-137) ion exchange columns. The selected resin for technetium removal is organic. Most likely, the removal media for iodine will be inorganic, thus leading to a mixed bed type of removal column. Due to these uncertainties, a definite treatment and disposal pathway has not been determined, however, provided the waste is not GTCC a processor will be able to accept the waste, treat it, and provide turnkey disposal to an appropriate facility, most likely Waste Control Services (WCS) due to potential radionuclide content.

11.2.2.2.7 Process Vessel Vent System

The process vessel vent system (PVVS) generates spent scrubber solution. It is assumed this waste stream (Class A) is routed to the consolidated radioactive liquid waste tanks and the volume is estimated at 20,000 gallons per year (75,708 liters per year).

11.2.2.2.8 Decontamination Waste

This is a liquid waste stream (Class A) resulting from the decontamination of SSCs during normal operation.

11.2.2.2.9 Coolant Clean-up Systems

The coolant clean-up subsystems generate spent ion exchange resins. Volumes for this waste stream (Class A) will be provided in the FSAR.

11.2.2.2.10 Radioactive Liquid Waste Processing

~~[Proprietary Information]~~Cesium-137 (Cs-137) and cerium-144 (Ce-144)/praseodymium-144 (Pr-144) removal from selected liquid waste streams is accomplished using ion exchange resins. These ~~[Proprietary Information]~~resins will be GTCC waste.

11.2.2.2.11 Noble Gas Removal System

The TSV off-gas system (TOGS) is designed to treat the off-gas produced by radiolysis during irradiation of the target solution. The irradiation process produces hydrogen and oxygen, as well as small mass quantities of Kr, Xe and iodine. The noble gas removal system (NGRS) collects off-gases from TOGS for decay. Once the acceptance criteria are met, the gases are discharged to PVVS. From PVVS, they are sent to RCA ventilation Zone 1 (RVZ1) where they are treated by charcoal and HEPA filters. From here, the gases are then released from the facility through the stack.

11.2.2.3 Technical Specifications

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

11.2.3 RELEASE OF RADIOACTIVE WASTE

Release for the purposes of this subsection means that wastes are processed and packaged as required to meet the waste acceptance criteria of an established LLW disposal facility. Processing may comprise one or more of several operations, including compaction, solidification with an appropriate medium (e.g., Aquaset, Portland cement grout), adsorption onto a solid medium (e.g., elemental iodine onto activated carbon filters), interim storage for decay of short-lived radionuclides, extraction and consolidation of short lived radionuclides by segregation, and mixing (possibly from more than one waste stream) so that the bulk volume of waste is readily disposed of and a small volume of high dose rate material is held for decay.

The SHINE facility does not discharge any material from the RCA to the sanitary sewer. Waste disposal methods meet the requirements of 10 CFR 20 Subpart K.

Table 11.2-1, shows the shipment classifications and expected disposal sites for the identified waste streams.

11.2.3.2 Liquid Waste Streams

Several waste streams are solidified on-site to meet United States Department of Transportation (DOT) and waste acceptance criteria. Process liquids are consolidated into a set of tanks prior to treatment via pH adjustment ~~[Proprietary Information]~~and removal of Cs-137 via ion exchange resins. The [Proprietary Information] waste may be treated together or separately. One process cell is dedicated to waste solidification. The consolidated liquid waste stream (post-treatment) is amenable for disposal as Class A waste at Energy*Solutions* whereas the [Proprietary Information] waste stream and ion exchange resin waste stream are greater than Class A and as such disposal at WCS is the ultimate disposal pathway.

11.2.3.2.1 Consolidated Liquids

Consolidated waste streams consist of spent washes, rotary evaporate condensate, UREX raffinate, scrubber solution, decontamination waste, and spent eluate solution. Total yearly volumes are based on the conservative assumption that 400 batches of TSV solution are processed per year and the UREX is performed (approximately) [Proprietary Information].

This waste stream is processed through pH adjustment, ~~[Proprietary Information]~~removal of Cs-137, evaporation, solidification and re-use. The evaporator concentrates are Class A waste and solidified. The evaporator overheads are of very low radioactivity and are re-used within the SHINE process. The current target receiving facility for the solidified evaporator concentrates is Energy*Solutions.* Requirements for this waste stream are presented in Table 11.2-5.

11.2.3.2.2 [Proprietary Information]

This waste stream is processed separately from the consolidated liquid evaporator concentrate. The waste stream is solidified using Portland cement as well as some small additions to treat sulfates. Requirements for this waste stream are presented in Table 11.2-6.

11.2.3.2.3 [Proprietary Information]

This waste stream is processed separately from the consolidated liquid evaporator concentrate. The waste stream is solidified using Portland cement. This waste stream does not require a hold-for-decay period but may be held for shipping consolidation purposes. Requirements for this waste stream are presented in Table 11.2-7.

11.2.3.3 Gaseous Waste Streams

The NGRS collects off-gases from TOGS for decay. Once the acceptance criteria are met, the gases are discharged to the PVVS and then RCA ventilation Zone 1 (RVZ1), where they are treated by charcoal and HEPA filters. From there, the gases are released from the facility through the stack. The expected activity of the annual release of gaseous effluents is provided in Table 11.1-9. These releases are monitored by the stack release monitors and their effect on the surrounding environment is monitored by the program described in Subsection 11.1.7. Discussion of the environmental effects of the release of gaseous effluents is provided in Subsection 11.1.1.1.

