CHAPTER 11

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

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Acronyms and Abbreviations

Acronym/Abbreviation	Definition
10 CFR	Title 10 of the Code of Federal Regulations
ALI	annual limit on intake
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
APF	assigned protection factor
ATS	accelerator target systems
Bq/100 cm^2	Becquerel per 100 square centimeters
CAM	continuous air monitors
CEMP	Community Environmental Monitoring Program
CEO	chief executive officer
Ci	curies (unit of measurement of radioactivity)
Ci/yr	curies per year
cm	centimeter
COO	chief operations officer
D/Q	ground level deposition factor
DAC	derived air concentration
DOT	United States Department of Transportation
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
DQO	data quality objectives
DSSI	Diversified Scientific Services, Inc.
EPA	U.S. Environmental Protection Agency
ES&H	environment, safety, and health
ft^3/yr	cubic feet per year
FSAR	final safety analysis report
GTCC	Greater-than-Class C

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
hr	hour
I	iodine
I-131	iodine-131
ICRP	International Commission on Radiological Protection
IF	irradiation facility
in.	inch
IU	irradiation unit
km	kilometers
Kr	krypton
kW	kilowatt
LEM	liquid effluent monitor
LEU	low-enriched uranium
LLW	low level waste
LSA	low specific activity
MCNP	Monte Carlo N-Particle
MEI	maximum exposed individual
MEPS	molybdenum extraction and purification system
mi.	miles
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

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
N-16	nitrogen-16
NCRP	National Council on Radiation Protection and Measurements
NDAS	neutron driver assembly system
NGRS	noble gas removal system
NIOSH	National Institute of Occupational Safety and Health
NNSS	Nevada National Security Site
NRC	U.S. Nuclear Regulatory Commission
pCi/cm^2	picocuries per square centimeter
PCLS	primary closed loop cooling system
PPE	personal protective equipment
PSAR	preliminary safety analysis report
QA	quality assurance
QC	quality control
RAM	radiation area monitor
RCA	radiologically controlled area
RCRA	Resource Conservation and Recovery Act
REMP	radiological environmental monitoring plan
RLWE	radioactive liquid waste evaporation and immobilization system
RPF	radioisotope production facility
RSC	Radiation Safety Committee
RWP	radiation work permit
SCAS	subcritical assembly system
Se	selenium
SHINE	SHINE Medical Technologies, Inc.

Acronyms and Abbreviations

Acronym/Abbreviation	Definition
SRM	stack release monitoring
Sv	sievert
Tc	technetium
TCAP	thermal cycling absorption process
TCLP	toxicity characteristic leaching procedure
TDN	thermal de-nitration
Te	tellurium
TOGS	TSV off-gas system
TPCS	TSV process control system
TLD	thermoluminescent dosimeter
TPS	tritium purification system
TRPS	TSV reactivity protection system
TSPS	target solution preparation system
TSV	target solution vessel
U-234	uranium-234
U-235	uranium-235
U-236	uranium-236
UNCS	uranyl nitrate conversion system
UREX	uranium extraction
WAC	waste acceptance criteria
WCS	waste control services
wt. %	weight percent
Xe	xenon
Zr	zirconium

CHAPTER 11

RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 RADIATION PROTECTION

11.1.1 RADIATION SOURCES

The SHINE Medical Technologies, Inc. (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 source term for normal operations in the SHINE facility. [redacted]. As the target solution is processed in the facility for Mo-99 extraction and fission product clean-up, radiation sources are transferred within the facility by means of pipes in shielded trenches.

There are three scenarios with assumptions listed in Table 11.1-1: nominal, limiting, and bounding values. The nominal parameter values or ranges are the targeted operating conditions for achieving the optimal operation of the TSV system. The limiting parameter values incorporate operation uncertainties for maximizing the nominal normal operations radionuclide inventory. The bounding parameter values envelope the limiting parameter values, and define the bounding normal operations source term relative to the TSV, TSV dump tank, supercell, and uranium extraction (UREX) processing cell.

The combination of input parameters in the bounding values column was shown to produce the maximum source term for normal operations. The “max” in the U-234 and U-236 concentration column refers to the maximum concentration in the specification for low enrichment uranium metal as delivered to SHINE. The other parameters ensure minimal extraction of [redacted] for each cycle and the minimal turn-over time between each cycle to maximize the source term after [redacted] cycles.

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 preliminary results for total activity in curies (Ci) from actinides and fission products contained within each TSV batch of target solution after [redacted] irradiation cycles. The “at shutdown” values represent the activity contained within the target solution immediately after shutdown of the neutron driver. The [redacted] values are the target solution activity when it is ready to be pumped 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. The “post extraction” value is the activity remaining in the target solution following

extraction of Mo, [ ].

