CHAPTER 11

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

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ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
CAAS	criticality accident alarm system
CAM	continuous air monitor
CAS	continuous air sampler
CDBEM	carbon delay bed effluent monitor
CEDE	committed effective dose equivalent
CEMP	Community Environmental Monitoring Program
CEO	Chief Executive Officer
Ci	curies
Ci/yr	curies per year
cm	centimeter
COO	Chief Operating Officer
D/Q	ground level deposition factor
DAC	derived air concentration
DOT	U.S. Department of Transportation

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
DQO	data quality objectives
EPA	U.S. Environmental Protection Agency
ft^3/yr	cubic feet per year
HEPA	high efficiency particulate air
hr	hour
HRA	high radiation area
I-131	iodine-131
IF	irradiation facility
IU	irradiation unit
IXP	iodine and xenon purification and packaging
km	kilometers
Kr	krypton
kW	kilowatt
LEU	low-enriched uranium
LLD	lower level of detection

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
LLW	low level waste
LSA	low specific activity
LSC	liquid scintillation counter
LWPS	light water pool system
MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols Manual
MEI	maximum exposed individual
MEPS	molybdenum extraction and purification system
MLLW	mixed low level waste
Mo	molybdenum
Mo-99	molybdenum-99
mrem	millirem
mrem/hr	millirem per hour
mrem/yr	millirem per year
mSv	millisievert
N-16	nitrogen-16
NDAS	neutron driver assembly system

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
NFDS	neutron flux detection system
PCLS	primary closed loop cooling system
PNNL	Pacific Northwest National Laboratory
PPE	personal protective equipment
PSB	primary system boundary
PVVS	process vessel vent system
RA	radiation area
RAM	radiation area monitor
RCA	radiologically controlled area
RCRA	Resource Conservation and Recovery Act
REMP	radiological environmental monitoring program
RLWI	radioactive liquid waste immobilization
RLWS	radioactive liquid waste storage
RPF	radioisotope production facility
RSC	Radiation Safety Committee
RSICC	Radiation Safety Information Computational Center

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
RVZ1	radiological ventilation zone 1
RVZ2	radiological ventilation zone 2
RWP	radiation work permit
SASS	subcritical assembly support structure
SCAS	subcritical assembly system
SRM	stack release monitor
SSC	system, structure, and component
Sv	sievert
TCLP	toxicity characteristic leaching procedure
TEDE	total effective dose equivalent
TOGS	TSV off-gas system
TPS	tritium purification system
TSPS	target solution preparation system
TSSS	target solution storage system
TSV	target solution vessel
U-235	uranium-235

ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
URSS	uranium receipt and storage system
VHRA	very high radiation area
VTS	vacuum transfer system
WAC	waste acceptance criteria
WCS	Waste Control Specialists
χ/Q	annual average relative atmospheric concentration
Xe	xenon

CHAPTER 11 – RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 RADIATION PROTECTION

11.1.1 RADIATION SOURCES

The SHINE facility is designed to generate molybdenum-99 (Mo-99) for use as a medical isotope. The process of producing Mo-99 involves irradiating a uranyl sulfate target solution with a neutron source in a subcritical assembly to cause fission. Irradiation of the target solution creates Mo-99 along with other radioactive fission and activation products. When the irradiation cycle is complete, the radioactive materials are transferred to various locations in the facility to complete the separation and purification processes. This section identifies sources of radiation and radioactive materials received, used, or generated in the facility; sources and the nature of airborne, liquid or solid radioactive materials; and the type of radiation emitted (alpha, beta, gamma, and neutron).

Analysis has been performed that quantifies the radionuclide inventory for normal operations in the SHINE facility. The highest radionuclide inventory for one target solution batch exists in the target solution vessel (TSV) at the end of the irradiation cycle. As the target solution is processed in the facility for Mo-99 and other medical isotope extraction, solution adjustments, and waste handling, radiation sources are transferred within the facility by means of pipes in shielded trenches.

There are two scenarios with assumptions listed in Table 11.1-1: nominal and safety basis. The nominal parameter values or ranges are the best estimate operating conditions for full power operation of the facility. The safety basis parameter values define the bounding radionuclide inventory relative to the TSV, TSV dump tank, and supercell.

The safety basis inventories throughout the facility are generated by using the limiting values for each parameter to maximize the individual inventories. This includes using bounding values for element partitioning during the extraction process. This approach of maximizing inventories at each location results in an overall facility fission product inventory that is greater than originally generated in the irradiation process. This ensures that the individual safety basis inventories are bounding when being used to calculate releases for the safety analysis but makes them unsuitable for use in analyzing normal operations.

Operation of the TSV results in the production of radioactive fission products and actinides predominantly through neutron capture in uranium. Table 11.1-2 provides a summary of the results for total activity in curies (Ci) from actinides and fission products contained within each TSV batch of target solution after [ ]$^{PROP/ECI}$ of irradiation, [ ]$^{PROP/ECI}$ nominal cycles or [ ]$^{PROP/ECI}$ safety basis cycles. The “at shutdown” values represent the activity contained within the target solution immediately after shutdown of the neutron driver. The “pre-extraction” values are the target solution activity when it is ready to be transferred from the TSV dump tank in the irradiation unit (IU) cell to one of the supercells in the radioisotope production facility (RPF) to begin the molybdenum extraction process. This represents the maximum expected activity for a target solution batch as it is processed through the RPF. For the nominal inventory, the “post extraction” values are the activity remaining in the target solution following extraction of Mo and other elements according to best estimate partitioning fractions. For the safety basis inventory, only noble gases were removed during extraction, at bounding (low) element partitioning fractions.

Table 11.1-3 lists the activity associated with the radionuclides listed in NUREG/CR-4467, Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions (USNRC, 1986) for the nominal and safety basis radionuclide inventories after [ ]$^{PROP/ECI}$ of irradiation and the subsequent decay time in the TSV dump tank. At this time, it is ready to be pumped into the supercell to begin the molybdenum extraction and fission product removal processes. The cycle and decay times used for the radionuclide inventory generation are listed in Table 11.1-1.

SHINE uses the following radiation area designations, as defined in 10 CFR 20, including consideration for neutron and gamma dose rates:

- Unrestricted Area means an area to which access is neither limited nor controlled by SHINE. This would be the area beyond the site boundary.
- Radiation Areas (RAs) are those accessible areas in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 millirem (mrem) in 1 hour (hr) at 30 centimeters from the radiation source or from any surface that the radiation penetrates.
- High Radiation Areas (HRAs) are those accessible areas in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 mrem in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.
- Very High Radiation Areas (VHRAs) are those accessible areas 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 in 1 hour at 1 meter from the radiation source or 1 meter from any surface that the radiation penetrates.

The SHINE facility is designed and constructed so that the measurable dose rate in the unrestricted area due to activities at the plant are less than the limits of 10 CFR 20.1301(a)(2).

The radiation shielding is designed to ensure that during normal operation internal facility radiation dose rates are consistent with as low as reasonably achievable (ALARA) radiological practices required by 10 CFR 20. The goal for the normal operations dose rate for normally occupied locations in the facility is 0.25 mrem/hr at 30 centimeters from the surface of the shielding. Radiation levels may rise above the 0.25 mrem/hr level during some operations such as tank transfers. At full-power operation of the eight units, portions of the normally occupied area in IF and RPF exceed the 0.25 mrem/hr goal but remain below 5 mrem/hr, except in small sections above the pipe trench during solution transfers. These dose rates were calculated using the maximum specified shield plug gap sizes, minimum density shielding materials, and the nominal inventories for full power operation.

A tabulation of normally and transient-occupied areas, dose rates, and designations is provided in Table 11.1-4. Figure 11.1-1 provides the probable radiation area designations, above grade, within the radiologically controlled area (RCA) at the SHINE facility.

Procedures for transient access to shielded vaults, cells, and rooms ensure doses are maintained ALARA by addressing the following:

- job planning,
- radiation protection coverage,
- survey techniques and frequencies,

- training of workers,
- pre-work briefing,
- frequency for updating radiation work permits or their equivalent, and
- placement of measuring and alarming dosimeters.

Shielded vaults, cells, and rooms designated as high radiation areas or very high radiation areas as denoted in Figure 11.1-1 are not normally occupied when those conditions exist. Administrative procedures address the management oversight and specific control measures needed for entry into high radiation areas and very high radiation areas, if it is ever necessary to do so. The procedures include the process for gaining entry to these areas, such as the control and distribution of keys.

Typical transient access for maintenance or other necessary work to the shielded vaults, cells, and rooms that are usually high radiation areas or very high radiation areas is normally performed after dose rates have been reduced to at least the level of a radiation area. This is done by removing the radioactive materials or changing the conditions (such as shutting down the accelerator in an IU cell), using temporary shielding, and waiting for sufficient decay.

Major radiation sources in the facility originate in the target solution. At the end of the TSV irradiation cycle, irradiated target solution is transferred to one of the three extraction cells for processing. Off-gas that is purged from the primary system boundary (PSB) is sent to the process vessel vent system (PVVS), where it travels through carbon guard beds and a series of carbon delay beds to allow for capture of iodine and decay of short-lived noble gas nuclides before being released through the facility exhaust stack. Facility special nuclear material (SNM) inventories are tabulated in Table 4b.4-1.

The three sections below describe the major radiation sources in the facility. Other radiological sources in the facility are bounded by the fission product source coming from the TSV described in Subsection 11.1.1.2.

11.1.1.1 Airborne Radioactive Sources

Radioactive sources that could become airborne at the SHINE facility are primarily tritium and radioactive gases produced as a byproduct of the Mo-99 production process. The systems handling gaseous radioactive materials include the tritium purification system (TPS) and the TSV off-gas system (TOGS), both located in the irradiation facility (IF) area; and the PVVS and vacuum transfer system (VTS) located in the RPF. These airborne radioactive materials are contained within closed systems consisting of piping components and tanks. Table 11.1-5 provides information on the various locations, types, and expected dose rates from gaseous radioactive sources.

Argon-41 is produced in the IU cells during irradiation. Due to the low flow rate out of the primary confinement boundary to radiological ventilation zone 1 (RVZ1), most argon-41 decays prior to being released. Approximately 0.02 curies per year (Ci/yr) of argon-41 are released to the environment through the facility stack.

Nitrogen-16 is produced within the primary cooling loop and the light water pool. Dose rates from these sources are mitigated by delay tanks and biological shielding that limits radiation dose to occupied areas adjacent to the shielding.

The design of the SHINE facility maintains airborne radioactive material at very low concentrations in normally occupied areas. Confinement and ventilation systems are designed to protect workers from sources of airborne radioactivity during normal operation and minimize worker exposure during maintenance activities, keeping with the ALARA principles outlined in 10 CFR 20.

Although most process gas systems within the facility are maintained below atmospheric pressure, some leakage of process gases is expected due to the difference in partial pressure between the system and the surrounding environment. A conservative best estimate of airborne releases due to normal operation and maintenance was performed to estimate derived air concentrations (DACs) for the facility.

Leakage from process systems was estimated based on the number of components and fittings, achievable leak tightness per fitting, permeation through equipment, and partial pressures of airborne radionuclides. For processes in hot cells that require routine disconnection of components (e.g., extraction columns) special fittings are used to minimize process leakage.

The effects of the confinement systems are incorporated into the analysis. The results of the evaluation, broken down into particulates, halogens, noble gases, and tritium, are provided in Table 11.1-6. These values provide a conservative best estimate of the facility DACs. Figure 11.1-2 provides the DAC zoning map for the facility, using the following definitions:

- Zone 1 (< 1.0 DAC);
- Zone 2 (1.0 – 10 DAC); and
- Zone 3 (> 10 DAC).

Gaseous activity from the TSV and process operations is routed through the PVVS which includes carbon delay beds to allow for airborne radionuclides to decay to low enough levels such that normal releases are below the 10 CFR 20 limits. Additional airborne release pathways are RVZ1 ventilation of the facility hot cells, flow out of the primary confinement boundary to RVZ1, and radiological ventilation zone 2 (RVZ2) ventilation of any leakage to the general area (material evaluated for the DAC). These additional pathways do not pass through the carbon delay beds but do contain filters as described in Subsection 9a2.1.1. Table 11.1-7 lists key parameters used in the normal release calculation. Tritium releases that are treated by TPS are negligible in comparison to tritium releases to the general area due to maintenance and leakage and are not included in Table 11.1-7 or Table 11.1-8.

Annual off-site doses due to the normal operation of the SHINE facility have been calculated using the computer code GENII2 (PNNL, 2012). The GENII2 computer code was developed for the Environmental Protection Agency (EPA) by Pacific Northwest National Laboratory (PNNL) and is distributed by the Radiation Safety Information Computational Center (RSICC). Annual average relative atmospheric concentration (χ/Q) values were determined using the methodology in Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (USNRC, 1977) with the meteorological data in Section 2.3. The χ/Q values for the maximally exposed individual (MEI), which is the nearest point on the site boundary, and the nearest full-time resident are 7.1E-5 sec/m^3 and 5.3E-6 sec/m^3, respectively.

Table 11.1-8 contains the estimated annual release from maintenance and normal operation of eight irradiation units. The release is comprised of release inventories from the four airborne release pathways described above: PVVS, hot cells, primary confinement boundary, and material leaked to the general area. The dominant source term is the process gases released through PVVS. Only nuclides with greater than 1 Ci/yr released are included in the table.

The dose analysis considered the release of airborne radionuclides and exposure to off-site individuals through direct exposure and potential environmental pathways, such as leafy vegetable ingestion, meat ingestion, and milk ingestion. The analysis considered variations in consumption and other parameters by age group. The estimated annual doses at the MEI and the nearest resident are 3.9 mrem and 0.3 mrem, respectively, which are less than the limit in 10 CFR 20.

Calculational methodologies related to accidental releases of airborne radioactive sources are discussed in Chapter 13.

11.1.1.2 Liquid Radioactive Sources

There are numerous locations within the SHINE facility where the presence of radioactive liquids results in a source of radiation. These sources (except for as noted below) are derived from the irradiated uranyl sulfate target solution as it is being processed through the facility. The first exception is the primary cooling water, which carries nitrogen-16 and other activation products as it is pumped through the primary closed loop cooling system (PCLS). The second exception is the production of low-activity fresh uranyl sulfate target solution. These radioactive materials are contained within closed systems consisting of piping components and tanks.

In addition, there are two locations where tritium is expected to collect due to operation of the neutron driver assembly system (NDAS). These are the light water pool and the oil used in the NDAS pumps. The small quantities of tritium released into the IU cell by permeation through and leakage from the NDAS components is expected to be converted to tritiated water and slowly increase the tritium concentration in the pool water. The oil used in the NDAS pumps is in direct contact with the tritium in the accelerator, causing it to become contaminated with tritium over time. Table 11.1-9 provides information on the various locations, types, and expected doses from liquid radioactive sources.

Liquid radioactive wastes generated at the facility are generally solidified and shipped to a disposal facility. Table 11.2-1 contains a list of liquid radioactive waste generated at the facility including the annual quantities and disposal destinations. Radioactive liquid discharges from the SHINE facility to the sanitary sewer are made in accordance with 10 CFR 20.2003 and 10 CFR 20.2007. See Section 11.2 for additional information on liquid discharges from the RCA.

11.1.1.3 Solid Radioactive Sources

Solid radioactive sources exist in several locations in the SHINE facility. Fresh, low enriched uranium is received at the facility in the form of uranium metal or uranium oxide that has been enriched to a nominal 19.75 percent by weight in uranium-235 (U-235). If uranium metal is received, it is converted to uranium oxide and then to a liquid uranyl sulfate solution. Other solid radioactive sources are listed in Table 11.2-1 and include spent extraction columns from the molybdenum extraction process, glassware, spent filters, and solidified liquid waste.

The natural uranium neutron multiplier is located in the subcritical assembly. The uranium interacts with the neutron flux producing both activation products and fission products that are retained within the metal structure.

In addition, metal components in the IU cell are activated and components of the TOGS contain radioactive material. The subcritical multiplication sources for the subcritical assemblies are also located in the IU cell.

These solid radioactive sources are contained within IU cells, shielded cells, hot cells, or preparation areas within the RCA of the facility. Table 11.1-10 provides information on the major solid radioactive sources including their location and activity. The radionuclide inventory in the solid waste system is a function of the TSV system operation.

A list of solid radioactive wastes including annual quantities and disposal destinations is provided in Table 11.2-1.