Table 11.2-1 Waste Stream Summary

Description	Matrix	Class as Generated	As Generated Amount	As Generated Units	As shipped (cubic feet (ft^3))	Ship Type	Number Shipments	Destination
Neutron Generator	Solid	A	4338	ft^3/yr	4338	LSA	3.0	Energy *Solutions*
Extraction Columns	Solid	A						
Class A Trash	Solid	A						
Coolant Cleanup Ion Exchange Resin	Resin	A	48	ft^3/yr	48	LSA	1.0	Energy *Solutions*
Spent Solvent	Liquid	A	22	gallons/yr	--	LSA	1.0	Diversified Scientific Services, Inc. (DSSI)
Tc/I Columns	Resin	C	16	gallons/yr	23	Type B	0.3	Waste Control Specialists
Zeolite Beds	Solid	GTCC	0.4	ft^3/yr	0.4	Type B	1.0	Waste Control Specialists
~~[Proprietary Information]~~ Cs/Ce Media	~~[Proprietary Information]~~ Resin	GTCC	16	gallons/yr	23	Type B	0.3	Waste Control Specialists
[Proprietary Information]	[Proprietary Information]	B	295	gallons/yr	79	Type B	1.0	Waste Control Specialists
Spent Washes	Liquid[a]	A	~~59,708~~2100	gallons/yr	~~9738~~ 7540[b]	LSA	~~18~~ 19	Energy *Solutions*
Rotary Evaporator Condensate	Liquid[a]	A	200					
UREX Raffinate	Liquid[a]	B	27,000					
NO_x Scrubber Solution	Liquid[a]	A	20,000					
Decontamination Waste	Liquid[a]	A	400					
Spent Eluate Solution	Liquid[a]	A	2600					

a) This liquid waste discharged from the various processes at the SHINE facility is either solidified and then shipped to a waste depository or reused.

b) As shipped volume of waste is in the form of concrete. Total liquid volume of approximately 52,000 gallons/yr is reduced via evaporation to approximately 35,000 gal/yr (volume reduction factor of 1.5). The liquid waste is then solidified by adding Portland cement (or equivalent). Concrete volume is estimated using a conservative waste to cement mass ratio of 0.5.

Table 11.2-5 Waste Methodology for Consolidated Liquids

Requirement	Basis
Provide two [Security-Related Information] storage tanks for consolidation of streams.	Needed to provide sufficient storage for holding for decay and processing of liquid waste.
Switch waste flow from one tank to another.	When waste fills first [Security-Related Information] tank, need to switch output to second [Security-Related Information] tank. When second tank has reached 25% of capacity, start processing first tank.
Provide pH adjustment.	Liquid waste must have a pH between 3 and 10 ~~[Proprietary Information]~~prior to treatment via ion exchange resin.
Sample influent waste stream.	Sampling of influent waste stream is performed to gather initial characterization data.
~~[Proprietary Information]~~ Provide ion exchange treatment equipment	~~[Proprietary Information]~~Required to remove Cs-137 resulting in a final waste form meeting less than 1 Ci/m^3. Column is projected to be replaced after 400 column volumes of treated liquid. This leads to approximately 140 gallons of resin per year expended.
Provide means to evaporate waste.	Approximate volume reduction factor is 1.5. Influent volume is approximately ~~55,000~~52,000 gallons per year (roughly ~~275~~1040 gallons per week based on 50 weeks of operation per year).
Provide means to process evaporator concentrate through solidification hot cell keeping concentrate separate from other waste streams.	Liquid waste must be solidified. Approximate influent volume is ~~36,000~~35,000 gallons per year (roughly ~~180~~700 gallons per week based on 50 weeks of operation per year).
Sample storage tank after it is filled and mixed.	A representative sample is required for accurate characterization of liquid waste prior to solidification and disposal.
Solidify waste by adding Portland cement.	Using Portland cement (or equivalent) to ensure final waste form meets requirements. Approximate ratio of waste to cement is 0.5 to 0.7. Required to meet WAC maximum free liquids requirement for solidified waste forms (0.5% by volume).
Limit void space.	WAC requirement to minimize void space.
Re-use evaporator overheads.	SHINE's goal is to have zero liquid effluents from the RCA. As such, evaporator overheads are planned to be re-used in the system. Approximate yearly volume is 19,000 gallons.
Establish dedicated area in the waste staging and shipping building for final decay time and shipment consolidation.	Solidified waste requires several months of decay post-processing in order to meet both DOT and Class A waste limits.
Maintain records relative to drums in the storage area.	Drums may not be removed from the storage area before decaying to Class A and DOT limits.

Table 11.2-7 Waste Methodology for ~~[Proprietary Information]~~Ion Exchange Resins

Requirement	Basis
Waste stream segregated from evaporator concentrate.	~~[Proprietary Information]~~Cs-137 must not be re-introduced into the evaporator concentrate after selective Ci removal is completed.
Waste stream characterized indirectly through data collected from the influent consolidated liquid waste stream.	Waste must be less than 4600 Ci/m^3 to be less than Class C waste. Based on 400 column volumes per year change out rate, the influent treatment volume and Ci count the approximate concentration prior to solidification is [Security-Related Information].
Provide capacity to solidify ~~[Proprietary Information]~~ion exchange resins.	Volume is approximately 140 gallons per year but contains residual liquid. Using Portland cement (or equivalent) with the potential need for small additions to treat sulfate.
Solidify ~~[Proprietary Information]~~ion exchange resins.	Approximate ratio of waste to cement is 0.5 to 0.7. Required to meet WAC maximum free liquids requirement for solidified waste forms (0.5% by volume).
Limit void space.	WAC requirement to minimize void space.
Establish dedicated area in the waste staging and shipping building for final decay time and shipment consolidation.	Waste stream is consolidated for Type B shipment. Minimal footprint needed (approximately space for three 55 gallon shielded containers).