Table 11.1-3 lists the activity associated with the radionuclides listed in NUREG/CR-4467 contained in one TSV batch of target solution that has been through 5.5-day irradiation cycles and after it has decayed for [ ] 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 activities in Table 11.1-3 are the bounding values case at].

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 Title 10 of the Code of Federal Regulations (10 CFR) 20 and 71, and with Department of Transportation regulations for shipment of product and waste (49 CFR). The goal for the normal operations dose rate for normally occupied locations in the facility is 0.25 millirem per hour (mrem/hr) at the surface. Radiation levels may rise above the 0.25 mrem/hr level during some operations such as tank transfers. Facility shielding is designed to meet this goal.

Major radiation sources in the balance of the facility originate in the target solution and in the TOGS. At the end of each TSV irradiation cycle, irradiated target solution is piped to one of the three supercells for processing. After irradiation cycles the target solution is also processed through the UREX process where fission products and uranium are separated. After each irradiation cycle off-gas is purged to one of the noble gas storage tanks to allow for decay of short-lived noble gas nuclides before being released through the facility exhaust stack. Airborne, liquid, and solid radioactive materials are contained in piping systems and tanks. Radiation levels for major system components and locations are provided below. RPF and process special nuclear material inventories are tabulated in Tables 4b.4-1 and 4b.4-13, respectively.

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 in the form of 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 noble gas removal system (NGRS) and the process vessel vent system (PVVS), both located in the RPF. These airborne radioactive materials are contained within closed systems consisting of piping components and tanks. Table 11.1-4 provides information on the various locations, types, and expected dose rates from gaseous radioactive sources.

Argon-41 is produced in the IU cells during operations. Argon-41 levels in the IU cell are diluted and diffused by RVZ1. Approximately 15.2 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 loops. Dose rates from this source are mitigated by biological shielding that limits radiation dose to occupied areas adjacent to the shielding.

Gaseous activity released from the target solution during the production process is collected and sent to the NGRS. This includes activity that may be released into the hot cells. No activity is released to the general access area of the facility, so no worker exposure to airborne activity is expected during process operations. Activity may be released during maintenance operations that require the opening of process systems. Radiation protection procedures are used to ensure that worker exposure to airborne activity is minimized during maintenance operations. Predicted personnel dose rates (including maintenance activity dose rates) and the associated methodology will be provided in the FSAR.

Gaseous activity from the TSV and process operations is held in noble gas storage tanks until radiodecay has reduced the activity such that releases are below the 10 CFR 20 limits. 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 (NRC, 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 11.1-9 contains the estimated gaseous activity production rates for a single TSV. Noble gases are the primary gaseous radionuclides produced in the TSV. Iodine is also volatile and is assumed to become airborne. Iodine is removed in the TOGS and a small fraction is transferred to the NGRS. The minimum holdup time in the NGRS is 960 hours (40 days). As shown in Table 11.1-9, many noble gas and iodine nuclides are short-lived and decay away during this holdup. The resulting annual release, with eight TSVs operating, is limited to a few nuclides as shown 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 (excluding tritium) at the MEI and the nearest resident are 9.0 mrem and 0.6 mrem, respectively, which is a small fraction of the 10 CFR 20.1301 limit of 100 mrem. The tritium purification system and neutron driver are designed such that the estimated annual doses to the MEI and the nearest resident are below the dose constraint specified in 10 CFR 20.1101(d).

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 two) are derived from the irradiated uranyl sulfate target solution as it is being processed through the facility. The first exception is the primary coolant, which carries activation product nitrogen-16 (N-16) along as it is pumped through the primary cooling heat exchangers. The second exception is the production of

low-activity fresh uranyl sulfate target solution. [redacted] These 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 doses from liquid radioactive sources.

Concentrations and solubilities of the major sources in the identified liquid radioactive source will be provided in the FSAR.

There are no radioactive liquid discharges from the SHINE facility. Liquid radioactive wastes generated at the facility are generally solidified and shipped to a disposal facility. See Table 11.2-1 for a list of liquid radioactive waste generated at the facility including the annual quantities and disposal destinations.

11.1.1.3 Solid Radioactive Sources

Radioactive sources exist in several locations in the SHINE facility. Fresh, low enrichment uranium is received at the facility in the form of uranium metal that has been enriched to a nominal 19.75 percent by weight in uranium-235 (U-235). The uranium metal is converted to uranium oxide and then to a liquid uranyl sulfate solution. The uranyl sulfate solution is stored until it is needed for a fresh TSV batch load. Other solid radioactive sources include the spent extraction columns from the molybdenum extraction process, spent filters, and the solidified liquid waste, which is a mixture of the waste and Portland cement with other solids and is contained in 55 gallon drums. At the end of the plant lifetime, the spent [redacted] neutron multiplier will also become solid radioactive waste requiring disposal.