Disposal of solid radioactive waste with respect to storage, monitoring, and management is discussed in Section 11.2.

11.1.1.4 Technical Specifications

Certain material in this section provides information that is used in the technical specifications. This includes limiting conditions for operation, setpoints, design features, and means for accomplishing surveillances. In addition, significant material is also applicable to, and may be referenced by, the bases that are described in the technical specifications.

11.1.2 RADIATION PROTECTION PROGRAM

The radiation protection program protects the radiological health and safety of workers and members of the public and complies with the regulatory requirements in 10 CFR 19, 20, and 70.

11.1.2.1 Commitment to Radiation Protection Program Implementation

SHINE has established a radiation protection program with the specific purpose of protecting the radiological health and safety of workers and members of the public. The objectives of the program are to prevent acute radiation injuries (non-stochastic or deterministic effects) and to limit the potential risks of probabilistic (stochastic) effects (which may result from chronic exposure) to acceptable levels. The SHINE radiation protection program was developed and is implemented commensurate with the risks posed by a medical isotope facility. The program contains the SHINE management policy statement to maintain occupational and public radiation exposures ALARA.

The radiation protection program meets the requirements of 10 CFR 20, Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2, Revision 1, Administrative Practices in Radiation Surveys and Monitoring (USNRC, 2011), and ANSI/ANS 15.11-2016, Radiation Protection at Research Reactor Facilities (ANSI/ANS, 2016). Procedures and engineering controls are based upon sound radiation protection principles to achieve occupational doses to on-site personnel and doses to members of the public that are ALARA. The radiation protection program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c).

The radiation protection program includes written procedures, periodic assessments of work practices and internal/external doses received, work plans, and the personnel and equipment required to implement the ALARA goal. Protection of plant personnel requires (a) surveillance of and control over the radiation exposure of personnel and (b) maintaining the exposure of personnel not only within permissible limits, but also within ALARA philosophy and exposure goals.

SHINE's administrative personnel exposure limits for radiation workers are set below the limits specified in 10 CFR 20. This provides assurance that regulatory radiation exposure limits are not exceeded and that the ALARA principle is emphasized. Administrative exposure limits are provided in Table 11.1-11.

The radiation exposure policy and control measures for personnel are established in accordance with requirements of 10 CFR 20 and the guidance in the following regulatory guides:

- Regulatory Guide 8.10, Revision 2, Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (USNRC, 2016)
- Regulatory Guide 8.13, Revision 3, Instruction Concerning Prenatal Radiation Exposure (USNRC, 1999)
- Regulatory Guide 8.29, Revision 1, Instruction Concerning Risks from Occupational Radiation Exposure (USNRC, 1996)

The SHINE corrective action process is implemented if (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; (2) if an incident results in airborne occupational exposures exceeding the administrative limits; or (3) the dose limits in 10 CFR 20 are exceeded.

Information developed from reportable occurrences is tracked in the corrective action program and is used to improve radiation protection practices, decreasing the probability of similar incidents.

11.1.2.1.1 Responsibilities of Key Program Personnel

The key personnel responsible for implementing the radiation protection program are shown in Figure 11.1-3 and are discussed below. Chapter 12 discusses the SHINE organization and responsibilities of key management personnel in further detail.

Chief Executive Officer

The Chief Executive Officer (CEO) is responsible for the overall management and leadership of the company.

Chief Operating Officer

The Chief Operating Officer (COO) reports to the CEO and is responsible for overall company operations.

Vice President Regulatory Affairs & Quality

The Vice President Regulatory Affairs & Quality reports to the CEO and is responsible for licensing and quality activities.

Quality Manager

The Quality Manager reports to the Vice President Regulatory Affairs & Quality and is responsible for assuring compliance with regulatory requirements and procedures.

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 designates the authority to approve procedures related to personnel radiation protection to the Radiation Protection Manager in accordance with the guidance provided in ANSI/ANS-15.1-2007 (ANSI/ANS, 2007). The Plant Manager reports to the COO.

Radiation Protection Manager

The Radiation Protection Manager is responsible for implementing the radiation protection program. The Radiation Protection Manager reports directly to the Plant Manager, independent from facility operations. The Radiation Protection Manager has direct access to executive management for matters involving radiation protection. The Radiation Protection Manager and radiation protection personnel are responsible for:

- Establishing the radiation protection program.
- Generating and maintaining procedures associated with the program.
- Ensuring that ALARA is incorporated into procedures and practiced by personnel, including stopping work when unsafe practices are identified.
- Ensuring the efficacy of the program is reviewed and audited for compliance with NRC and other governmental regulations and applicable regulatory guides.
- Modifying the program based upon experience, facility history, regulatory updates, and changes to guidance documents.
- Adequately staffing the Radiation Protection Department to implement the radiation protection program.
- Ensuring that the occupational radiation exposure dose limits of 10 CFR 20 are not exceeded under normal operations.
- Ensuring administrative radiation dose limits are not exceeded without prior approval from the Radiation Safety Committee.
- Establishing and maintaining an ALARA program.
- Demonstrating, where practical, familiarity and reasoning associated with improvements in ALARA principles and practices, including modifications that were considered and implemented.
- Establishing and maintaining a Respiratory Protection Program.
- Establishing and maintaining the Radiological Environmental Monitoring Program.
- Establishing and maintaining a Radioactive Waste Management Program.
- Monitoring worker doses, both internal and external.

- Assuring that the proper radiation protection instrumentation, equipment, and supplies are available at workplaces, in good working order, and are used properly.
- Ensuring calibration and quality assurance of health physics associated radiological instrumentation.
- Establishing and maintaining a radiation safety training program for personnel working in radiologically controlled areas.
- Posting restricted areas and, within these areas, posting radiological areas, as required by the radiation protection program (e.g., airborne radioactivity area, high radiation area, contamination area).
- Informing management of any radiation protection concerns.

Operations Manager

The Operations Manager is responsible for operating the facility safely and in accordance with facility 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 responsible for performing their work activities in a safe manner. SHINE has established policies, procedures and practices to ensure that personnel can work safely in the facility. The policies, procedures and practices implement rules and regulations intended to ensure workers and the public are protected from specific hazards encountered at the facility. 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 Radiation Protection Program Staffing and Qualifications

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, at a minimum, a Radiation Protection Manager and radiation control technicians.

Staff selection and qualification are addressed in Chapter 12. The Radiation Protection Manager selection and qualification is consistent with the requirements for a Level 2 position. Radiation control technicians are considered “Other Technical Personnel,” as described in Subsection 12.1.4.

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 to maintain a high standard of radiation protection during facility operations. The RSC oversees activities at the SHINE facility to protect personnel from unnecessary radiation exposure, prevent contamination of natural resources, and to ensure compliance with state and federal regulations governing the possession, use, and disposal of radioactive materials. The RSC meets periodically, but at least annually, to monitor facility radiological performance and ALARA implementation, review proposed changes to the radiation protection program, identify trends, and set ALARA policy and goals for the facility. The RSC reviews the results of audits and regulatory inspections, worker suggestions, reportable occurrences, and exposure incidents. The RSC assesses changes to the facility for the effect on the radiation protection program and the license.

The Radiation Protection Manager chairs the RSC. The RSC Charter defines the purposes, functions, responsibility, composition, qualifications, quorum, meeting frequency, and reporting requirements of the RSC.

11.1.2.1.5 Commitment to Written Radiation Protection Procedures

Radiation protection procedures are 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 or designee to incorporate facility or operational changes.

Radiation protection procedures provide direction for the following activities:

- Facility radiation monitoring, including surveys, personnel monitoring, and sampling and analysis of solid, liquid and gaseous wastes processed or released from the facility
- Calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, personnel radiation monitors and portable instruments
- Access control, radiological posting, and monitoring of radiological work activities
- Radioactive materials handling and shipment
- Contamination control
- Control of exposures and ALARA implementation
- Control of instrument alarm setpoints
- Administration of the radiation work permit (RWP) process

Radiation protection procedures undergo technical verification and review to ensure compliance with regulatory requirements, applicable license conditions and the radiation protection program, as well as conformance with industry standard practices, as applicable. Radiation protection procedures are reviewed at least once every three years in accordance with the guidance in Regulatory Guide 8.10. Radiation protection procedures related to personnel radiation protection are reviewed by the SHINE Review and Audit Committee.

Work performed in radiologically controlled areas is performed in accordance with the RWP process. The RWP specifies radiological controls for intended work activities and provides written authorization for entry into and work within Radiation Areas, High Radiation Areas, Very High Radiation Areas, Contamination Areas and Airborne Radioactivity Areas. The RWP informs workers of area radiological conditions and entry requirements and provides a mechanism to

relate worker exposure to specific work activities. The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10 (USNRC, 2016).

11.1.2.1.6 Commitment to Radiation Protection Training

The design and implementation of the radiation protection training program complies with the requirements of 10 CFR 19.12. Records are maintained in accordance with 10 CFR 20, Subpart L.

The development and implementation of the radiation protection training program is consistent with the guidance provided in the following regulatory guidance documents:

- Regulatory Guide 8.10 - Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable (USNRC, 2016)
- Regulatory Guide 8.13 - Instructions Concerning Prenatal Radiation Exposure (USNRC, 1999)
- Regulatory Guide 8.29 - Instructions Concerning Risks from Occupational Radiation (USNRC, 1996)
- ASTM E1168-95 - Radiological Protection Training for Nuclear Facility Workers (ASTM, 2013).

Individuals who require unescorted access into restricted areas (as defined in Subsection 11.1.5.1.1) receive training that is commensurate with the radiological hazard to which they may be exposed. Non-facility visitors and fire or emergency responders requiring access to restricted areas are provided with trained escorts who have received radiation protection training.

The level of radiation protection training provided is based on the potential radiological health risks associated with an employee's work responsibilities and incorporates the provisions of 10 CFR 19.12. In accordance with 10 CFR 19.12, any individual working at the facility who is likely to receive in a year a dose in excess of 100 mrem (1 millisievert [mSv]) is:

- Kept informed of the storage, transfer, or use of radioactive material.
- Instructed in the health protection problems associated with exposure to radiation and radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed.
- Provided with access to and training on the use of personal protective equipment (PPE).
- Required to observe, to the extent within the worker's control, the applicable provisions of the NRC regulations and licenses for the protection of personnel from exposure to radiation and radioactive material.
- Instructed of their responsibility to report promptly to the facility management any condition which may cause a violation of NRC regulations and licenses or unnecessary exposure to radiation and radioactive material.
- 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.

Workers who perform or supervise the shipment of radioactive materials are trained and qualified in accordance with 49 CFR 172, Subpart H, in accordance with 10 CFR 71.5.

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 workplace.

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 the SHINE records management system.

Facility 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

Radiation safety audits are conducted, at a minimum, on an annual basis for the purpose of reviewing all functional elements of the radiation protection program to meet the requirement of 10 CFR 20.1101(c). The audit activity is led by a member of the Review and Audit Committee, or other designated independent individual, with the knowledge and experience to perform the activity. The audits provide sufficient information to assess:

- Compliance with NRC regulations
- Compliance with the terms and conditions of the license
- Occupational doses and doses to members of the public for ALARA compliance
- Maintenance of radiation protection program required records

Deficiencies identified during the audit are addressed through the corrective action program. The results of the radiation safety audits are provided to the Radiation Safety Committee, the COO and the CEO for review. Section 12.2 provides additional details of audit activities.

11.1.2.1.8 Record Keeping

Radiation protection records are used for developing trend analysis, for keeping staff and management informed regarding radiation protection matters, and for reporting to regulatory agencies. In addition, the records are used to formulate action based on data obtained (such as survey or sample results), including historical trends.

In accordance with 10 CFR 20, Subpart L, the following records are retained until termination of the facility operating license:

- Records documenting provisions of the radiation protection program [10 CFR 20.2102(b)].

- Results of surveys to determine individual dose from external sources [10 CFR 20.2103(b)(1)].
- Results of measurements and calculations used to determine individual intakes of radioactive material used in the assessment of internal dose [10 CFR 20.2103(b)(2)].
- Results of air sampling, surveys and bioassays required pursuant to 10 CFR 20.1703(c)(1) and (2) for the Respiratory Protection Program [10 CFR 20.2103(b)(3)].
- Results of measurements and calculations used to evaluate release of radioactive effluents to the environment [10 CFR 20.2103(b)(4)].
- NRC Form 4, Cumulative Occupational Dose History [10 CFR 20.2104(f)].
- Planned Special Exposure documentation [10 CFR 20.2105(b)].
- Dose received by all individuals for whom monitoring was required pursuant to 10 CFR 20.1502, and records of doses received during planned special exposures, accidents and emergency conditions. Dose to an embryo/fetus are maintained with record of dose to the declared pregnant woman. [10 CFR 20.2106(a) through (f)].
- Declaration of pregnancy [10 CFR 20.2106(e)].
- Compliance with dose limit for individual members of the public [10 CFR 20.2107(b)].
- Disposal of licensed materials and disposal by burial in soil [10 CFR 20.2108(b)].

In accordance with 10 CFR 20, Subpart L, the following records are retained for three years:

- Records of audits and reviews of the radiation protection program [10 CFR 20.2102(b)].
- Records of surveys and calibrations required by 10 CFR 20.1501, Surveys and Monitoring, and 20.1906(b), Receiving and Opening Packages [10 CFR 20.2103(a)].
- Records used in preparing NRC Form 4 [10 CFR 20.2104(f)].

In accordance with 10 CFR 20.2110, records will be legible throughout the retention period. The record may be an original, or reproduced copy or microform provided it is authenticated by authorized personnel and the microform is capable of producing a clear copy throughout the retention period. Records may be stored in electronic media with the capability for producing legible, accurate and complete records during the required retention period. Records, such as letters, drawings and specifications, include all pertinent information, such as stamps, initials and signatures.

11.1.2.1.9 Technical Specifications

Activities related to the administration and audit of the radiation protection are contained in the facility technical specifications.

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 (USNRC, 2011), 8.13 (USNRC, 1999), and 8.29 (USNRC, 1996). The operation of the facility is consistent with the guidance provided in Regulatory Guide 8.10 (USNRC, 2016).

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 an embryo/fetus of a declared pregnant worker is maintained at or below the limit in 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/areas 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 is responsible for implementing the ALARA program and ensuring that adequate resources are committed to make the program effective. The Radiation Protection Manager prepares an annual ALARA program evaluation report. The report reviews (1) radiological exposure and effluent release data for trends, including ALARA dose goals, (2) results of 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. The effectiveness of the ALARA program is reviewed by the RSC. The RSC sets the ALARA goals for the facility and reviews new activities to ensure ALARA principles are considered. Efforts for improving the effectiveness of equipment used for effluent and exposure control are also evaluated by the RSC. Any resulting recommendations from the committee reviews and evaluations are documented in RSC meeting minutes. The committee's recommendations are dispositioned in the facility’s corrective action process.

11.1.3.1 ALARA Program Considerations

The SHINE facility is designed to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives.

11.1.3.1.1 Design and Construction Policies

ALARA principles were applied during the design of the SHINE facility, consistent with the recommendations in Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be As Low As Is Reasonably Achievable (USNRC, 1978).

Design considerations for maintaining personnel external doses ALARA include the following:

- Materials of construction
- Radioactive material processing, storage, and disposal facilities
- Radiation monitoring systems

- Facility layout for personnel traffic and equipment maintainability and accessibility
- Systems and devices to control access to High Radiation Areas and Very High Radiation Areas
- Utilizing the ALARA concepts of time, distance and shielding. For example:
  - Design work stations to minimize time operators need to be in radiation fields to perform work
  - Locate equipment that require access at a maximum distance from radiation sources or provide remote equipment operation, where practicable
  - Incorporate shielding, where appropriate, to achieve the design condition of ≤ 0.25 mrem/hr at 12 inches [30 cm] from the shielding surface

Design considerations for preventing personnel contamination and minimizing the spread of contamination within the facility include the following:

- Ventilation and filter systems
- Confinement to keep contamination ALARA
- Enclosures to prevent the spread of contamination
- Materials of construction to facilitate decontamination
- Facility layout, with emphasis on personnel and material movement patterns

The following design considerations are used to control radioactive effluent releases:

- Control of airborne effluents by incorporating confinement, radioactive gaseous waste system disposal capabilities, and exhaust system features
- Control of liquid effluents to ensure radioactive materials in excess of the limits are not released
- Use of radioactivity monitoring systems to monitor radioactive effluents

The original facility design concepts to maintain exposures ALARA are presented in Subsection 11.1.3.2.