- Either every 12 months thereafter, or periodically at a frequency determined by a physician.

A respirator fit test requires a minimum fit factor of at least 10 times the assigned protection factor (APF) for negative pressure devices, and a fit factor of at least 500 times the APF for any positive pressure, continuous flow, and pressure-demand devices. The fit testing is performed before the first field use of tight fitting, face-sealing respirators. Subsequent testing is performed at least annually thereafter. Fit testing must be performed with the face piece operating in the negative pressure mode:

- Each user is informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.
- In the selection and use of respirators, the facility provides for vision correction, adequate communication, low temperature work environments, and the concurrent use of other safety or radiological protection equipment. Radiological protection equipment is used in such a way as not to interfere with the proper operation of the respirator.
- Atmosphere-supplying respirators are supplied with respirable air of grade D Quality or better as defined by the Compressed Gas Association in publication G-7.1, Commodity Specification for Air, (CGA, 2011) and included in the regulations of the Occupational Safety and Health Administration 29 CFR1910.134 (i) (1) (ii) (A) through (E).
- Standby rescue persons are used whenever one-piece atmosphere-supplying suits are in use. Standby rescue personnel are also used when any combination of supplied air respiratory protection device and personnel protective equipment is in use that presents difficulty for the wearer to remove the equipment. The standby personnel are equipped with respiratory protection devices or other apparatus appropriate for the potential hazards. The standby rescue personnel observe and maintain continuous communication with the workers (visual, voice, signal line, telephone, radio, or other suitable means).
- The rescue personnel are immediately available to assist the workers in case of a failure of the air supply or for any other emergency. The Radiation Protection ~~Manager~~Supervisor specifies the number of standby rescue personnel that must be immediately available to assist all users of this type of equipment and to provide effective emergency rescue if needed.
- No objects, materials or substances (such as facial hair), or any conditions that interfere with the face-to-face piece seal or valve function, and that are under the control of the respirator wearer, are allowed between the skin of the wearer's face and the sealing surface of a tight-fitting respirator face piece.

The dose to individuals from the intake of airborne radioactive material is estimated by dividing the ambient air concentration outside the respirator by the assigned protection factor. If the actual dose is later found to be greater than that estimated initially, the corrected value is used. If the dose is later found to be less than the estimated dose, the lower corrected value may be used. Records of the respiratory protection program (including training for respirator use and maintenance) are maintained in accordance with the records management program, as described in Section 12.6.

Respiratory protection procedures are to be revised as necessary whenever changes are made to the facility, processing or equipment.

List of Figures

CHAPTER 12

CONDUCT OF OPERATIONS

12.1 ORGANIZATION

This section describes the SHINE Medical Technologies, Inc. (SHINE) organizational structure, functional responsibilities, levels of authority, and interfaces for establishing, executing, and verifying the organizational structure. The organizational structure includes internal and external functions for SHINE including interface responsibilities for multiple organizations. Management gives careful consideration to the timing, extent, and effects of organizational structure changes. The organizational structure facilitates the execution of the conduct of operations (ConOps) program. ConOps is a philosophy of working in a formalized, disciplined manner to achieve operational excellence. The ConOps program emphasizes safety in every aspect of plant operations. The organizational aspects of the radiation protection (RP) program, the production facility safety program, staffing, and selection and training of personnel will also be discussed in this section.

12.1.1 STRUCTURE

The organizational structure for the management and operation of the SHINE facility is provided in Figure 12.1-1. SHINE management levels are described as follows:

- Level 1: Individual responsible for the medical isotope facility license.
- Level 2: Individual responsible for the facility operation.
- Level 3: Individual responsible for day-to-day operation or shift.
- Level 4: Operating staff.

~~The SHINE functional organization is provided in Figure 12.1-1. The staff implementing the radiation safety function supports on-shift plant operations and interacts with Executive Management through the chain of command. The RP Manager reports directly to the Environment, Safety, and Health (ES&H) Manager. The ES&H Manager and the Quality Manager (QM) report to the Chief Operating Officer (COO) and, for independence, have a line of open communication directly to the Chief Executive Officer (CEO). The CEO reports directly to the SHINE Board of Directors. The Operations Manager (OM) and the Training Manager (TM) report to the Plant Manager (PM). The PM reports to the COO.~~Plant Operators (Level 4) report directly to the Shift Supervisors (Level 3). The RP Supervisor reports directly to the Environment, Safety, and Health (ES&H) Manager (Level 2) and has communication lines with the Shift Supervisors. The Shift Supervisors report directly to the Operations Manager (Level 2). The Review/Audit Committee reports to the Plant Manager (Level 1) and has communication lines with the Operations Manager. The ES&H Manager and Operations Manager report to the Plant Manager. The Plant Manager reports to the Chief Operating Officer (Level 1).