11.1.3.1.2 Operation Policies

The activities conducted by management personnel who have plant operational responsibility for radiation protection are addressed in Subsection 11.1.2. These activities are consistent with the recommendations of Regulatory Guide 8.10 (USNRC, 2016).

11.1.3.2 ALARA Facility Design Considerations

Facility design considerations for maintaining personnel exposures ALARA are presented in the following paragraphs. The basic management philosophy guiding the SHINE facility design to maintain radiation exposures ALARA includes:

- Designing structures, systems and components such that radioactive material, to the greatest extent practical, is remotely handled and isolated from on-site personnel by shielded compartments and hot cells.
- Designing structures, systems and components for reliability and maintainability, thereby reducing the maintenance requirements on radioactive components.

- Designing structures, systems and components to reduce the radiation fields and control streaming, thereby reducing radiation exposure during operation, maintenance, and inspection activities.
- Designing structures, systems and components to reduce access, repair and removal times, thereby reducing the time spent in radiation fields during operation, maintenance, and inspection.
- Designing structures, systems and components to accommodate remote and semi-remote operation, maintenance and inspection, thereby reducing the time spent in radiation fields.

11.1.3.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods to maintain in-plant radiation exposures ALARA consistent with the recommendations of Regulatory Guide 8.8 (USNRC, 1978) have two objectives:

- Minimizing the necessity for access to and personnel time spent in radiation areas.
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

The following operations are considered during the equipment and facility design to maintain exposures ALARA:

- Normal operation.
- Maintenance and repairs.
- In-service inspection and calibrations.
- Other anticipated operational occurrences.
- Decommissioning.

Examples of features that assist in maintaining exposures ALARA include:

- Design provisions for maintenance of the PCLS and light water pool chemistry conditions, such that corrosion and resulting activation product source terms are minimized.
- Features to allow draining, flushing, and decontamination of equipment and piping.
- Shielding for personnel protection during maintenance or repairs and during decommissioning.
- Means and adequate space for the use of movable shielding.
- Separation of more highly radioactive equipment from less radioactive equipment and separate shielded compartments for adjacent items of radioactive equipment.
- Shielded access openings for installation and removal of plant components.
- Design features, such as the means to provide surface decontamination within hot cells.
- Means and adequate space for the use of remote operations, maintenance, and inspection equipment.
- Separating clean areas from potentially contaminated ones.

11.1.3.2.2 Equipment Design Considerations for ALARA Exposures

Equipment design considerations to minimize the necessity for, and amount of, time spent in a radiation area include:

- Reliability, availability, maintainability, inspectability, constructability, and other design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.
- Design features to facilitate ease of maintenance or repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair or disposal.
- Capabilities to remotely or mechanically operate, repair, service, monitor, or inspect equipment.
- Consideration of redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when there is no feasible method available to reduce radiation levels.
- Capabilities for equipment to be operated from accessible areas both during normal and abnormal operating conditions.

Equipment design considerations directed toward minimizing radiation levels near equipment or components requiring personnel access include:

- Selection of materials that minimize the creation of radioactive contamination.
- Equipment and piping designs that minimize the accumulation of radioactive materials (e.g., the use of buttwelding fittings and minimizing the number of fittings reduces radiation accumulation at the seams and welds).
- Provisions for draining, flushing, or, if necessary, remote cleaning or decontamination of equipment containing radioactive materials.
- Design to limit leaks or control the fluid that does leak. This includes the use of hermetically sealed valves and directing leakage via drip pans and piping.
- Provisions for isolating equipment from radioactive process fluids.

11.1.3.2.3 Facility Layout Design Considerations for ALARA Exposures

Facility layout design considerations to minimize the amount of personnel time spent in a radiation area include the following:

- Locating equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, to promote ease of access and minimize occupancy time in radiation areas.
- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of contaminated equipment.
- Providing, where practicable, for movement of equipment or components requiring service to a lower radiation area.

Design considerations directed toward minimizing radiation levels in occupied areas and in the vicinity of equipment requiring personnel access include the following:

- Separating radiation sources and occupied areas, where practicable.
- Redundant components requiring periodic maintenance that are a source of radiation are located in separate compartments, where practicable, to allow maintenance of one component while the other component is in operation.
- Highly radioactive passive components with minimal maintenance requirements are located in shielded enclosures and are provided with access via shielded openings or removable blocks.

- Providing means and adequate space for using movable shielding when required.
- Designing of the plant layout so that access to a given radiation zone does not require passing through a higher radiation zone.
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.
- Providing control panels to permit remote operation of essential instrumentation and controls from the lowest radiation zone practicable.
- Providing means to control contamination by maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.
- Providing means to facilitate decontamination of potentially contaminated areas.

11.1.4 RADIATION MONITORING AND SURVEYING

11.1.4.1 Radiation Monitoring

An inventory of calibrated radiation detection and measurement instruments is maintained to perform functions such as radiation surveys, contamination surveys, package surveys, sealed source leak tests, air sampling measurements, effluent release measurements, and dose rate measurements. Radiation monitoring equipment, their function and location is shown in Table 11.1-12 and is discussed below.

a. Personnel Monitors

   Personnel who enter radiologically restricted areas (as defined in Subsection 11.1.5.1) are required to wear personnel monitoring devices. In addition, personnel are required to monitor themselves prior to exiting restricted areas which may have the potential for contamination.

b. Continuous Air Monitors

   Continuous air monitors (CAMs) provide indication of the airborne activity levels in the restricted areas of the facility. Alarms are used to provide early warning of unanticipated increases in airborne radioactivity levels. Procedures provide detailed instructions for using and determining CAM alarm setpoints. When deemed necessary, portable air samplers may be used to collect a sample on filter paper for subsequent analysis in the laboratory.

c. Continuous Tritium Detectors

   Tritium is monitored at specific locations where airborne tritium may be present and present a potential hazard to individuals. Tritium monitoring is accomplished using fixed continuous instruments for room air sampling and ventilation duct sampling.

d. Gaseous Effluent Monitoring

   The stack release monitor (SRM) on the facility effluent stack and the carbon delay bed effluent monitor (CDBEM) must be capable of:

   - Continuous monitoring of radioactive stack releases for noble gases.
   - Generating real time data for control room display and recording.

- Allowing periodic collection of filters to allow for laboratory analysis for particulate and iodine.

The SRM provides continuous on-line sampling of releases of gaseous effluents from the facility to demonstrate that releases are within the regulatory limits. The CDBEM is provided to monitor the safety-related alternate release path.

e. Detection and Monitoring of Radioactivity in Liquid Systems and Liquid Effluents

There are no piped radioactive liquid effluent discharges from the facility; therefore, there are no installed liquid effluent monitors. However, liquid effluent releases are collected and sampled prior to release.

Closed loop process cooling water systems are monitored (through sampling or installed instrumentation) to detect leakage between process fluids and cooling water due to failure in a heat exchanger or other system boundary component.

f. Radiation Area Monitors

Radiation area monitors (RAMs) provide radiation monitoring and alarms to alert personnel and the control room of radiation levels that are in excess of normal background levels. RAMs are located in areas to monitor the environment for radioactivity during normal operations, operational occurrences and postulated accidents. Procedures provide detailed instructions for determining and employing alarm set points for RAMs.

RAMs may be provided in High Radiation Areas in order to provide a remote readout. If a RAM is not provided in a particular High Radiation Area, then portable instruments are required by the RWP to measure dose rates when personnel access the area.

g. Control Point Monitoring

Monitor stations are located at the access points for restricted areas. Monitors are provided to detect radioactive contamination of personnel. Monitoring station locations are evaluated and moved as necessary in response to changes in the facility radiological conditions.

Monitoring equipment used at the facility access points are shown in Table 11.1-12.

h. Criticality Monitoring

Criticality monitoring in the RPF is provided by the criticality accident alarm system (CAAS). This system is described in Section 7.7.

Radiation monitoring systems, their functions, and their interfaces with the engineered safety features in the facility are described in Section 7.7.

11.1.4.1.1 Calibration and Maintenance of Radiation Monitoring Equipment

Procedures are prepared for each of the radiation monitoring instruments used and specify the frequency and method of calibration. Radiation monitoring equipment is calibrated before being put into use and after any maintenance or repair that may affect instrument performance.

Calibration of portable radiological monitoring equipment used to document radiological survey results is performed in accordance with ANSI N323AB-2013, American National Standard for Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments (ANSI/ANS, 2014).

Radiation monitoring equipment is calibrated in accordance with manufacturer recommendations.

Maintenance and repair of radiation protection instrumentation is performed in accordance with approved procedures and instrument manufacturer recommendations.

11.1.4.1.2 Operational Tests of Radiation Monitoring Equipment

Operation and response tests of radiation monitoring, counting, and air sampling instruments are performed by personnel trained in the use of the instrument and following approved procedures. These tests are consistent with the manufacturer's recommendations and applicable regulatory requirements. Operation and response tests are conducted at a frequency consistent with industry practices and is addressed in detailed instructions.

11.1.4.2 Radiation Surveys

Radiation surveys are conducted for two purposes: (1) to ascertain radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility; and (2) to detect releases of radioactive material from facility equipment and operations.

To assure compliance with the requirements of 10 CFR 20, Subpart C, there are written procedures for the radiation survey and monitoring programs. The radiation survey and monitoring programs assure compliance with the requirements of 10 CFR 20, Subpart F, Subpart C, Subpart L, and Subpart M.

The radiation survey and monitoring practices are consistent with the guidance provided in the following references:

- Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring (USNRC, 2011)
- Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data (USNRC, 2018)
- Regulatory Guide 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (USNRC, 1993)
- Regulatory Guide 8.24, Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (USNRC, 2012) (applicable to target solution preparation processes)

- Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses (USNRC, 1992)
- ANSI N323AB-2013, American National Standard for Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments (ANSI/ANS, 2014)

Procedures include sampling protocol and data analysis methods. Equipment selection is based on the type of radiation being monitored.

Survey procedures also specify the frequency of measurements and record keeping and reporting requirements. Survey records include:

- Radiation dose rate survey results
- Surface contamination survey results
- Airborne radioactivity survey results

11.1.4.3 Technical Specifications

Certain material in this section provides information that is used in the technical specifications. This includes limiting conditions for operation, setpoints, design features, and means for accomplishing surveillances. In addition, significant material is also applicable to, and may be referenced by, the bases that are described in the technical specifications.

11.1.5 RADIATION EXPOSURE CONTROL AND DOSIMETRY

11.1.5.1 Controlled Access Area

The area of the SHINE site within the security fence, including within the main production facility physical structure beyond the main reception area, but outside any restricted area is part of the controlled access area. Due to the presence of administrative and physical barriers, members of the public do not have direct access to this controlled access area of the site and must be processed by security and authorized to enter the facility. Training for access to a controlled access area is provided commensurate with the radiological hazard.

Facility visitors include delivery people, tour guests, and service personnel who are transient occupants of the controlled area. Area monitoring demonstrates compliance with public dose limits for such visitors. Exposure to SHINE employees or contractors who work only in the controlled access area, but do not enter restricted areas, is limited such that the exposures do not exceed 100 mrem per year.

11.1.5.1.1 Radiological Zones

Radiological zones with varied definitions and span of control have been designated for the facility site and areas surrounding the facility site. The purpose of these zones is to (1) control the spread of contamination, (2) control personnel access to avoid unnecessary exposure of personnel to radiation, and (3) control access to radioactive sources present in the facility. Public access to radiological areas is restricted as detailed in this section and as directed by facility management. Areas where personnel spend substantial amounts of time are designed to minimize the exposure received when routine tasks are performed, in accordance with the ALARA principle.

The following definitions are provided to describe how the radiation protection program is implemented to protect workers and the general public on the site:

a. Unrestricted Area

   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 facility 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 (TEDE) 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

   10 CFR 20.1003 defines a restricted area as an area where access 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 main production facility and locations in the material staging building where radioactive material is normally stored. Radioactive material may be temporarily stored in outdoor areas during transfer between areas. These temporary areas may require that a restricted area be established with the controls described in this section.

c. Radiologically Controlled Area

   The RCA is a restricted area. The RCA is an area within the restricted area posted for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Only individuals who have successfully completed training in radiation protection procedures are permitted to access this area without escort by trained personnel.

Additional radiological areas may exist within the restricted area. The 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. The areas are conspicuously posted in accordance with the requirements of 10 CFR 20 as shown on Table 11.1-13.

Radiation areas and expected dose rates are shown in Table 11.1-4.

11.1.5.2 Access and Egress Control

SHINE establishes and implements an access control program that ensures that (a) signs, labels, and other access controls are properly posted and operative, (b) restricted areas are established to prevent the spread of contamination and are identified with appropriate signs, and (c) step-off pads, change facilities, protective clothing facilities, and personnel monitoring instruments are provided in sufficient quantities and locations, as necessary.

Personnel access to high radiation areas is controlled to prevent unplanned radiation exposures. Personnel access is controlled through administrative methods, including procedures and RWPs. Active and passive safety features are provided to control access to high radiation areas in accordance with 10 CFR 20.1601. These safety features include:

- Neutron driver service cell personnel access door interlocks de-energize the accelerator to reduce the level of radiation upon personnel entry (defense-in-depth design attribute), and accelerator key switches prevent activation of the accelerator while personnel are present.
- Hot cells requiring periodic/routine entry where there is potential for excessive personnel exposures are equipped with door interlocks to prevent the hot cell door from being opened when the evaluated hazard exists (e.g., excessive radiation field, target solution transfer occurring in cell).
- The neutron driver service cell and hot cells are equipped with audible and visual warnings so that an individual attempting to enter the High Radiation Area and the supervisor of the activity are made aware of the entry or are controlled by locked entry with positive access controls over each individual entry, consistent with 10 CFR 20.1601(a).
- High radiation 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, consistent with 10 CFR 20.1601(a)(3).

Access to and egress from the restricted area is through one of the monitor stations at the restricted area boundary. Access to and egress from each Radiation Area, High Radiation Area, Contaminated Area or Airborne Radioactivity Area within the restricted area may also be individually controlled. A monitor (frisker), step-off pad, and container for any discarded protective clothing may be provided at the egress point from certain of these areas to prevent the spread of contamination.

11.1.5.3 Posting for Radiation Protection Awareness

Radiological postings are clearly identified by physical means such as placarding or boundary marking in accordance with 10 CFR 20.1902.

11.1.5.4 Protective Clothing and Equipment

Personnel working in areas that are classified as airborne radioactivity areas or contaminated areas must wear appropriate PPE. If the areas containing the surface contamination can be isolated from adjacent work areas via a barrier such that dispersible material is not likely to be transferred beyond the area of contamination, personnel working in the adjacent area are not required to wear PPE. Areas requiring PPE are posted at each of their entry points. The radiation worker training program provides instruction to personnel on the proper use of PPE.

Radiation protection management and associated technical staff are responsible for determining the need for PPE in each work area and documenting the PPE requirements on the applicable RWP. For areas with removable contamination from beta/gamma emitters or uranium above 1,000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2) or from alpha emitters other than uranium above 20 dpm/cm^2 PPE is required. PPE includes coveralls, gloves,

shoe covers, and rubber boots. Guidance for selecting and using PPE is provided in the facility radiation protection program.

The respiratory protection program is described in Section 11.3.

11.1.5.5 Personnel Monitoring for External Exposures

External exposures are received primarily from the fission products produced in the target solution. Other potential sources of exposure include neutrons (e.g., from operational neutron drivers), activation products, and tritium gas. The nuclides of radiological significance are identified above in Section 11.1.

Personnel whose duties require them to enter restricted areas wear individual external dosimetry devices that are sensitive to beta, gamma and neutron radiation. Personnel handling licensed sources and working around radioactive materials outside restricted areas (e.g., transportation-related surveys) wear individual external dosimetry devices that are sensitive to beta, gamma and neutron radiation. Any individual entering a High Radiation Area or Very High Radiation Area wears personal dosimetry, and supplemental dosimetry with dose and dose rate alarm capability.

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 11.1-11. The administrative limits are reflective of ALARA principles.

Investigation levels are set at 25 percent of the annual administrative limit for any worker's occupational dose received during a calendar quarter. An investigation is performed and documented to determine what types of activities may have contributed to the worker's external exposure. The investigation may include, but is not limited to, procedural reviews, efficiency studies of the ventilation system, uranium storage protocol, and work practices.

Any time an administrative limit is exceeded, the Radiation Protection Manager is informed. The Radiation Protection Manager is responsible for determining the need for and recommending investigations or corrective actions to the responsible manager(s). Copies of the Radiation Protection Manager's recommendations are provided to the RSC.

Exposure limits for volunteer emergency responders are controlled and administered by the facility Emergency Plan.

11.1.5.6 Determination of Internal Exposures

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.

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.

The radiation protection program includes detailed methodology for determination of internal exposures.

11.1.5.7 Evaluation and Record of Doses

Individual worker occupational dose is assessed on a quarterly basis and is performed more frequently when reasonable suspicion exists regarding an abnormal exposure. External dosimetry devices are processed and evaluated by a provider accredited by the National Voluntary Laboratory Accreditation Program.

- Procedures for the evaluation and summation of doses are based on guidance contained in Regulatory Guides 8.7 (USNRC, 2018) and 8.34 (USNRC, 1992).

Records are maintained of doses received by all individuals for whom monitoring is required under 10 CFR 20.1502, in accordance with 10 CFR 20.2106. The records include the following, as applicable:

- The deep-dose equivalent to the whole body, lens dose equivalent, shallow-dose equivalent to the skin and shallow-dose equivalent to the extremities;
- The estimated intake of radionuclides;
- The committed effective dose equivalent (CEDE) assigned to the intake of radionuclides;
- The specific information used to calculate the CEDE under 10 CFR 20.1204(a) and (c), when required by 10 CFR 20.1502;
- The TEDE, when required by 10 CFR 20.1202; and
- The total of the deep-dose equivalent and the committed dose to the organ receiving the highest total dose.

See also Subsection 11.1.2.1.8 for retained individual dose evaluation records.

11.1.5.8 Planned Special Exposures

SHINE may authorize an adult worker to receive (non-emergency) doses, in addition to and accounted for separately, from the doses received under the limits specified in 10 CFR 20.1206(e), provided that each of the requirements of 10 CFR 20.1206(a), (b), (c), and (d) are met.

SHINE maintains records of the conduct of a planned special exposure and submit a written report as required by 10 CFR 20.1206(f). In accordance with 10 CFR 20.2105, the record of a planned special exposure includes:

- The exceptional circumstances requiring the use of a planned special exposure.
- The name of the management official who authorized the planned special exposure and a copy of the signed authorization.
- What actions were necessary.
- Why the actions were necessary.
- How doses were maintained ALARA.

- What individual and collective doses were expected to result, and the doses actually received in the planned special exposure.

11.1.6 CONTAMINATION CONTROL EQUIPMENT AND FACILITY LAYOUT GENERAL DESIGN CONSIDERATIONS FOR 10 CFR 20.1406

Contamination control is part of the radiation protection program described in Subsection 11.1.2. Personnel receiving Radiation Worker Training are instructed on the sources, detection and control of radioactive contamination. Procedures provide instruction for identifying and controlling contamination. Records of contamination events are entered into the corrective action process, reviewed by the RSC, and maintained as records, as applicable, in accordance with the radiation protection program requirements described in Subsection 11.1.2.

General equipment and facility layout design considerations to prevent the spread of contamination in the facility and to the environment and to facilitate eventual decommissioning in accordance with 10 CFR 20.1406 include the features discussed in the following subsections.

11.1.6.1 Shielded Compartments and Hot Cells

Process equipment containing significant radioactive material is located within shielded compartments or hot cells.

Process equipment which does not require local operator interaction during production, such as the neutron driver assembly and the subcritical assembly, is located in shielded compartments (access is provided via shielded openings as required). Where operator intervention is required during processing activities, for example molybdenum extraction and purification, the equipment is located in shielded hot cells and the operator is provided with a means for remote viewing and manipulation of components.

These shielded compartments and shielded hot cells are provided to facilitate confinement, isolation, and collection of potential liquid spills to minimize the spread of contamination to the facility and the environment. With the exception of the below grade confinement, these shielded compartments and shielded hot cells are provided with ventilation systems which are operated at negative pressures with respect to the surrounding environment (see Section 9a2.1).

11.1.6.2 Piping

Where shielding is required, radioactive piping is located inside shielded compartments or hot cells. For transfers between hot cells the piping is located in shielded pipe trenches which provide for liquid and airborne confinement and detection of leakage. Inspection ports are provided to allow for visual inspection of piping. Use of embedded piping is minimized to facilitate inspection and detect leakage.

11.1.6.3 Light Water Pool

The light water pool which provides shielding and cooling for the subcritical assembly system (SCAS) is designed with leak detection to prevent unidentified leakage to the facility and the environment.

11.1.6.4 Process Tanks

Process tanks are seismically supported and are located in seismically designed concrete vaults that are designed to prevent unidentified leakage to the facility and the environment.

11.1.6.5 Monitoring and Controlled Entry and Egress to Restricted Area

Access to and egress from these areas is strictly controlled via administrative procedures and passive confinement structure design.

Personnel access and egress is controlled by Radiation Protection personnel, equipment and procedures. Prior to entry, personnel must don appropriate PPE to minimize the potential for physical contamination of the worker and the subsequent spread of contamination beyond the restricted area. This PPE is either removed and disposed of or monitored for contamination prior to release from the restricted area. Personnel must then pass through appropriate portal monitoring equipment prior to egress from the restricted area.

Potentially contaminated materials removed from the restricted area (for example, production material, tools, disposed equipment, various process and maintenance consumables) are surveyed and released, when appropriate, following radiation protection program implementing procedures. Disposal of contaminated materials is performed in accordance with radioactive waste management program implementing procedures (see Section 11.2).

Restricted areas in the main production facility are provided with fixed CAMs to detect the potential spread of airborne contamination within the restricted area. Additionally, RAMs are in place to detect potential increases in background radiation levels.

Radiation protection personnel routinely perform radiation and contamination assessments of accessible areas within restricted areas. Special surveys are performed, prior to entry, if access is required to normally unoccupied areas.

11.1.7 ENVIRONMENTAL MONITORING

11.1.7.1 Environmental Monitoring Program

SHINE maintains a radiological environmental monitoring program (REMP) as required by 10 CFR 20.1302. The REMP is used to verify the effectiveness of facility measures which are used to control the release of radioactive material and to verify that measurable concentrations of radioactive materials and levels of radiation are not higher than expected based on effluent measurements and modeling of the environmental exposure pathways.

Guidance provided in Regulatory Guide 4.1, Radiological Environmental Monitoring for Nuclear Power Plants (USNRC, 2009) and Table 3.12-1 of NUREG-1301, Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors (USNRC, 1991), was considered when developing the REMP for the SHINE facility. In addition, the REMP was developed using the data quality objectives (DQO) process which is a scientific systematic planning method. The DQOs were developed according to the U.S. Environmental Protection Agency (EPA) Guidance on Systematic Planning Using the Data Quality Objectives Process (EPA, 2006).

Environmental monitoring is conducted at potential receptor locations. Details of the REMP are presented in the following sections.

11.1.7.2 Effluent Release Pathways

Airborne effluents from the facility include noble gases, iodine and other halogens, particulates, and tritium. The following pathways represent plausible public exposure scenarios from airborne effluents:

- Direct radiation exposure pathway monitored using dosimeters.
- Inhalation pathway monitored using continuous air samples.
- Ingestion exposure pathway.

There are no routine radioactive liquid effluent discharges from the RCA. Radioactive liquid discharges from the SHINE facility to the sanitary sewer are infrequent and made in accordance with 10 CFR 20.2003 and 10 CFR 20.2007. There are no piped liquid effluent pathways from the RCA to the sanitary sewer. Sampling is used to determine suitability for release. See Section 11.2 for additional information on liquid discharges from the RCA.

11.1.7.2.1 Direct Radiation Monitoring

Direct exposure to gamma and beta emitting radionuclides released through the stack of the SHINE production facility is monitored and measured at receptor locations using environmental dosimeters. The dosimeters measure direct radiation from radiation sources contained within the SHINE main production facility, from sources within the material staging building, from radioactivity in the airborne effluent, and from deposition of airborne radioactivity onto the ground.

A description of dosimeter locations and the rationale for locations are provided in Table 11.1-14. Dosimeter locations are shown on Figure 11.1-4. Table 3.12-1 of NUREG-1301 (USNRC, 1991) recommends 40 dosimeter locations (i.e., an inner ring and an outer ring of dosimeters with one dosimeter in each ring at each of the 16 meteorological sectors and the balance of dosimeters to be located at special interest areas). At least one dosimeter is to serve as a control, i.e., located a significant distance from the facility such that it represents a background dose. Considering the size of the SHINE facility and the low power level of the SHINE subcritical IUs, 24 dosimeter locations are specified. These dosimeters are located in order to provide annual direct dose information at on-site locations which are expected to have occupancy and at property line locations which ensure all directions are monitored. The property line locations include the direction of the theoretical MEI and the direction of the nearest occupied structure. At least one location includes a paired dosimeter so that data quality can be determined. Three of the dosimeters are stationed off site at special interest areas and one dosimeter is located a significant distance from the facility to represent background dose.

Dosimeter values are calculated using the reports from the laboratory providing results.

Background radiation is subtracted from the dosimeter results. The background radiation values are those established during the baseline environmental survey which obtained baseline dosimeter readings at each dosimeter location.

11.1.7.2.2 Iodine and Particulate Monitoring for Releases via Airborne Pathway

Airborne effluent releases from the SHINE facility contribute to off-site doses. Air monitoring detects iodine or particulate releases from the SHINE facility. Noble gas and tritium measurements are not included in the REMP. Noble gas and tritium measurements are performed by the radiation protection program.

Environmental airborne sampling is performed to identify and quantify particulates and radioiodine in airborne effluents. Regulatory Position C.3.b of Regulatory Guide 4.1 (USNRC, 2009) indicates that airborne sampling should always be included in the environmental monitoring programs for nuclear power plants since the airborne effluent pathway exists at all sites. Since the SHINE facility includes airborne effluent releases and radioactivity in the airborne effluent can result in measurable off-site doses and since there is a potential for a portion of the dose to be attributable to radioactive iodine and airborne particulate radioactivity releases, the REMP includes airborne sampling.

11.1.7.2.2.1 Air Sampling Locations

The DQO process and the guidance provided in Table 3.12-1 of NUREG-1301 (USNRC, 1991) were used to establish locations for airborne sample acquisition, sampling frequency, and type of sample analysis. Continuous air sample locations are specified in accordance with guidance provided in Table 3.12-1 of NUREG-1301 (USNRC, 1991). The continuous air sampling is performed using continuous air samplers (CAS) which include a radioiodine canister for iodine-131 (I-131) analysis and a particulate sampler which is analyzed for gross beta radioactivity.

Four CAS locations (CAS 2 – CAS 5) are near the facility property line in the north, south, east and west direction sectors co-located with ED1, ED9, ED5, and ED13 (refer to Figure 11.1-4), respectively, to ensure all directions are monitored. The north and east direction sectors (with respect to the SHINE facility vent stack) have the highest calculated annual ground level deposition factor (D/Q) values (CAS 2 and CAS 4). There is also a control CAS (CAS 1) located a sufficient distance from the SHINE medical isotope production facility to provide background information for airborne activity. Table 3.12-1 of NUREG-1301 (USNRC, 1991) suggests an additional air sample location in the vicinity of a community having the highest calculated annual average ground level deposition factor, D/Q. This CAS requirement is combined with the air sample location at the site boundary location in the north direction (refer to Table 11.1-14). A description of air sample locations and the rationale for air sample locations are provided in Table 11.1-14.

The air sampling data is used to validate the effluent monitoring and dose compliance data sets. Results are compared to the radionuclide-specific values provided in 10 CFR 20, Appendix B. A sum-of-the fractions approach is used wherein the isotopic values measured are compared with their associated limits in 10 CFR 20, Appendix B. This allows the calculation of dose due to iodine and particulate activities and includes both inhalation dose and cloud immersion dose. Background subtraction is based on results of the baseline environmental survey, thus providing a location-specific and statistically valid means to subtract background.

11.1.7.2.3 Ingestion Pathway (Biota Monitoring)

NUREG-1301 (USNRC, 1991) suggests sampling of various biological media as a means to indirectly assess doses due to particulate and iodine ingestion. This type of monitoring may include sampling of soils, broad leafed plants, fish, meat, or milk. Nuclear power plants have long monitored this pathway and have seen neither appreciable dose nor upward trending of deposition. Since the SHINE source term is expected to be several orders of magnitude lower than that of a nuclear power plant and particulate and iodine radionuclides are not normally expected to be present in measurable quantities within airborne effluent releases from the SHINE facility, biota monitoring is not routinely included in the REMP.

11.1.7.2.4 Groundwater Monitoring

Surface waters of the rivers in the vicinity of the plant (e.g., the Rock River and its tributaries) are not expected to accumulate detectable levels of radioactivity. As such, surface water sampling is not included in the REMP. Similarly, marine life in the rivers is not expected to accumulate detectable levels of radioactivity and thus sampling of fish or other marine creatures for the ingestion pathway is not included in the REMP.

Measured local water table elevations for the site identify the groundwater gradient and indicate that the groundwater flow is to the west and to the south. The nearest drinking water source is a well located approximately a third of a mile (0.54 km) to the northwest of the facility.

There are four test wells within the property boundary for the SHINE facility that were used for monitoring groundwater in support of a hydrological assessment of the site. One test well is located north, one south, one east, and one west of the SHINE main production facility. Although there are no defined liquid effluent release pathways and the groundwater is not expected to be contaminated due to operation of the SHINE facility, the test wells to the west and the south are sampled for the presence of radionuclide contaminants. Sampling is in accordance with the recommendations in Table 3.12-1 of NUREG-1301 (USNRC, 1991) (i.e., quarterly with gamma isotopic and tritium analysis). The rationale for sampling the test wells to the west and south of the SHINE facility is provided in Table 11.1-14.

11.1.7.3 Community Environmental Monitoring Program

In addition to the monitoring that is performed by the REMP to meet regulatory requirements, SHINE has a Community Environmental Monitoring Program (CEMP). The CEMP includes voluntary environmental monitoring based on public or SHINE interests that are not regulatory in nature.

11.1.7.4 Preoperational Baseline Monitoring

Preoperational monitoring, beginning approximately two years prior to anticipated licensed activity, serves to provide baseline data for evaluating the impact of operation of the SHINE facility. The collection of samples and analysis of data follow the sampling and analyses schedule specified in Subsection 11.1.7.5 and continue into the operational phase of facility operation. The preoperational monitoring is conducted so that the preoperational radiological conditions are understood in sufficient detail to allow future reasonable, direct comparison with data collected after licensed operation of the facility.

11.1.7.5 Sampling and Analysis

The following frequencies are used; however, alterations may be made based upon data and trends, and the justification of any such alterations are described in the Annual Report. If sample or analysis frequencies are reduced, the changes are not to reduce the overall effectiveness of the REMP.

- Air sample filters – monthly, or more frequently if required by dust loading on media
- Environmental dosimeters – quarterly
- Groundwater test wells – quarterly

Sample analysis employs analytical techniques so that an appropriate analytical sensitivity (e.g., a priori Lower Level of Detection [LLD]) is achieved. SHINE may also use the analytical detection sensitivities as determined based on the Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP). Deviations from the a priori analytical sensitivity levels due to interference from other radionuclides or other factors are evaluated and documented. SHINE reports analytical sensitivity capabilities of the REMP in the Annual Report.

In accordance with Regulatory Guide 4.1 (USNRC, 2009), Revision 2, analyses for carbon-14 in environmental media are not required since the facility produced component is a small fraction of the naturally occurring carbon-14.

11.1.7.6 Environmental Monitoring Program Procedures

Environmental surveys conducted in support of the REMP are performed in accordance with facility implementing procedures. Document control measures are employed to ensure that changes to the REMP or implementing procedures are reviewed for adequacy, approved by authorized personnel and are distributed to and used at the appropriate locations throughout the facility.

11.1.7.7 REMP Reports

An Annual Report is provided to the NRC in accordance with ANSI/ANS 15.1-2007 (ANSI/ANS, 2007). The Annual Report provides summarized results of environmental surveys performed outside the facility.