12.1.2 RESPONSIBILITY

12.1.2.1 SHINE MEDICAL TECHNOLOGIES, INC.

SHINE Medical Technologies, Inc. is the entity with legal responsibility for holding the Construction Permit and the facility Operating License.

12.1.2.2 CHIEF EXECUTIVE OFFICER

The CEO is responsible for the overall management and leadership of the company. The CEO provides direction to the COO and reports to the Board of Directors.

12.1.2.3 CHIEF OPERATING OFFICER

The COO reports to the CEO and is responsible for operational aspects of the company including safety, quality, environmental stewardship, regulatory affairs, and security.

12.1.2.9 ENVIRONMENT, SAFETY, AND HEALTH MANAGER

The ES&H Manager reports to the COO and is responsible for all matters regarding environment, safety, and health, including radiation protection (RP). The ES&H Manager works closely with the QM on matters involving adherence to safety requirements defined in the Quality Assurance Program Description (QAPD) and other federal, state, and local regulatory requirements. The ES&H Manager works with other managers to ensure consistent interpretations of ES&H requirements, perform independent reviews, and support facility and operations change control reviews. The ES&H Manager has the ability and responsibility to report to the CEO any safety issues that cannot be resolved at the COO level.

12.1.2.10 RADIATION PROTECTION ~~MANAGER~~SUPERVISOR

The RP ~~Manager~~Supervisor reports to the ES&H Manager and is responsible for establishing and implementing the RP program and the as low as reasonably achievable (ALARA) program, monitoring worker doses, and calibration and quality assurance of all health physics instrumentation.

12.1.3 STAFFING

SHINE provides sufficient resources in personnel and materials to safely conduct operations. Facility staffing considerations including minimum staffing levels, allocation of control functions, overtime restrictions, facility status updates during turnover between shifts, procedures, training, and availability of Senior Operators during routine operations will be defined in the FSAR.

12.1.4 SELECTION AND TRAINING OF PERSONNEL

SHINE establishes and maintains formal and informal indoctrination and training programs for personnel performing, verifying, or managing facility operation activities to ensure that suitable proficiency is achieved and maintained. The Training Manager (TM) is responsible to the PM for development and implementation of training that ensures satisfactory operational behavior and performance in the areas of nuclear, industrial, and radiological safety. “American National Standard for the Selection and Training of Personnel for Research Reactors,” American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4-2007 is used in the selection and training of personnel as applicable (ANSI/ANS, 2007). Records of personnel training and qualification are maintained.

Required minimum qualifications for facility staff will be provided in the FSAR.

Personnel who are likely to receive an occupational dose in excess of 100 mrem per year (in accordance with 10 CFR 19.12(b)) are kept informed, advised, and instructed per the requirements of 10 CFR 19.12(a)(1) through (6). Details of the training programs for facility personnel to meet the requirements of 10 CFR Part 19 will be provided in the FSAR.

~~The Senior Operator and Operator selection and licensing program will conform to 10 CFR 55 as appropriate for the SHINE facility.~~The licensed operator training program, including the requalification training program, will be developed and implemented in accordance with 10 CFR 55 as it pertains to non-power facilities (e.g., 10 CFR 55.40(d)). SHINE will comply with the requirements of 10 CFR 55 as it pertains to non-power facilities (e.g., 10 CFR 55.53(j), 10 CFR 55.53(k), 10 CFR 55.61(b)(5)).