11.1.7.8 Records, Periodic Review and Corrective Actions

Records of off-site environmental surveys are retained in accordance with the SHINE records management program for the lifetime of the facility.

An annual environmental monitoring program review is conducted to examine the adequacy and effectiveness of the REMP to achieve its objectives. The program review evaluates the need to expand (or reduce) the environmental monitoring program given the results of the environmental data and trends in environmental radioactivity. Any reductions shall be thoroughly evaluated and justified, given that environmental data indicating the absence of facility-related radioactivity are important. The review confirms exposure pathways and sampling media and validates that the principal radionuclides being discharged are the same nuclides being analyzed in the environmental program.

Any adverse trends or anomalies identified during the conduct of the program, during Annual Report preparation, or during periodic reviews, are entered into the facility corrective action program for disposition.

Table 11.1-1 – Parameters Applicable to Target Solution Radionuclide Inventories

Parameter	Nominal Values	Safety Basis Values
Power	125 kW	137.5 kW
Irradiation Time	5.5 days	30 days
Total Time Between Irradiations	[ ]PROP	[ ]PROP/ECI
Number of Cycles	[ ]PROP/ECI	[ ]PROP/ECI
Element Partitioning (Extraction) Between Cycles	Nominal	None
Element Partitioning (Extraction) on Final Cycle	Nominal	Bounding (noble gases only)
TSV Dump Tank Decay Time	[ ]PROP/ECI	[ ]PROP/ECI
Supercell Extraction Time	[ ]PROP/ECI	[ ]PROP/ECI

Table 11.1-2 – Nominal Versus Safety Basis Radionuclide Inventories in Target Solution

	Actinide Activity (Ci)			Fission Product Activity (Ci)		
Case	**At Shutdown**	**Pre-Extraction**	**Post-Extraction**	**At Shutdown**	**Pre-Extraction**	**Post-Extraction**
Nominal Values[a]	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
Safety Basis Values[b]	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
Difference	13 percent	20 percent	28 percent	20 percent	70 percent	170 percent

a. Pre-Extraction: [ ]PROP/ECI post-shutdown; Post-Extraction: [ ]PROP/ECI post-shutdown
b. Pre-Extraction: [ ]PROP/ECI post-shutdown; Post-Extraction: [ ]PROP/ECI post-shutdown

Table 11.1-3 – Irradiated Target Solution Activity for Select Radionuclides Pre-Extraction (Sheet 1 of 3)

Radionuclide	Nominal Activity (Curies)	Safety Basis Activity (Curies)
Kr-85	[ ]PROP/ECI	[ ]PROP/ECI
Kr-85m	[ ]PROP/ECI	[ ]PROP/ECI
Kr-87	[ ]PROP/ECI	[ ]PROP/ECI
Kr-88	[ ]PROP/ECI	[ ]PROP/ECI
Rb-86	[ ]PROP/ECI	[ ]PROP/ECI
Sr-89	[ ]PROP/ECI	[ ]PROP/ECI
Sr-90	[ ]PROP/ECI	[ ]PROP/ECI
Sr-91	[ ]PROP/ECI	[ ]PROP/ECI
Sr-92	[ ]PROP/ECI	[ ]PROP/ECI
Y-90	[ ]PROP/ECI	[ ]PROP/ECI
Y-91	[ ]PROP/ECI	[ ]PROP/ECI
Y-92	[ ]PROP/ECI	[ ]PROP/ECI
Y-93	[ ]PROP/ECI	[ ]PROP/ECI
Zr-95	[ ]PROP/ECI	[ ]PROP/ECI
Zr-97	[ ]PROP/ECI	[ ]PROP/ECI
Nb-95	[ ]PROP/ECI	[ ]PROP/ECI
Mo-99	[ ]PROP/ECI	[ ]PROP/ECI
Tc-99m	[ ]PROP/ECI	[ ]PROP/ECI
Ru-103	[ ]PROP/ECI	[ ]PROP/ECI
Ru-105	[ ]PROP/ECI	[ ]PROP/ECI
Ru-106	[ ]PROP/ECI	[ ]PROP/ECI
Rh-105	[ ]PROP/ECI	[ ]PROP/ECI
Sb-127	[ ]PROP/ECI	[ ]PROP/ECI
Sb-129	[ ]PROP/ECI	[ ]PROP/ECI
Te-127	[ ]PROP/ECI	[ ]PROP/ECI
Te-127m	[ ]PROP/ECI	[ ]PROP/ECI
Te-129	[ ]PROP/ECI	[ ]PROP/ECI
Te-129m	[ ]PROP/ECI	[ ]PROP/ECI

Table 11.1-3 – Irradiated Target Solution Activity for Select Radionuclides Pre-Extraction (Sheet 2 of 3)

Radionuclide	Nominal Activity (Curies)	Safety Basis Activity (Curies)
Te-131m	[ ]PROP/ECI	[ ]PROP/ECI
Te-132	[ ]PROP/ECI	[ ]PROP/ECI
I-131	[ ]PROP/ECI	[ ]PROP/ECI
I-132	[ ]PROP/ECI	[ ]PROP/ECI
I-133	[ ]PROP/ECI	[ ]PROP/ECI
I-134	[ ]PROP/ECI	[ ]PROP/ECI
I-135	[ ]PROP/ECI	[ ]PROP/ECI
Xe-131m	[ ]PROP/ECI	[ ]PROP/ECI
Xe-133	[ ]PROP/ECI	[ ]PROP/ECI
Xe-133m	[ ]PROP/ECI	[ ]PROP/ECI
Xe-135	[ ]PROP/ECI	[ ]PROP/ECI
Xe-135m	[ ]PROP/ECI	[ ]PROP/ECI
Xe-138	[ ]PROP/ECI	[ ]PROP/ECI
Cs-134	[ ]PROP/ECI	[ ]PROP/ECI
Cs-136	[ ]PROP/ECI	[ ]PROP/ECI
Cs-137	[ ]PROP/ECI	[ ]PROP/ECI
Ba-139	[ ]PROP/ECI	[ ]PROP/ECI
Ba-140	[ ]PROP/ECI	[ ]PROP/ECI
La-140	[ ]PROP/ECI	[ ]PROP/ECI
La-141	[ ]PROP/ECI	[ ]PROP/ECI
La-142	[ ]PROP/ECI	[ ]PROP/ECI
Ce-141	[ ]PROP/ECI	[ ]PROP/ECI
Ce-143	[ ]PROP/ECI	[ ]PROP/ECI
Ce-144	[ ]PROP/ECI	[ ]PROP/ECI
Pr-143	[ ]PROP/ECI	[ ]PROP/ECI
Nd-147	[ ]PROP/ECI	[ ]PROP/ECI
Np-239	[ ]PROP/ECI	[ ]PROP/ECI
Pu-238	[ ]PROP/ECI	[ ]PROP/ECI
Pu-239	[ ]PROP/ECI	[ ]PROP/ECI
Pu-240	[ ]PROP/ECI	[ ]PROP/ECI
Pu-241	[ ]PROP/ECI	[ ]PROP/ECI

Table 11.1-3 – Irradiated Target Solution Activity for Select Radionuclides Pre-Extraction (Sheet 3 of 3)

Radionuclide	Nominal Activity (Curies)	Safety Basis Activity (Curies)
Am-241	[ ]PROP/ECI	[ ]PROP/ECI
Cm-242	[ ]PROP/ECI	[ ]PROP/ECI
Cm-244	[ ]PROP/ECI	[ ]PROP/ECI
Rb-88	[ ]PROP/ECI	[ ]PROP/ECI
Y-91m	[ ]PROP/ECI	[ ]PROP/ECI
Nb-97m	[ ]PROP/ECI	[ ]PROP/ECI
Nb-97	[ ]PROP/ECI	[ ]PROP/ECI
Rh-103m	[ ]PROP/ECI	[ ]PROP/ECI
Rh-105m	[ ]PROP/ECI	[ ]PROP/ECI
Rh-106	[ ]PROP/ECI	[ ]PROP/ECI
Ba-136m	[ ]PROP/ECI	[ ]PROP/ECI
Ba-137m	[ ]PROP/ECI	[ ]PROP/ECI
Pr-144	[ ]PROP/ECI	[ ]PROP/ECI
Pr-144m	[ ]PROP/ECI	[ ]PROP/ECI

Table 11.1-4 – Radiation Areas at the SHINE Facility

Area	Dose Rate	Designation
Normally occupied areas within the RCA TPS room NDAS service cell without accelerator operation	≤ 5 mrem/hr	Normally occupied area
IU cells, hot cells, and other shielded vaults; cells; and rooms – material not present or accelerator not in operation, after sufficient decay period Above RPF trench during solution transfers Primary cooling rooms during operation IF general area during accelerator operation in NDAS service cell	> 5 mrem/hr but ≤ 100 mrem/hr	Radiation Area (transient occupation)
IU cells, hot cells, and other shielded vaults; cells; and rooms – material present or accelerator in operation or shutdown without sufficient decay period NDAS service cell with accelerator operation	> 100 mrem/hr (High Radiation Area) or > 500 rad/hr (Very High Radiation Area)	High Radiation Area or Very High Radiation Area (rarely occupied, per ALARA controls)

Table 11.1-5 – Airborne Radioactive Sources

System	Component	Location	Major Sources	Estimated Maximum Activity (Ci)	Exterior Dose Rate (mrem/hr)[a]
TPS	Tritium purification system	TPS gloveboxes	H-3	300,000[b]	< 0.25
NDAS	Driver vacuum hardware	IU cell	H-3	[ ]PROP/ECI(c)	< 0.25
TOGS	Off-gas piping, zeolite beds	TOGS shielded cell	I, Kr, Xe	120,000[d]	< 0.25
RVZ1	IU cell atmosphere and PCLS	IU cell	Ar-41 and N-16	Ar-41: 1E-05 N-16: 10[d]	N/A
RVZ1	Supercell atmosphere	Supercell gloveboxes	I, Kr, Xe, and particulates	3	< 0.2
PVVS and VTS	PVVS and VTS piping	Pipe trenches, valve pits, and PVVS hot cell	I, Kr, Xe	25,000[d]	< 1

a. Dose contribution from listed source in normally occupied area, includes direct dose at 30 cm from the exterior of the shielding surface and contributions from the derived air concentration.
b. Includes inventory in NDAS units.
c. H-3 activity is per NDAS unit.
d. Value is per irradiation unit (IU).

Table 11.1-6 – Estimated Derived Air Concentrations

Source Description	Location	Particulate	Halogen	Noble Gas	Tritium	Total
Primary System Boundary	IF General Area	-	0.4%	0.1%	-	0.4%
Tritium Systems	TPS Room	-	-	-	1.4%	1.4%
	IF General Area, Normal Operation	-	-	-	3.2%	3.2%
	IF General Area, Maintenance	-	-	-	5.2%	5.2%
Below-Grade Vaults	RPF General Area	-	0.1%	0.0%	-	0.1%
PVVS Hot Cell	PVVS Hot Cell	-	12%	1.9%	-	14%
	RPF General Area	-	0.0%	0.0%	-	0.0%
Extraction Hot Cell	Extraction Hot Cell	13%	> 10 DAC	76%	0.0%	> 10 DAC
	RPF General Area	0.0%	2.1%	0.0%	0.0%	2.1%
Purification Hot Cell	Purification Hot Cell	38%	> 10 DAC	220%	0.0%	> 10 DAC
	RPF General Area	0.0%	4.2%	0.0%	0.0%	4.2%
IF General Area Total		-	0.4%	0.1%	8.3%	8.8%
RPF General Area Total		0.0%	6.4%	0.0%	-	6.4%

Table 11.1-7 – Key Parameters for Normal Yearly Release Calculation

Parameter	PVVS Pathway	Hot Cells	Primary Confinement Boundary	General Area
Primary Nuclide Inventory Constituents	Kr, Xe, I	Kr, Xe, I, particulates	Kr, Xe, Ar 41, N-16	Kr, Xe, I, H-3
Type of Radiation Emitted	Beta and Gamma	Beta and Gamma	Beta and Gamma	Beta and Gamma
Total Curies	9.0E+05	320	430	32
Primary Constituents Released	Kr, Xe	Kr, Xe	Kr, Xe	Kr, Xe, H-3
Type of Radiation Emitted	Beta and Gamma	Beta and Gamma	Beta and Gamma	Beta and Gamma
Total Curies	9.3E+03	16	9.5	32
Delay Time Credited for Decay	1.7 days (Kr) 40 days (Xe)	None	1 minute	None
Iodine Removal Mechanisms	Carbon Guard Bed Carbon Delay Beds	Carbon Filter on Hot Cell RVZ1 Exhaust Carbon Filter on Facility RVZ1 Exhaust	Carbon Filter on Facility RVZ1 Exhaust	Carbon Filter on Facility RVZ1 Exhaust

Table 11.1-8 – Estimated Annual Releases from Normal and Maintenance Operations (Nuclides with Greater than 1 Ci Annual Release)

Radionuclide	Annual Release (Ci)
Kr-83m	5.9E+00
Kr-85	1.2E+02
Kr-85m	5.0E+01
Kr-88	2.2E+00
Xe-131m	1.3E+03
Xe-133	7.8E+03
Xe-133m	1.1E+00
Xe-135	6.2E+00
Xe-135m	1.0E+01
H-3	7.3E+01

Table 11.1-9 – Liquid Radioactive Sources (Sheet 1 of 2)

System[a]	Component[a]	Location[a]	Major Sources	Estimated Maximum Activity (Ci)	Exterior Dose Rate (mrem/hr)[d]
TSPS	Target solution, unirradiated	Target Solution Preparation Area	U-234, U-235, U-238	3	N/A
SCAS	Target solution in TSV (operating)	IU cell	U-235 Fission (Neutrons and Photons)	[ ]PROP/ECI(b)	< 0.25
SCAS	Target solution in TSV, TSV dump tank (shutdown)	IU cell	(see Table 11.1-3)	[ ]PROP/ECI(b)	< 0.03
LWPS	Water in the light water pool	IU cell	H-3	30(b)	N/A
NDAS	Oil in NDAS pumps	IU cell	H-3	2000(b)	N/A
PCLS	Primary cooling water in pump and piping	IU cell and primary cooling room	N-16	7.5(b)	< 2
MEPS	Target solution in pump, extraction column, and lift tanks	Supercell	(see Table 11.1-3)	[ ]PROP/ECI(c)	< 5
MEPS	Mo eluate in Mo eluate hold tank	Supercell	Mo, [ ]PROP/ECI	[ ]PROP/ECI(c)	< 3
MEPS	Mo-99 product	Supercell	Mo-99, Tc-99	[ ]PROP/ECI(c)	< 0.2
TSSS	Target solution in target solution hold tank	Tank vault	(see Table 11.1-3)	[ ]PROP/ECI(b)	< 0.25

Table 11.1-9 – Liquid Radioactive Sources (Sheet 2 of 2)

System[a]	Component[a]	Location[a]	Major Sources	Estimated Maximum Activity (Ci)	Exterior Dose Rate (mrem/hr)[d]
RLWS	Liquid waste in annular waste tank	Tank vault	[ ]PROP/ECI and other fission products	3.8E+04	< 0.1
RLWS	Liquid waste in RLWS collection tank	Tank vault	[ ]PROP/ECI and other fission products	5.7E+04	< 0.1

a. Physical and chemical properties of process solutions, special nuclear material inventories, and descriptions of the systems can be found in Chapter 4.
b. Value is per irradiation unit (IU).
c. Value is per cycle.
d. For normally-occupied areas.

Table 11.1-10 – Solid Radioactive Sources

System[a]	Component[a]	Location	Major Sources	Estimated Maximum Activity (Ci)	Exterior Dose Rate (mrem/hr)
NDAS	Neutron Driver	IU Cell	Activation Products	300[b]	N/A
TOGS	TOGS Components	IU Cell and TOGS Cell	Rb, Cs, Ba, Sr, Y, La, and Ce	5.6E+04[b]	< 0.25
SCAS	Neutron Multiplier, SASS	IU Cell	Activation and Fission Products	1.5E+05[b]	N/A
MEPS	Spent Extraction [ ]PROP/ECI	Supercell	[ ]PROP/ECI	2.6E+04[c]	< 5
MEPS	Glassware	Supercell and Solid Waste Drum Storage	[ ]PROP/ECI	100[c]	N/A
TSPS and URSS	Fresh Uranium Metal and Uranium Oxide	Target Solution Preparation and Storage Areas	U-234, U-235, U-238	3	N/A
RLWI	Solidified Waste Drum	Liquid Waste Solidification Cell	Activation and Fission Products	125[d]	< 0.25
Solid Radwaste	Spent Filters	Supercell	Iodine	400	< 1
SCAS	Subcritical Multiplication Source	IU Cell	Alpha-neutron Source (PuBe or AmBe)	[ ]SRI	N/A

a. Descriptions of the systems and their physical characteristics can be found in Chapter 4.
b. Value is per irradiation unit (IU).
c. Value is per cycle.
d. Value is per drum.

Table 11.1-11 – Administrative Radiation Exposure Limits

Type of Dose	10 CFR 20 Limit (rem/year)	SHINE Administrative Limit (rem/year)
The more limiting of:		
Total effective dose equivalent to whole body, or	5	2
Sum of deep-dose equivalent and committed dose equivalent to any organ or tissue other than lens of eye	50	20
Eye dose equivalent to lens of eye	15	6
Shallow-dose equivalent to skin of the whole body or any extremity	50	20
Declared Pregnant Worker		
Dose to embryo/fetus during the entire pregnancy: taken as the sum of the deep-dose equivalent to the woman and the dose to the embryo/fetus from radionuclides in the embryo/fetus and the woman	0.5 rem per gestation period	0.5 rem per gestation period
Individual Members of the Public		
Total effective dose equivalent	0.1	0.1

Table 11.1-12 – Radiation Monitoring Equipment

Radiation Monitoring Instrument Type[a][b]	Location	Function
Radiation Survey Instruments		
Portable dose rate – neutron	Various	Routine and job coverage surveys
Portable dose rate – beta/gamma	Various	Routine and job coverage surveys
Friskers	Various egress points within the RCA	Ensure effective control of the spread of contamination
Personnel contamination monitors	Egress points from RCA	Verify effectiveness of contamination controls
Laboratory/Benchtop Instruments		
Liquid Scintillation Counter (LSC)	Counting room	Tritium and low energy beta-emitting radionuclide sample analysis
Low Background Sample Counter – alpha/beta	Counting room	Count smears and air samples
Gamma Spectroscopy	Counting room	Various gamma-emitting radionuclide sample analyses
Air Sampling and Monitoring		
Personnel Lapel Sampler	Various as specified by procedure or RWP	Representative air monitoring during work; internal dose assignment
Air Samplers	Various as specified by procedure or RWP	Airborne radioactivity concentration measurement
Continuous Alpha / Beta Air Monitor (CAM)	Areas where airborne contamination may be present, as specified by procedure	Early detection of unanticipated increases in airborne radioactivity concentration
Continuous Tritium Air Monitor	See Table 7.7-3	See Table 7.7-3
Radiation Area Monitors		
Radiation Area Monitors (RAM)	See Table 7.7-2	See Table 7.7-2
Radiological Effluent Monitor		
Stack Release Monitor	Located in the main production facility stack	Direct exposure to gamma and beta emitting radionuclides released through the stack of the SHINE main production facility is monitored and measured
Charcoal Delay Bed Effluent Monitor	Located at the outlet of the process vessel vent system (PVVS) charcoal delay beds	Monitor to trend the performance of the charcoal delay beds. Ensure the PVVS effluent stream is monitored if the safety-related effluent release point is in use.

a. See Table 7.7-1 for safety-related process radiation monitors.
b. See Table 11.1-14 for Environmental Monitoring equipment and locations.

Table 11.1-13 – Radiological Postings

Posting	Requirement
CAUTION RADIATION AREA	Accessible area in which radiation levels could result in an individual receiving in excess of 5 mrem in one hour 30 cm from the radiation source or surface that the radiation penetrates.
CAUTION HIGH RADIATION AREA or DANGER HIGH RADIATION AREA	Accessible area in which radiation levels could result in an individual receiving in excess of 100 mrem in one hour 30 cm from the radiation source or surface that the radiation penetrates.
GRAVE DANGER VERY HIGH RADIATION AREA	Accessible area in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or from any surface that the radiation penetrates.
CAUTION AIRBORNE RADIOACTIVITY AREA or DANGER AIRBORNE RADIOACTIVITY AREA	Licensed airborne radioactive materials in a room, enclosure, or area exists in concentrations exceeding the derived air concentrations specified in 10 CFR 20, Appendix B, Table I, or when 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% of the annual limit on intake or 12 DAC–hours.
CAUTION CONTAMINATION AREA	An area where removable contamination levels are above 20 dpm/100 cm^2 of alpha activity or 1,000 dpm/100 cm^2 beta/gamma activity.
CAUTION RADIOACTIVE MATERIAL(S) or DANGER RADIOACTIVE MATERIAL(S)	Areas or rooms in which there is use of, or stored, an amount of licensed radioactive material exceeding 10 times the quantity of material in Appendix C to 10 CFR 20.

Table 11.1-14 – Environmental Monitoring Locations

Monitoring Type	Location	Rationale
Groundwater Sampling		
Test Well SM-GW4A Sampling	Test well located directly west of the SHINE facility	The groundwater gradient is to the west and the south and thus any groundwater contamination is likely to flow to the west and to the south.
Test Well SM-GW2A Sampling	Test well located directly south of the SHINE facility	The groundwater gradient is to the west and the south and thus any groundwater contamination is likely to flow to the west and to the south.
Environmental Dosimeters		
ED 1 - 16	Site Boundary	One in each of the 16 compass directions from the site center.
ED 17 - 20	Outside the main production facility but within the site boundary (ED 17 north, ED 18 east, ED 19 south, ED 20 west)	One in each of the four cardinal directions surrounding the main production facility.
ED 21 - 23	Rock County Christian Elementary School (ED 21) Jackson Elementary School (ED 22) University of Wisconsin – Rock County (ED 23)	Special interest areas (e.g., population centers, nearby residences or schools).
ED 24	Kennedy Elementary School	To serve as a control (i.e., located a significant distance from the facility such that is represents a background dose).
Air Samplers		
Air Sampler (CAS 1)	Off-site location, co-located with ED 24	Control air sampler located a sufficient distance from the SHINE facility such that airborne samples are unaffected by airborne effluent releases from the facility.
Air Sampler (CAS 2)	Close to property line, north of the main production facility, co-located with ED 1	This direction has high ground level deposition factor (D/Q) and is in the direction of Janesville. Since the community of Janesville is relatively close to the site boundary, this air sampler location is credited with satisfying two of the conditions for air sample location recommendations in Table 3.12-1 of NUREG-1301.
Air Sampler (CAS 3)	Close to property line, east of the main production facility, co-located with ED 5	This direction has high D/Q and is in the direction of dairy production and the horse pasture.
Air Sampler (CAS 4)	Close to property line, west of the main production facility, co-located with ED 9	This location ensures all directions are monitored.
Air Sampler (CAS 5)	Close to property line, south of the main production facility, co-located with ED 13	This location is in the direction of the nearest occupied structure.

Figure 11.1-1 – Probable Radiation Area Designations Within the SHINE RCA, Ground Floor Level

Figure 11.1-2 – Estimated Derived Air Concentrations, Ground Floor Level

Figure 11.1-3 – Radiation Protection Organization

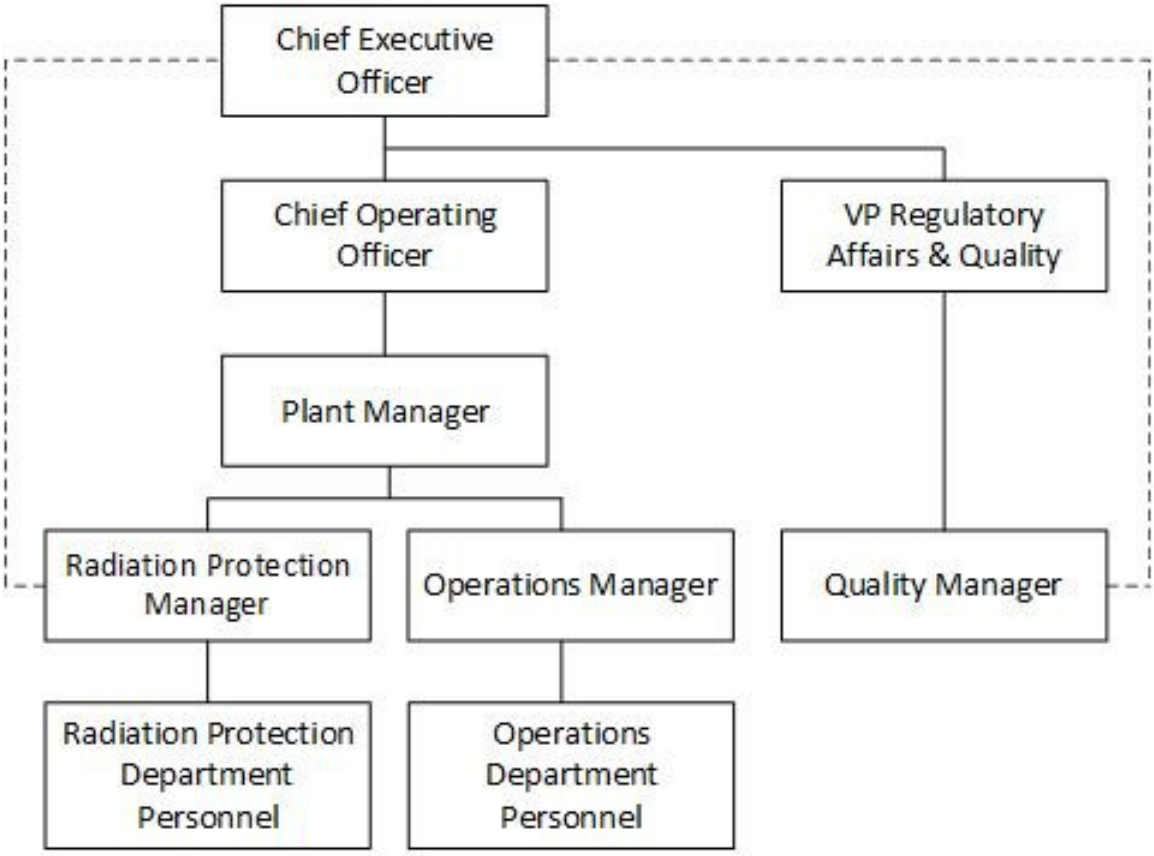
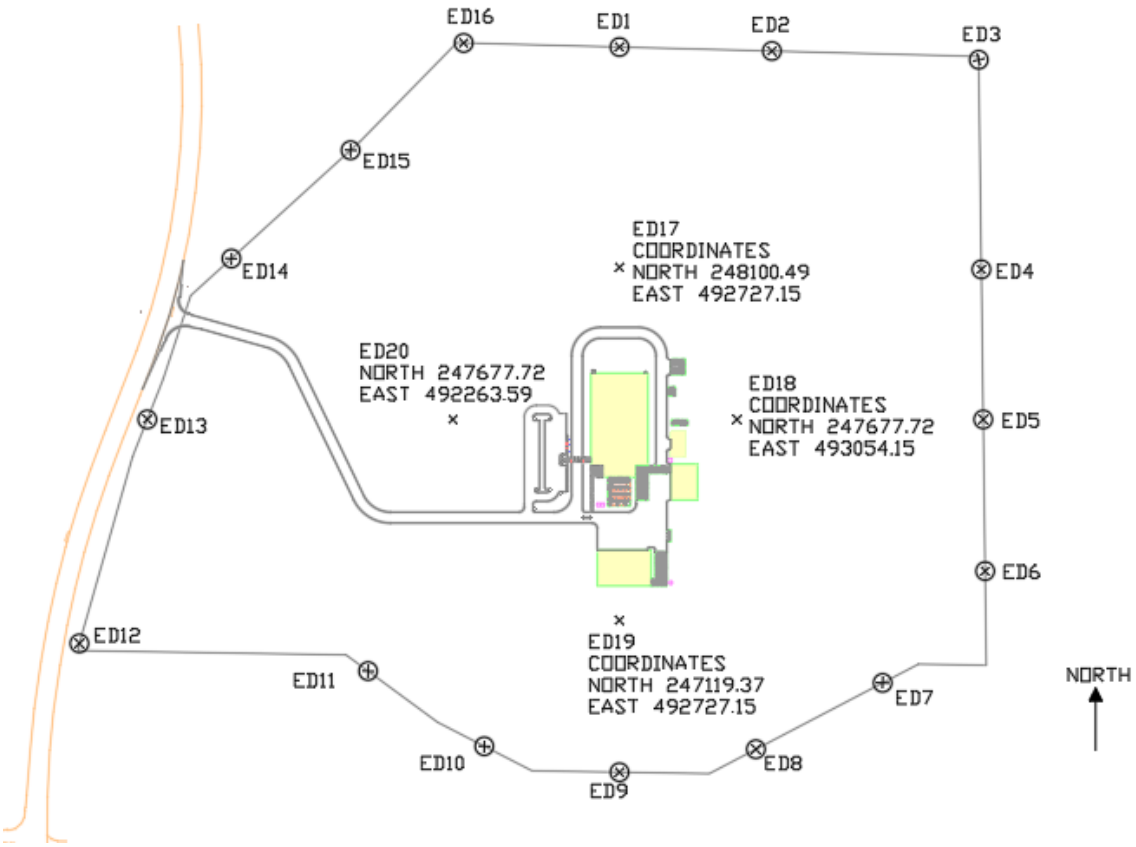


Figure 11.1-4 – Environmental Dosimeter Locations



11.2 RADIOACTIVE WASTE MANAGEMENT

SHINE produces medical isotopes by the fission of low enriched uranium (LEU) driven by accelerator-produced neutrons. Several irradiation and processing steps create liquid, gaseous, or solid radioactive waste materials. This section describes the management program, controls, and disposal pathways established to ensure proper identification, classification, control, processing (as required), and packaging, for each anticipated radioactive waste stream generated by the SHINE facility. SHINE is committed to comply with all applicable local and national regulations for managing radioactive wastes.

SHINE will comply with the following federal regulations related to radioactive wastes:

- 10 CFR 20, Standards for Protection Against Radiation
- 10 CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste
- 10 CFR 71, Packaging and Transportation of Radioactive Material
- 40 CFR, Chapter I, Subchapter F, Radiation Protection Programs
- 40 CFR, Chapter I, Subchapter I, Solid Wastes
- 49 CFR, Chapter I, Subchapter C, Hazardous Materials Regulations

SHINE is regulated by the NRC. The State of Wisconsin regulates radioactive waste once it leaves the SHINE facility and is transported. SHINE complies with Wisconsin regulations relating to the transportation and disposal of hazardous waste per Wisconsin Administrative Code Chapter NR 662. The State of Wisconsin implements the U.S. Department of Transportation (DOT) radioactive waste transportation regulations.

Radioactive wastes are prepared for shipment in approved shipping containers and shipped off-site using common or contract carriers in compliance with DOT regulations (49 CFR) and 10 CFR 20, 10 CFR 61 and 10 CFR 71, as applicable.

SHINE complies with the waste acceptance criteria (WAC) of the selected licensed disposal facilities, including any local or state regulations specified in those criteria. The State of Wisconsin is in the Midwest Interstate Low-Level Radioactive Waste Compact. Waste disposal sites available for this compact include:

- Energy*Solutions* in Clive, UT
- Waste Control Specialists (WCS) in Andrews, TX

Section 11.1 describes the program and procedures for controlling and assessing radioactive exposures associated with radioactive sources, including radioactive waste streams.

11.2.1 RADIOACTIVE WASTE MANAGEMENT PROGRAM

The Radioactive Waste Management Program is coordinated with the Radiation Protection Program under the Plant Manager. The goal of the Radioactive Waste Management Program is to minimize waste generation, minimize exposure of personnel, and to protect the general public and environment. The authority, duties, and responsibilities of personnel in the waste management organization are prescribed in the Radioactive Waste Management Program document.

11.2.1.1 Plant Manager

The Plant Manager reports to the Chief Operating Officer. The Plant Manager has overall responsibility for the safe operation of the SHINE facility and is responsible for ensuring the protection of personnel from radiation exposure resulting from processing, handling and storing radioactive material and waste. The Plant Manager's responsibilities are to:

- Assign responsibility and delegates commensurate authority to implement the Radioactive Waste Management Program.
- Provide waste management staff appropriate to the scope of operations and experienced in waste management operations.
- Ensure that the waste management self-assessment program is implemented.
- Ensure compliance with applicable federal and state regulations, and facility license conditions.
- Approve changes to the facility Process Control Program.