12.1.5 RADIATION SAFETY

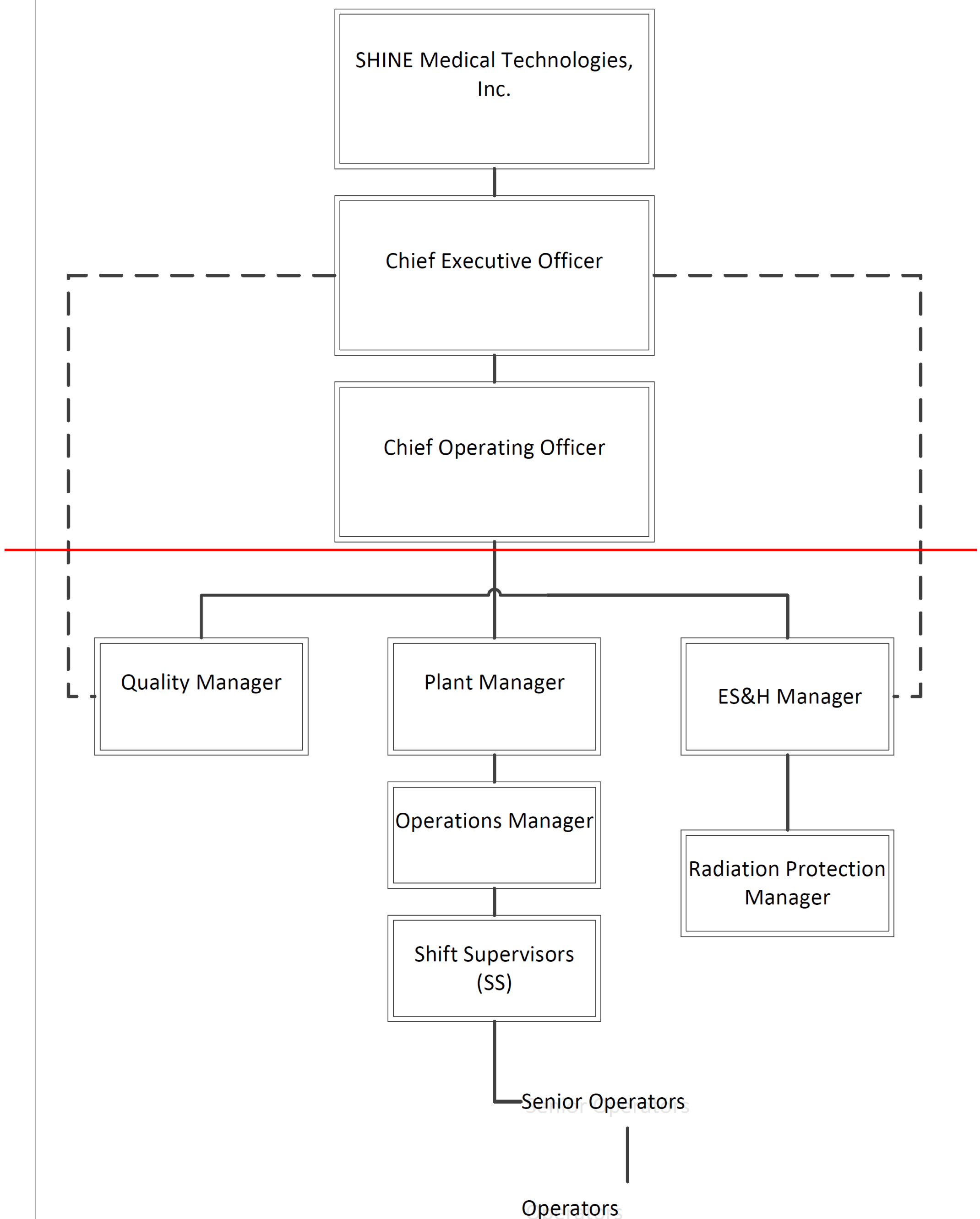
The RP program meets the requirements of 10 CFR 20, Standards for Protection Against Radiation and is consistent with the guidance provided in Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Surveys and Monitoring. The facility develops, documents, and implements the RP program commensurate with the risks posed by a medical isotope production facility. The facility uses, to the extent practicable, procedures and engineering controls, based upon sound RP principles, to achieve occupational doses to facility personnel and doses to members of the public that are ALARA. The RP staff reports to the RP ~~Manager~~Supervisor, who in turn implements the RP program by supporting ongoing activities in the SHINE facility. The RP program content and implementation are reviewed at least annually, as required by 10 CFR 20.1101(c). Sufficient resources in terms of staffing and equipment are provided to implement an effective RP program. Further details related to the authority of the radiation safety staff with respect to facility operations will be provided in the FSAR.

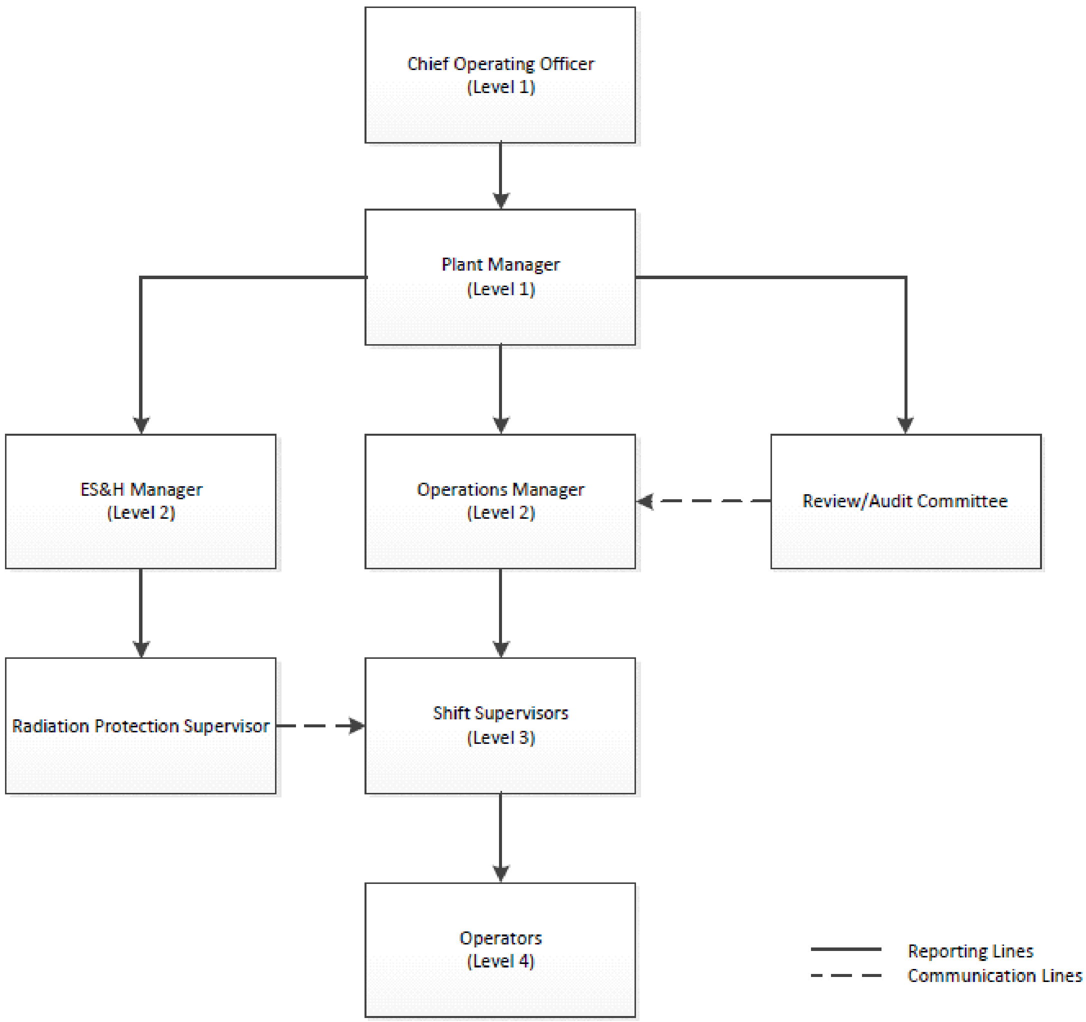
The RP program is described in greater detail in Subsection 11.1.2.