11.2.1.2 Radiation Protection Manager

The Radiation Protection Manager reports to the Plant Manager. The Radiation Protection Manager is responsible for establishing and maintaining the Radioactive Waste Management Program. The Radiation Protection Department maintains organizational independence from the Operations Department. The Radiation Protection Manager and Radiation Protection staff responsibilities are to:

- Develop waste management procedures for the processing, packaging and shipment of radioactive waste from the facility.
- Ensure that the concept of ALARA is incorporated into the Radioactive Waste Management Program procedures and is practiced by personnel.
- Process radioactive waste generated at the facility.
- Provide technical input to the design of equipment and processes.
- Perform radiological analysis tasks supporting the Radioactive Waste Management Program.
- Provide technical input to the Radioactive Waste Management Program training program.
- Maintain contractual relationships with waste disposal sites, waste processing facilities, and radioactive waste carriers.
- Maintain working knowledge of waste disposal acceptance criteria, regulations, standards and guides.
- Conduct self-assessments of radioactive waste management practices and compliance with procedures.

11.2.1.3 Training Manager

The Training Manager reports to the Plant Manager and is responsible for implementation of the Radioactive Waste Management Program training as described in the Radiation Protection Program. The Training Manager has the following responsibilities:

- Develops the waste management training and qualification program in accordance with facility procedures and ensuring compliance with 49 CFR 172, Subpart H, Training.
- Provides training to personnel commensurate with the radiological waste hazard to which they may be exposed.

- Provides re-training of previously-trained waste management personnel at least once every three years. Includes training on procedure changes and changes in required skills.
- Evaluates the waste management and qualification training program periodically. Reviews program content to ensure it remains current and adequate to ensure worker safety.

11.2.1.4 Quality Manager

The Quality Manager reports to the Vice President Regulatory Affairs & Quality. The Quality Manager has the following responsibilities:

- Review and audit facility radioactive waste handing, storing and shipping activities in accordance with the Quality Assurance Program Description to verify compliance with facility procedures, applicable federal and state regulations and applicable regulatory guides.

11.2.1.5 Shipping Personnel

Individuals who perform the duties of shipping radioactive waste, are trained in accordance with 49 CFR 172, Subpart H, Training.

11.2.1.6 Radioactive Waste Management Procedures

Radioactive Waste Management Program implementing procedures are developed to provide direction for efficient and safe conduct of waste operations. The procedures include applicable controls and limits significant to the waste management operation. The procedures include:

- Waste minimization and pollution prevention, including process controls to minimize generation of waste and separation of radioactive waste and nonradioactive waste to reduce volumes of radioactive wastes.
- Radiological characterization and waste classification.
- Operating and process controls with parameters for processing wastes.
- Verification of compliance with disposal and processor site WAC.
- Preparation of radioactive waste for shipment, including preparation of manifests and notifications, and measures for security on site and during transport.
- Container specifications, selection, packaging wastes, inspections, vehicle inspections, and proper loading and shoring of shipments.
- Marking, labeling and placarding requirements.
- Radioactive materials and contamination survey requirements and limits for shipment on public highways.
- Waste disposal recordkeeping.
- Interim waste storage controls and recordkeeping.

The Radiological Waste Management Program and implementing procedures are developed and controlled in accordance with SHINE's document control requirements.

11.2.1.7 Record Keeping and Document Controls

Records are developed and retained in accordance with the requirements specified in the Radiation Protection Program (see Subsection 11.1.2.1.8), the SHINE Document Control

Program, and as specified in federal and state regulations applicable to the Radioactive Waste Management Program.

11.2.1.8 Waste Management Audits

Facility radioactive waste management audits are conducted, at a minimum, on an annual basis in accordance with 10 CFR 20.1101(c) for the purpose of reviewing the functional and safety elements of the radioactive waste management program. The audits also evaluate programmatic efforts to minimize production of radioactive wastes. The audit activity is led by the Review and Audit Committee (see Section 12.2) as a subset of the Radiation Protection Program audit and the results are sent to executive management. Any deficiencies identified by the audit are addressed by the corrective action process.

11.2.1.9 Technical Specifications

Variables, conditions, or other items that may be subjects of a technical specification associated with radioactive waste management are contained in the facility Technical Specifications.

11.2.2 RADIOACTIVE WASTE CONTROLS

Radioactive waste is generally considered to be any item or substance which is no longer of use to the facility and which contains radioactivity above the established natural background radioactivity. The wastes generated by the SHINE facility are not spent nuclear fuel, high-level waste, or byproduct material as defined in paragraphs (2), (3) and (4) of the definition of Byproduct Material set forth in 10 CFR 20.1003. Therefore, the radioactive wastes generated by the SHINE facility are all classified as low level waste (LLW). The LLW generated by the SHINE facility during operation is expected to be classified as Class A, Class B or Class C waste. The neutron multipliers are designed for the life of the facility and will be disposed of as greater-than Class C (GTCC) waste during decommissioning.

For the purposes of transportation, packaged wastes may be categorized as low specific activity (LSA), requiring Type A packaging, or requiring Type B packaging.

For the purposes of both transportation and operational ALARA, wastes may be categorized as either contact handled or remote handled. The upper limit for remote handled waste dose rates is defined based on payload limits for the specific shielded transportation casks used and on WAC for the intended disposal site.

Radiation Protection Program requirements and the ALARA Program (see Section 11.1) apply to radioactive waste management, including, but not limited to, control of materials, monitoring and surveys, radiologically controlled area (RCA) access control, contamination control and personnel monitoring. ALARA goals and implementation are detailed in Subsection 11.1.3.

The material staging building is used for interim storage of wastes for decay and for preparation for shipment. Wastes are not stored for more than five years. The material staging building design evaluated the shielding provided by the building to ensure 10 CFR 20 site dose limits are met and ALARA principles are followed.

Radioactive waste management operating procedures are discussed in Subsection 11.2.1.6. These procedures ensure proper identification, characterization, and separate treatment of radioactive wastes.

11.2.2.1 Radioactive Waste Minimization

Waste minimization and pollution prevention are key elements of the Radiological Waste Management Program. Implementing procedures (see Subsection 11.2.1.6) address:

a. Responsibilities for waste minimization and pollution prevention.
b. Employee training and education on general environmental activities and hazards regarding the facility, operations, pollution prevention, waste minimization requirements, goals and accomplishments.
c. Setting goals for reducing the volume or radioactivity in each waste stream.
d. Sorting and compaction to reduce the volume of solid waste.
e. Segregation of nonradiological and radiological wastes to reduce the volume of radiological waste due to contamination.
f. Process controls that minimize generation of wastes.
g. Periodic assessments to identify opportunities to reduce or eliminate the generation of wastes.
h. Recognition of employees for efforts to improve waste minimization and environmental conditions.

11.2.2.2 Waste Stream Sources

Waste management operations occur in the main production facility and the material staging building (see Figure 1.3-1 and Figure 1.3-3). At least 5,600 square feet (ft^2) of the material staging building is for temporary storage to allow for decay. As allowed by the waste drum design, building design, and programmatic controls (e.g., inspection requirements), drums may be stored in multiple layers. Equipment and associated features for containment and/or packaging, storage, and disposal of solid, liquid, and gaseous radioactive waste are discussed in Subsection 9b.7.3, Subsection 9b.7.4, and Subsection 9b.7.5.

Changes to the facility will be performed in accordance with 10 CFR 50.59, Changes, Tests and Experiments, and will be assessed for their impact on radioactive waste sources or management, as applicable.

Table 11.2-1 summarizes the facility waste streams, characteristics, generation rates, and shipment categories. The waste streams and typical waste classifications are described in the following subsections.

11.2.2.2.1 Uranium Receipt and Storage System

Waste generated by uranium receipt and storage includes used cannisters in which new uranium metal and uranium oxide are received. The used cannisters are processed as Class A waste, if not returned to the supplier. The uranium receipt and storage system (URSS) utilizes gloveboxes with high efficiency particulate air (HEPA) filters in the air supply and return lines. The spent HEPA filters are Class A waste.

11.2.2.2.2 Target Solution Preparation System

The target solution preparation process may generate waste in the form of spent filters from the uranyl sulfate dissolution tanks, if not cleaned and reused, and spent HEPA filters from glovebox air supply and return lines. The spent filters are Class A waste.

11.2.2.2.3 Irradiation Unit

An irradiation unit (IU) consists of a subcritical assembly system (SCAS) coupled with a neutron driver assembly system (NDAS). The IU components become activated during their service life. SCAS major components are designed for the life of the facility and are not anticipated waste streams. Spent NDAS components are Class A waste. Contaminated oil from the NDAS vacuum pumps is Class B waste.

11.2.2.2.4 TSV Off-Gas System

The target solution vessel (TSV) off-gas system (TOGS) removes radiolysis and fission product gases from the TSV during irradiation operation and from the TSV dump tank during cool down operation. There are a total of eight independent TOGS, one for each IU.

The TOGS contains skid-mounted equipment that includes recombiner beds, demisters, and zeolite beds. Skid replacement occurs infrequently. Skids containing recombiner beds and demisters are treated with an acid flush and processed as Class A or B waste. Zeolite beds are designed for the life of the facility, however, if replaced more frequently and processed separately from the remainder of the skid components, the zeolite beds are expected to be Class B or Class C waste.

11.2.2.2.5 Molybdenum Extraction and Purification System

The molybdenum extraction and purification system (MEPS) separates molybdenum from an irradiated uranyl sulfate target solution. The molybdenum is then concentrated and purified into a sodium molybdate solution. The MEPS is located within a series of hot cells. Waste generated from the MEPS includes spent molybdenum extraction columns, [ ]PROP/ECI, and purification glassware. MEPS liquid wastes are processed by the radioactive liquid waste immobilization (RLWI) system.

Spent extraction columns [ ]PROP/ECI are stored in a hot cell, then transferred to the drum storage bore holes for decay, and ultimately disposed as Class B or C waste.

The glassware used in this process is not expected to contain significant quantities of long-lived radionuclides and is Class A waste.

[ ]PROP/ECI associated with MEPS column washes occurs in the RLWI system. The [ ]PROP/ECI are disposed as Class B or Class C waste.

11.2.2.2.6 Process Vessel Vent System

The process vessel vent system (PVVS) removes radioactive particulates, iodine, and noble gases that are generated within the radioisotope production facility (RPF) and primary system boundary (PSB) prior to being discharged to the atmosphere. PVVS waste consists of spent HEPA filters and spent carbon guard beds. The spent HEPA filters are Class A waste and the spent carbon guard beds are Class A or Class B waste. Condensate from PVVS can be blended with other waste streams and processed by RLWI.

11.2.2.2.7 Iodine and Xenon Purification and Packaging System

The iodine and xenon purification and packaging (IXP) system separates the iodine fission products from the uranyl sulfate target solution or from [ ]PROP/ECI. The IXP system generates spent iodine recovery, [ ]PROP/ECI.

Iodine recovery, [ ]PROP/ECI will be regularly changed out and are Class B or Class C waste.

11.2.2.2.8 Hot Cells

Hot cells contain HEPA and carbon filter combinations on the air supply and return lines. Spent HEPA and carbon filters are Class A waste.

11.2.2.2.9 Primary Closed Loop Cooling System

The primary closed loop cooling system (PCLS) has potential for radioactive contamination due to minor leakage from the PSB and activation products. Contamination would collect on the PCLS filters and deionizer resins. PCLS filters could become contaminated with radionuclides due to activation of corrosion particles as the water passes through the TSV, however, corrosion of the stainless steel components is expected to be small. The spent PCLS filters are expected to be Class A waste. PCLS deionizer resins are contained in disposable deionizer units. The tanks are designed for complete replacement without removal of the ion exchange resins in the tanks. The disposable tanks are Class A waste.

11.2.2.2.10 Light Water Pool System

The light water pool has potential for radioactive contamination due to minor leakage from the PSB and activation products. Any contamination would collect on the filters and deionizer resins used to cleanup the light water pool. Similar to the PCLS, the deionizer resins are contained in disposable deionizer units and are expected to be Class A waste. Spent filters are expected to be Class A waste.

11.2.2.2.11 Radioactive Liquid Waste

Radioactive liquid waste streams include waste liquids from:

- MEPS
- IXP system
- PVVS

- Decontamination liquid waste from decontamination of structures, systems, and components (SSCs) during normal operation
- Laboratory liquid waste

The liquid waste streams are shown in Table 11.2-1.

Liquid waste streams are collected in uranium liquid waste and radioactive liquid waste collection tanks, consolidated in liquid waste blending tanks and treated for disposal using the RLWI system. The quantity and size of the tanks are managed to maximize decay time and provide a buffer for upset conditions. Each uranium liquid waste tank has at least [ ]PROP/ECI capacity and the liquid waste collection and blending tanks each have at least 600 gallons capacity. Hold times for decay are based on minimizing dose rates to workers during the immobilization process. Solidified liquid waste is expected to be Class A.

The chemical composition and relative radiological inventory of liquid waste streams is presented in Table 11.2-6.

11.2.2.2.12 Radioactive Gaseous Waste

Airborne radioactive sources are present in the tritium purification system (TPS), PVVS, TOGS, vacuum transfer system (VTS), and the NDAS. Airborne radioactive sources and release are addressed in Subsection 11.1.1.1 and Table 11.1-5.

The RCA ventilation systems generate spent prefilters, HEPA filters and carbon filters that are Class A generated solid waste.

11.2.2.3 Technical Specifications

Variables, conditions, or other items that may be subjects of a technical specification associated with radioactive waste controls are contained in the facility Technical Specifications.

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 WAC of an established, licensed LLW disposal facility. Processing may be comprised of one or more of several operations, including compaction, solidification with an appropriate solidification agent, adsorption onto a solid medium (e.g., elemental iodine onto activated carbon filters), interim storage for decay of radionuclides, consolidated handling and processing, extraction and consolidation of radionuclides by segregation, and mixing (possibly from more than one waste stream) so that the bulk volume of waste is readily disposable.

Radiation monitoring of effluent waste streams is described in Section 7.7. Radiation monitoring requirements are also described in the Radiation Protection Program. The Radiation Protection Program is described in detail in Subsection 11.1.2.

Liquid effluent is not routinely discharged from the RCA. Radioactive liquid discharges from the SHINE facility to the sanitary sewer are infrequent and made in accordance with 10 CFR 20.2003 and 10 CFR 20.2007. There are no piped liquid effluent pathways from the RCA to the sanitary sewer. Sampling is used to determine suitability for release.

Table 11.2-1 shows the anticipated waste generation, classifications, shipment types, and expected disposal sites for the identified waste streams. Final determinations of waste classification and management will be made in accordance with the Radioactive Waste Management Program implementing procedures.

11.2.3.1 Solid Wastes

The subsections below discuss the methodology for the eventual release of the major solid wastes generated by the SHINE facility. Processing requirements are in accordance with the receiving facility's WAC and will be modified as needed to reflect any change in the disposal site or WAC.

11.2.3.1.1 Irradiation Units

Solid waste streams associated with the IUs are the NDAS activated components. The NDAS is comprised of an accelerator section, pumping section, roots stack, and target chamber assembly. The target chamber assembly is expected to be Class A waste and the WAC specified by Energy*Solutions* will apply. The accelerator stage, pumping stage and roots stack are considered "oversize" and must meet specific WAC applicable to oversize components. Table 11.2-2 displays the typical methodology associated with disassembly and processing of this waste stream.

11.2.3.1.2 Spent Columns

Spent molybdenum extraction columns, [ ]$^{PROP/ECI}$, and IXP recovery, [ ]$^{PROP/ECI}$ will be held in hot cells for decay, then consolidated into supercell export waste drums prior to disposal.

The columns are removed from the process lines using quick-disconnect style inlet and outlet connectors specifically designed for use with remote manipulators in hot cell environments. Radiation and wear-resistant seals and automatically closing valves built into the connectors provide leak tightness to minimize or prevent leakage.

After removing a spent column from the originating process, it is stored in a hot cell for sufficient time to allow short-lived fission products to decay. After several columns have decayed, they are transported out of the cell in one transfer to reduce personnel exposure and the number of transfer operations. The number of columns transferred is limited based on export waste drum capacity. The export waste drum is shielded to ensure personnel doses are maintained ALARA and within procedure limits during the transfer. The estimated dose rate for an extraction column, at the time of process removal is approximately 9500 rem/hr at 3 feet unshielded. The peak dose rate drops to approximately 580 rem/hr at 3 feet unshielded after storage in the hot cell.

When a set of columns are to be transferred out of the hot cell, they are remotely loaded into an export waste drum within a shielded cask. Dose rates from the cask and contamination levels are confirmed to be within limits, then the cask is remotely transported to a bore hole for interim below-grade storage. The shielded cask is surveyed and decontaminated, if needed, prior to reuse.

When a shipment of columns is to be prepared, the export waste drum is retracted using the remote-controlled grappler and placed into a shielded cask and the cask is transported to an area for loading into a Type B shipping container.

The spent columns are expected to be Type B or C generated waste and have no specified time requirement in storage. The spent columns are stored in order to consolidate shipments to minimize handling for ALARA and to consolidate the columns to reduce disposal volumes. Requirements for this waste stream are presented in Table 11.2-3.

[

]PROP/ECI

11.2.3.1.3 Process Glassware

Spent molybdenum purification glassware is remotely handled to move the glassware from the hot cell to an export waste drum. The glassware may be crushed in the waste drum using a remotely controlled compactor and transported to the material staging building in a shielded transport cask. Requirements for this waste stream are presented in Table 11.2-4.

11.2.3.1.4 Zeolite Beds

The silver coated zeolite beds are a component of the TOGS and are provided to remove iodine from the sweep gas. Toxicity characteristic leaching procedure (TCLP) would result in the classification of this waste as Resource Conservation and Recovery Act (RCRA) waste; however, the waste is also radioactive and as such may be a mixed low level waste (MLLW). The waste classification for this material is a function of both the efficiency of the zeolite beds and the change out frequency of the beds. The design goal is for the beds to last the lifetime of the facility; however, this waste stream is assumed to be replaced every five years. The zeolite bed has the potential to be Class B or Class C waste.

11.2.3.1.5 Recombiner Beds, Demister and Component Replacement

This waste stream is associated with the TOGS. This waste stream is based on infrequent replacement of the TOGS skids. Acid flushing of the skid components (excluding the zeolite beds) will be performed prior to disposal. Cs-137 and Sr-90 are expected to dominate the waste classification. Remote handling and packaging may be required due to considerable dose rates expected should replacement be required. This waste stream is Class A or Class B waste.

11.2.3.1.6 PCLS and LWPS Deionizer Units

The PCLS and LWPS deionizer resins are contained in disposable deionizer units. The spent units are dewatered and disposed as Class A generated waste.

11.2.3.2 Liquid Waste Streams

Several waste streams are solidified on site to meet DOT criteria and disposal site WAC, as described in Subsection 11.2.2. The consolidated liquid waste stream (post-treatment) is amenable for disposal as Class A waste at Energy*Solutions*.

11.2.3.2.1 Consolidated Liquids

Radioactive liquid waste and estimated generated volumes are provided in Table 11.2-1.

Uranium liquid wastes and other radioactive liquid wastes are collected and processed separately, then blended prior to solidification. Uranium liquid wastes may consist of molybdenum extraction column acid wash, extraction column water wash, iodine recovery column [ ]PROP/ECI, VTS knockout pot contents, spent target solution, or decontamination waste. Radioactive liquid waste may consist of [ ]PROP/ECI, purification waste, [ ]PROP/ECI, or PVVS condensate. Blending of wastes is performed without exceeding the maximum uranium concentration applicable to the receiving disposal site. Certain fissile material may be exempted under 10 CFR 71.15.

This waste stream process includes removal of radionuclides, radioactive decay, pH adjustment, blending of uranium and radioactive liquid wastes, and solidification in 55-gallon drums using a solidification agent.

The anticipated disposal site for the solidified liquid waste is Energy*Solutions*.

Requirements for this waste stream are presented in Table 11.2-5.

11.2.3.3 Gaseous Waste Streams

Airborne radioactive sources are identified in Subsection 11.1.1 and Table 11.1-5. The RCA ventilation system filtering and exhaust stack discharge is described in Subsection 9a2.1.1. The exhaust stack location is shown on Figure 1.3-1. The stack release monitor provides continuous monitoring of radioactive noble gas stack releases and a means to sample and measure the stack air for particulate, iodine, and tritium concentration to ensure compliance with gaseous effluent regulatory limits. The estimate of annual release of radionuclides is provided in Table 11.1-8. The effect of releases on the surrounding environment is addressed by the Environmental Monitoring Program described in Subsection 11.1.7.

Table 11.2-1 – Estimated Annual Waste Stream Summary
(Sheet 1 of 2)

Description	Matrix	Class as Generated	As Generated Amount	As Generated Units	As Disposed (ft^3)	Shipment Type	Destination[a]
MEPS Extraction Columns [ ]PROP/ECI	[ ]PROP/ECI	B or C	[ ]PROP/ECI	ft^3/yr	270	Type B	WCS
[ ]PROP/ECI	[ ]PROP/ECI	B or C	72	ft^3/yr	72	Type B	WCS
IXP Separation Columns	[ ]PROP/ECI	B or C	[ ]PROP/ECI	ft^3/yr	35	Type B	WCS
LWPS Deionizer Units	Resin	A	48	ft^3/yr	80	Type A or LSA	Energy*Solutions*
PCLS Deionizer Units	Resin	A	48	ft^3/yr	80	Type A or LSA	Energy*Solutions*
Uranium Canisters	Solid	A	2.0[b]	ft^3/yr	3.3	Type A or LSA	Energy*Solutions*
NDAS Accelerator Subassembly	Solid	A	[ ]PROP/ECI	ft^3/yr	13,600	Type A or LSA	Energy*Solutions*
NDAS Target Chamber Subassembly	Solid	A	[ ]PROP/ECI	ft^3/yr	1330	Type A or LSA	Energy*Solutions*
TOGS Skids	Solid	A or B	922	ft^3/yr	1540	Type A, B, or LSA	Energy*Solutions* or WCS
TOGS Zeolite Beds	Solid	B or C	0.64	ft^3/yr	1	Type B	WCS
LWPS Filters	Solid	A	1.6	ft^3/yr	2.7	Type A or LSA	Energy*Solutions*
PCLS Filters	Solid	A	1.6	ft^3/yr	2.7	Type A or LSA	Energy*Solutions*
TSPS, URSS, PVVS, Hot Cell, RVZ1, RVZ2, RLWI HEPA Filters	Solid	A	182	ft^3/yr	142	Type A or LSA	Energy*Solutions*
Hot Cell, RVZ1, RVZ2 Charcoal Filters	Solid	A	32	ft^3/yr	54	Type A or LSA	Energy*Solutions*
TSPS Uranyl Sulfate Solution Filters	Solid	A	0.35[c]	ft^3/yr	0.58	Type A or LSA	Energy*Solutions*

Table 11.2-1 – Estimated Annual Waste Stream Summary (Sheet 2 of 2)

Description	Matrix	Class as Generated	As Generated Amount	As Generated Units	As Disposed (ft^3)	Shipment Type	Destination[a]
PVVS Carbon Guard Bed	Solid	A or B	0.48	ft^3/yr	0.81	Type B	Energy*Solutions* or WCS
MEPS Glassware	Solid	A	208	ft^3/yr	347	Type A or LSA	Energy*Solutions*
Class A Trash	Solid	A	400[d]	ft^3/yr	677	Type A or LSA	Energy*Solutions*
Contaminated Oil	Oil	B	2	ft^3/yr	3.3	Type A or LSA	WCS
Extraction Column Acid Wash	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI	2,599[f][g]	Type A or LSA	Energy*Solutions*
Extraction Column Water Wash	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
[ ]PROP/ECI	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
[ ]PROP/ECI	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
[ ]PROP/ECI	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
Iodine Recovery Column [ ]PROP/ECI	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
Spent Target Solution	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
Vacuum Transfer System Knockout Pot	Liquid[e]	A	14	gal/yr			
Radiological Laboratory Waste	Liquid[e]	A	275	gal/yr			
Decontamination Waste	Liquid[e]	A	2,768	gal/yr			
Cintichem Purification Waste & Rotary Evaporator Condensate	Liquid[e]	A	82	gal/yr			
[ ]PROP/ECI	Liquid[e]	A	[ ]PROP/ECI	[ ]PROP/ECI			
PVVS Condenser Condensate	Liquid[e]	A	701	gal/yr			

a. Waste destination may be subject to change.
b. Uranium metal and/or uranium oxide cannisters may be returned to the supplier in lieu of disposition as solid waste.
c. TSPS uranyl sulfate dissolution tank filter elements may not become a waste stream if reconditioned and reused.
d. Class A trash is exclusive of other solid wastes identified in the table.
e. Liquid waste streams may be reused or may be combined and treated as a homogenous influent waste stream and solidified together.
f. As shipped volume of liquid waste streams is in the form of a uniform solidified matrix using a solidification agent.
g. 25 percent margin has been added to volume of solidified liquid shipped waste.

Table 11.2-2 – Waste Methodology for Accelerator

Requirement	Basis
Disassemble irradiation unit (separate accelerator section, pumping section, and roots stack from the target chamber assembly).	Operational requirement.
Determine if free liquid is present and absorb liquids, if present.	Required to meet WAC maximum free liquids requirement of 1 percent. This is particularly applicable to drift tubes and target chamber section waste.
Make waste characterization measurements.	Waste must be characterized in the manner appropriate and in conformance with the procedures of the destination to which it will be sent.
Provide capability to load oversized debris into cargo container.	Ensure capability to maneuver radioactive oversize debris.
Provide storage, waste segregation, consolidation and packaging capacity.	Items meeting the "standard debris" definition are shipped in a roll-off. One roll-off may be continuously stored in the material staging building. Oversized items (non-standard debris) are shipped in a cargo container. One cargo container may be continuously on-site.
Fill void space (if required) in accordance with the WAC.	Required to meet WAC requirement to minimize void space.

Table 11.2-3 – Waste Methodology for Spent Columns[a]

Requirement	Basis
Hold spent columns in hot cell for a period of decay sufficient to allow short-lived fission products to decay.	Spent columns are highly radioactive when removed from active service. Hold time is for decay and consolidated processing.
Remote transfer from hot cell to export waste drum.	Maintain worker dose ALARA.
Provide safe, shielded storage outside of hot cell.	Protected on-site storage until a full shipment of spent columns is prepared for disposal.
Provide management controls to ensure proper hold time is applied to spent columns.	Since multiple columns can be held in each hot cell post service, it is necessary to ensure each column has been held for a sufficient time to meet radiological dose requirements during handling prior to being transferred.
Determine if free liquid is present and absorb liquids, if present.	Required to meet WAC maximum free liquids requirement of 1 percent.
Fill void space (if required) in accordance with the WAC.	Required to meet WAC requirement to minimize void space.

a. Applicable to spent molybdenum extraction columns [ ]PROP/ECI and IXP recovery, [ ]PROP/ECI.

Table 11.2-4 – Waste Methodology for Process Glassware

Requirement	Basis
Remote transfer from hot cell to export waste drum.	Maintain worker dose ALARA.
Smear sample glassware.	Waste characterization to confirm disposal site and applicable WAC.
Glassware is compacted.	Glassware can be compacted for efficient packaging and transportation.
Determine if free liquid is present and absorb liquids, if present.	Required to meet WAC maximum free liquids requirement of 1 percent.
Fill void space (if required) in accordance with the WAC.	Required to meet WAC requirement to minimize void space.

Table 11.2-5 – Waste Methodology for Consolidated Liquids

Requirement	Basis
Collect uranium liquid waste and non-uranium liquid wastes separately.	Process separately prior to blending.
Apply hold time to uranium liquid waste.	Radioactive decay to achieve solidification product suitable for LSA or Type A packaging.
Consolidate uranium and non-uranium liquid wastes into blending tanks.	Liquid waste consolidation and processing.
Sample blended wastes after mixing.	A representative sample is required to verify maximum uranium concentration is not exceeded and for accurate waste characterization prior to solidification.
Solidify waste.	Use of a solidification agent to ensure final waste form meets requirements. Required to meet WAC maximum free liquids requirement for solidified waste forms (0.5 percent by volume).
Limit void space.	WAC requirement to minimize void space.
Establish dedicated area in the material staging building for decay or shipment consolidation.	Solidified waste may require decay post-processing to meet DOT limits.
Maintain records relative to drums in the storage area.	Drums may be held to decay to DOT limits.

Table 11.2-6 – Chemical Composition and Radiological Properties of Liquid Waste Streams (Sheet 1 of 2)

Description	Chemical Composition (wt/wt)	Estimated Annual Volume	Radiological Inventory[1]	Qualitative Radiological Properties
Extraction Column Acid Wash	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI	Medium	Most fission products pass through separation columns, though some are expected to be retained on the columns and then be removed with column washes.
Extraction Column Water Wash	>99% H_2O trace H_2SO_4 trace UO_2SO_4 [ ]PROP/ECI	[ ]PROP/ECI		
[ ]PROP/ECI	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
[ ]PROP/ECI	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
[ ]PROP/ECI	[ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
Iodine Recovery Column Washes	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
[ ]PROP/ECI	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
[ ]PROP/ECI	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI		
Spent Target Solution	[ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI [ ]PROP/ECI	[ ]PROP/ECI	High	Fission products remaining in the target solution after useful lifetime contribute to a relatively high radiological inventory.

Table 11.2-6 – Chemical Composition and Radiological Properties of Liquid Waste Streams (Sheet 2 of 2)

Description	Chemical Composition (wt/wt)	Estimated Annual Volume	Radiological Inventory(1)	Qualitative Radiological Properties
Vacuum Transfer System Knockout Pot	100% H_2O	14 gal.	Low	Liquid collected in the Knockout pot generally consists of condensed water vapor.
Radiological Laboratory Waste	99% H_2O 1.0% H_2SO_4 trace UO_2SO_4 [ ]PROP/ECI	280 gal.	Low	Laboratory waste is expected to consist of small sample volumes of highly-diluted process fluids.
Decontamination Waste	98% H_2O 1.0% H_2SO_4 1.0% UO_2SO_4 [ ]PROP/ECI	2,800 gal.	Varies	Dependent on decontamination needs.
Cintichem Purification Liquid Waste including Rotary Evaporator Condensate	98% H_2O 1.1% NH_4OH 0.88% HNO_3 trace HCl trace K_3RuCl_6 trace $KMnO_4$ trace α-benzoin oxime (ABO) trace $MoO_2(ABO)_2$ trace MoO_2 trace $NaNO_3$ trace $RhCl_3$	82 gal.	Low	Fission products remaining after majority removed in prior MEPS processing steps.
Process Vessel Vent System Condenser Condensate	100% H_2O	700 gal.	Low	Process vessel vent system condensate generally consists of condensed water vapor.

(1) Radiological inventory relative to other liquid waste streams.

11.3 RESPIRATORY PROTECTION PROGRAM

In accordance with 10 CFR 20, Subpart H, the respiratory protection program:

- Incorporates process and engineering controls, pursuant to 10 CFR 20.1701, to control the concentration of radioactive material in the air. The design of heating, ventilation, and air conditioning systems is described in Section 9a2.1.
- Implements other controls, pursuant to 10 CFR 20.1702, when it is not practical to apply process or engineering controls to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area. Consistent with the as low as reasonably achievable (ALARA) program described in Section 11.1, the respiratory protection program implements increased monitoring and limiting intakes by controlling access, limiting exposure times, and using respiratory protection equipment.
- Implements controls, pursuant to 10 CFR 20.1703, for the use of individual respiratory protection equipment to limit the intake of radioactive material. The respiratory protection program includes evaluation of potential hazards and estimated doses by performing surveys, bioassays, air sampling, or other means as necessary. The program provides protection of personnel from airborne concentrations exceeding the limits of Appendix B to 10 CFR 20 and ensures that respiratory equipment is tested and certified, including testing of respirators for operability before usage. The program ensures that written procedures specify the selection, fitting, issuance, maintenance, testing, training of personnel, monitoring, medical evaluations, and recordkeeping for individual respiratory protection equipment and for specifying when such equipment is to be used. Procedures for the use of individual respiratory protection equipment are revised as applicable when making changes to processes, facility, or equipment. Records are maintained for the respiratory protection program, including training in respirator use and maintenance.

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