12.1.6 PRODUCTION FACILITY SAFETY PROGRAM

The production facility safety program is developed and integrated with the radiological safety program and additional facility safety programs and utilizes the methods described in 10 CFR 50. Further details of the facility safety program will be provided in the FSAR.

Figure 12.1-1 – SHINE ~~Functional~~Operational Organization Chart




Chief Operating Officer
(Level 1)
Plant Manager
(Level 1)
ES&H Manager
(Level 2)
Operations Manager
(Level 2)
Review/Audit Committee
Radiation Protection Supervisor
Shift Supervisors
(Level 3)
Operators
(Level 4)
Reporting Lines
Communication Lines

12.2 REVIEW AND AUDIT ACTIVITIES

The PM establishes review and audit committees and ensures that the appropriate technical expertise is available for review and audit activities. ~~These activities are summarized and reported to Executive Management.~~ The PM holds approval authority for review and audit activities. Independent audits of the SHINE facility are conducted periodically.

The review and audit committees will interact with facility management through the dissemination of meeting minutes and meeting reports. In accordance with Section 6.2.3 of ANSI/ANS 15.1-2007 (ANSI/ANS, 2007), SHINE will submit a written report or minutes of the findings and recommendations of the review group to Level 1 management and the review and audit group members in a timely manner after the review has been completed. In accordance with Section 6.2.4 of ANSI/ANS 15.1-2007, SHINE will immediately report deficiencies uncovered that affect nuclear safety to Level 1 management. SHINE will also submit a written report of the findings of the audit to Level 1 management and the review and audit group members within three months after the audit has been completed.

12.2.1 COMPOSITION AND QUALIFICATIONS

Audit and review committees with the appropriate expertise and experience are established and members, designated by the PM, provide the SHINE Management an independent assessment of the operation. The minimum number of committee members, qualification of committee members, and the potential use of members from outside the organization will be discussed in the FSAR.

12.2.2 CHARTER AND RULES

The charter and rules of the review and audit committees will be developed for the FSAR. The charter for the committees will address the required meeting interval (at least one per year), quorum required for meetings (not less than one-half the committee membership), issuance of meeting minutes, and voting methods.

12.2.3 REVIEW FUNCTION

At a minimum, the following items shall be reviewed:

- Determinations that proposed changes in equipment, systems, test, experiments, or procedures are allowed without prior authorization by the responsible authority.
- All new procedures and major revisions thereto having safety significance, proposed changes in production facility equipment, or systems having safety significance.
- All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.
- Proposed changes in technical specifications or license.
- Violations of technical specifications or license. Violations of internal procedures or instructions having safety significance.
- Operating abnormalities having safety significance.

- Reportable occurrences.
- Audit reports.
- 10 CFR 50.59 safety reviews.

Upon completion of a review, a written report of any findings and recommendations of the committee shall be provided to SHINE Executive Management.

12.2.4 AUDIT FUNCTION

All aspects of facility operations, including the RP and laboratory program, the emergency, physical security, and operator requalification plans will be audited, at a minimum of every two years. These areas do not have to be audited at the same time, but all will be audited within the designated intervals. Each audit will have its own audit plan. Discussions with personnel and observation of operations will be used as appropriate. In no case shall the individual immediately responsible for the area perform an audit in that area. SHINE will work to establish relationships with other entities to participate in audits of the facility. The following items are examples of activities that will be audited:

- ~~Facility operations for conformance to the technical specifications and applicable license conditions.~~
- ~~The retraining and requalification program for the operating staff.~~
- ~~The results of action taken to correct those deficiencies that may occur in the production facility equipment, systems, structures, or methods of operations that affect nuclear safety.~~
- ~~The SHINE facility emergency plan and implementing procedures.~~
- Facility operations for conformance to the technical specifications and applicable license conditions: at least once per calendar year (interval between audits not to exceed 15 months).
- The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months).
- The results of action taken to correct those deficiencies that may occur in the production facility equipment, systems, structures, or methods of operations that affect nuclear safety: at least once per calendar year (interval between audits not to exceed 15 months).
- The SHINE facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies identified during the audit will be entered into the corrective action program. Deficiencies uncovered that affect nuclear safety shall immediately be reported to Level 1 management. A written report of the findings of the audit shall be submitted to Level 1 management and the review and audit group members within three months after the audit has been completed.

12.3 PROCEDURES

Operating procedures provide appropriate direction to ensure that the facility is operated normally within its design basis, and in compliance with technical specifications. Operating procedures are written, reviewed, approved by appropriate management, as well as controlled and monitored to ensure that the content is technically correct and the wording and format are clear and concise.

SHINE procedures are prepared, approved, revised, canceled, and implemented in accordance with the procedure program. Document Control (DC) maintains the Master Procedures List and ensures that revisions are documented appropriately, approved for release by authorized personnel, and distributed for use at the location where the prescribed activity is performed. DC also retains and distributes procedures in accordance with DC procedures for SHINE. The process required to make changes to procedures, including substantive and minor permanent changes, and temporary deviations to deal with special or unusual circumstances during operation is in compliance with ANSI/ANS 15.1-2007 (ANSI/ANS, 2007). ~~A minimum list of procedural topics will be provided with the FSAR.~~

In accordance with Section 6.4 of ANSI/ANS 15.1-2007, SHINE shall prepare, review, and approve written procedures for the following basic topics:

1. startup, operation, and shutdown of the IU;
2. target solution fill, draining, and movement within the SHINE facility;
3. maintenance of major components of systems that may have an effect on nuclear safety;
4. surveillance checks, calibrations and inspections required by the technical specifications;
5. personnel radiation protection, consistent with applicable regulatory guidance. The procedures shall include management commitment and programs to maintain exposures and releases as low as reasonably achievable in accordance with applicable guidance;
6. administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect nuclear safety;
7. implementation of required plans (e.g., emergency, security); and
8. use, receipt, and transfer of byproduct material.

The specific procedures within these topic areas will be developed in accordance with Section 2.5 of the SHINE QAPD.

The procedures shall be reviewed by the SHINE review and audit committee and approved by Level 2 management or designated alternates, and such reviews and approvals shall be documented in a timely manner.

Substantive changes to procedures related to the activities listed above shall be made effective only after documented review by the SHINE review and audit committee and approval by Level 2 management or designated alternates. Minor modifications to the original procedure that do not change their original intent may be made by Level 3 management or higher, but the modifications must be approved by Level 2 or designated alternates. Temporary deviations from the procedures may be made by the responsible shift manager (Level 3 management) or higher individual present, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours or the next working day to Level 2 management or designated alternates.

The SHINE policy on use of procedures is documented and clearly understood by all applicable SHINE personnel. The extent of detail in a procedure is dependent on the complexity of the task; the experience, education, and training of the users; and the potential significance of the consequences of error. The process for making changes and revisions to procedures is documented. A controlled copy of all operations procedures is maintained in the control room or equivalent area. Activities and tasks are performed in accordance with approved implementing procedures.

12.9 QUALITY ASSURANCE

~~SHINE QA-1~~ 2000-09-01, Quality Assurance Program Description (QAPD), is based on ANSI/ANS 15.8–1995 (R2005) (ANSI/ANS, 1995), "Quality Assurance Program Requirements for Research Reactors," with guidance from Regulatory Guide 2.5, Revision 1.

The SHINE QAPD is provided in Appendix 12C.

Because the SHINE facility is being designed to withstand external events such as tornado, seismic, or man-made external events, scenarios that involve multiple IUs are not analyzed further. In addition, several internal events were eliminated as possible MHAs due to the design of the facility. Because production piping is located in covered, concrete trenches that are designed to contain loss of inventory and drain to criticality-safe sumps, this event was eliminated as a possible MHA for the RPF.

The postulated MHA in the IF is a ~~large rupture of the TSV dump tank resulting in a complete release of the target solution and fission product inventory into one IU cell~~release of irradiated target solution to the IU cell as a result of a loss of TSV integrity. It is assumed that the PSB and the subcritical assembly support structure (SASS) have breached. The presence of the light water pool is ignored for the purposes of the IF postulated MHA. The loss of TSV integrity encompasses either a TSV or TSV dump tank rupture. Due to the robust design of the TSV ~~dump tank~~, a rupture is not considered to be a credible event. However, for the purpose of the MHA analysis, it is postulated that a breach of the TSV ~~dump tank~~ occurs. Note that the MHA assumes that only one IU is compromised. Each IU cell is constructed with reinforced concrete walls and ceiling. Because of the robust design of each IU cell and the design against external events, events capable of rupturing more than one TSV ~~dump tank~~ inside of the IF are not considered to be credible.

The postulated MHA in the RPF area is a failure of the five noble gas removal system (NGRS) storage tanks with the inventory released inside the noble gas storage cell. Because the noble gas storage cell is designed as a robust structure to provide shielding and confinement, the release of noble gas is confined to the storage cell and the RCA ventilation Zone 1 (RVZ1) system piping, with some leakage assumed through storage cell penetrations.

For both MHAs considered above, (a complete loss of inventory of a TSV into the IU cell or a complete release of the NGRS inventory into the noble gas storage cell) the following initial conditions are assumed:

- Maximum radioisotope inventories in the TSV ~~dump tank~~ and the NGRS.
- The robust design of each IU cell provides isolation between IUs, therefore it is assumed that only one IU is affected by the event.
- IU cell penetrations for piping, ducts and electrical cables and airlocks are sealed within design specifications to limit the release of radioactive materials from the IU cell.
- The RVZ1 is operating normally at the time of the IE with:
  - One fan in operation and a second fan in standby mode.
  - Two passive multi-filter housing units containing two-stage high-efficiency particulate air (HEPA) filtration and single-stage carbon absorbers (Section 9a2.1).
  - Ventilation inside the IU cells, noble gas storage cell, and hot cells of both the IF and RPF.
- RVZ1 bubble-tight isolation dampers (normally open/fail closed) are installed at the IU cell, noble gas storage cell, and hot cells, for both supply and exhaust. These are designed to be closed both automatically and manually on high radiation. Both the ventilation supply and exhaust penetrations have redundant bubble-tight dampers.
- The TSV reactivity protection system (TRPS) is functioning as designed during operating conditions. Therefore, the neutron driver is deactivated and fusion and fission reactions are terminated.

Acronyms and Abbreviations

Acronym/Abbreviation	**Definition**
°F	degrees Fahrenheit
°C	degrees Celsius
μS/cm	micro-Siemens per centimeter
AEA	Atomic Energy Act of 1954
ac.	acre
AHA	acetohydroxamic acid
[Proprietary Information]	[Proprietary Information]
[Proprietary Information]	[Proprietary Information]
[Proprietary Information]	[Proprietary Information]
Btu/hr	british thermal units per hour
Btu/scf	british thermal units per standard cubic feet
~~[Proprietary Information]~~Ce	~~[Proprietary Information]~~Cerium
cfm	cubic feet per minute
CFR	Code of Federal Regulations
Ci	curies
CO_2	carbon dioxide
CP	Construction Permit
~~[Proprietary Information]~~Cs-137	~~[Proprietary Information]~~cesium-137
d or D	deuterium
D-T	deuterium-tritium
DSSI	Diversified Scientific Services, Inc.
EPA	U.S. Environmental Protection Agency
ER	Environmental Report
ES	EnergySolutions
FDA	U.S. Food and Drug Administration
ft.	feet

~~[Proprietary Information]~~The ion exchange resin used for removal of cesium-137 (Cs-137) and cerium (Ce) has a high capacity for Cs-137 capture and will be changed out based on curie limits at the receiving facility and also based on shipping limits. The spent ~~[Proprietary Information]~~resins are solidified in a shielded waste processing hot cell. The used resin is classified as GTCC waste and is shipped as Type B to an off-site location for long-term storage at WCS.

As discussed above, the target solution cleanup system uses an anion exchange column to remove technetium and iodine. When the anion exchange resin is replaced, the spent resin is solidified on-site and sent off-site for disposal (WCS in Andrews, Texas).

There will be no solid waste disposal at the SHINE site.

19.2.5.3.2 Liquid Radioactive Waste System

Liquid waste discharged from the various processes at the SHINE facility (other than spent solvent) are combined into one of two tanks. Two tanks are needed to allow liquid waste to decay and also so that a somewhat consistent radiological environment exists for waste processing. Once the first tank is filled the other tank will begin to fill. At this point the pH is adjusted so that the waste can be passed through an ~~[Proprietary Information]~~ion exchange resin for removal of Cs-137 and Ce-144/Pr-144. This allows the majority of the liquid stream to become Class A waste. This cleaned-up material is then sent to an evaporator for volume reduction. The evaporator overheads are reused and the bottoms are solidified and shipped to ES for final disposal. The spent resin treatment is discussed in the section above. No liquid radioactive waste is discharged from the SHINE facility.

The spent solvent is not a RCRA waste and is replaced once per year. The solvent is sent to a processor (Diversified Scientific Services, Inc [DSSI], in Kingston, Tennessee) for thermal treatment.

[Proprietary Information] This waste is classified as Class B waste and is shipped as Type B to WCS in Andrews, Texas.

[Proprietary Information] The waste is solidified in a hot cell using Portland cement. Some additives may be required based on the final chemistry of incoming resin and precipitate. These shipments are Type B shipments.

There will be no liquid waste disposal at the SHINE site.

19.2.5.4 Proposed Hazardous Material Disposal Activity

The only hazardous (or potentially hazardous) materials are [Proprietary Information] and the zeolite beds. Although small quantities of [Proprietary Information] is expected to pass TCLP, and is not considered hazardous waste. Waste streams with a hazardous component are mixed low-level waste such as the zeolite beds and are handled as described in Subsection 19.2.5.3.1.

Table 19.2.5-1 Estimated Type and Quantity of Radioactive Wastes Associated with the SHINE Facility (Sheet 1 of 3)

Description	Matrix	Class as Generated	Contents	Volume	Volume as shipped (ft^3)	55-gallon drum equivalent as shipped	Shipment Type	Number of Shipments/ yr	Destination
Neutron Generator	Solid	A	Activated metal parts	4338 ft^3/yr	4338	590	LSA	3.00	ES
Extraction Columns	Solid	A	Stainless resin columns						
Class A Trash	Solid	A	PPE, Mo-99 purification glassware, filters, etc						
Spent Solvent	Liquid[(a)]	A	n-dodecane, tributyl phosphate	22 gallons/ yr	--	0.4	LSA	1.00	DSSI
Tc/I columns	Resin	C	Resin	16 gallons/ yr	23	3.1	Type B	0.3	WCS
Zeolite Beds	Solid	GTCC	Silver coated beds	0.4 ft^3/yr	0.4	0.05	Type B	1.00	WCS
~~[Proprietary Information]~~ Cs/Ce Media	~~[Proprietary Information]~~ Resin	GTCC	Resin	16 gallons/ yr	23	3.1	Type B	0.3	WCS
[Proprietary Information]	[Proprietary Information]	B	[Proprietary Information]	295 gallons/yr	79	11	Type B	1.00	WCS