The [redacted] neutron multiplier is located in the subcritical assembly. The [redacted] interacts with the neutron flux producing [redacted] that are retained within the [redacted] structure.

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

These solid radioactive sources are contained within IU cells, shielded cells, hot cells, or preparation areas within the radiologically controlled area (RCA) of the facility. Table 11.1-6 provides information on the various locations, types, and major sources in solid radioactive sources. The radionuclide inventory in the solid waste system is a function of the TSV and UREX 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

Technical specifications associated with the facility radiation sources will be provided in the FSAR.

11.1.2 RADIATION PROTECTION PROGRAM

This subsection describes the 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 and 20.

11.1.2.1 Commitment to Radiation Protection Program Implementation

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. SHINE develops, documents, and implements its Radiation Protection Program commensurate with the risks posed by a medical isotope production facility. 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 program content and implementation are reviewed at least annually as required by 10 CFR 20.1101(c).

SHINE's philosophy for radiation protection is reflected in the establishment of a Radiation Protection Program that has the specific purpose of maintaining occupational radiation exposures and releases to the environment ALARA. The objectives of the Radiation Protection 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 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 help implement the ALARA goal. Protection of plant personnel requires (a) surveillance of and control over the radiation exposure of personnel, (b) maintaining the exposure of personnel not only within permissible limits, but also within ALARA philosophy and exposure goals, and (c) limiting releases to the environment.

SHINE's administrative personnel exposure limits 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. The preliminary administrative exposure limits are given in Table 11.1-7.

The radiation exposure policy and control measures for personnel are set up in accordance with requirements of 10 CFR 20 and the guidance of applicable NRC regulatory guides. Recommendations from the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) may also be used in the formulation and evolution of the Radiation Protection Program.

The radiation protection program incorporates a corrective action process that is implemented if (1) personnel dose monitoring results or personnel contamination levels exceed the administrative personnel limits; or (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, precluding the recurrence of similar incidents.

11.1.2.1.1 Responsibilities of Key Program Personnel

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

Plant Manager

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

Environment, Safety, and Health Manager

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

Radiation Protection Manager

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

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

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

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

Operations Manager

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

On-site Personnel

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

11.1.2.1.2 Staffing of the Radiation Protection Program

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

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

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

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

11.1.2.1.3 Independence of the Radiation Protection Program

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

11.1.2.1.4 Radiation Safety Committee

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

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

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

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

11.1.2.1.5 Commitment to Written Radiation Protection Procedures

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

Work performed in radiologically restricted areas is performed in accordance with a radiation work permit (RWP). The procedures controlling RWPs are consistent with the guidance provided in Regulatory Guide 8.10. A RWP is required whenever the radiation protection manager determines one is necessary. Activities involving licensed materials not covered by operating procedures and where radioactivity levels are likely to exceed airborne radioactivity limits require the issuance of a RWP. Both routine and non-routine activities are performed under a RWP. The RWP provides a description of the work to be performed. The RWP summarizes the results of recent dose rate surveys, contamination surveys, and airborne radioactivity measurements. The RWP specifies the precautions to be taken by those performing the task. The specified

precautions may include personal protective equipment to be worn while working (e.g., gloves, respirators, personnel monitoring devices), stay-times or dose limits for work in the area, record keeping requirements (e.g., time or dose spent on job) and the attendance of a radiation protection technician during the work. At the minimum, the RWP requires approval by a radiation protection staff member. RWPs have a predetermined period of validity with a specified expiration or termination time.

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.
- Regulatory Guide 8.13 - Instructions Concerning Prenatal Radiation Exposure.
- Regulatory Guide 8.29 - Instructions Concerning Risks from Occupational Radiation.
- ASTM E1168-95 - Radiological Protection Training for Nuclear Facility Workers (ASTM, 2008).

Personnel and visitors entering restricted areas or controlled 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. Alternatively, visitors are provided with trained escorts who have received radiation protection training.

The level of radiation protection training 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 1 millisievert (mSv) (100 millirem [mrem]) 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.
- 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.

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

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

11.1.2.1.7 Radiation Safety Audits

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

11.1.2.1.8 Record Keeping

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

11.1.2.1.9 Technical Specifications

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

11.1.3 ALARA PROGRAM

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

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

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

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

The radiation protection manager 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, (2) audits and inspections, (3) use, maintenance, and surveillance of equipment used for exposure and effluent control, and (4) other issues that may influence the effectiveness of the radiation protection/ ALARA programs. Copies of the report are submitted to the plant manager and the RSC. The RSC monitors the duties of the radiation protection staff. This is to ensure they are specifically involved during the planning and implementing of operation and maintenance activities and the management and disposition of radioactive wastes.

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

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

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

incorporate, as appropriate, new technologies or approaches and operating procedures or changes that could cost-effectively reduce potential radiation exposures.

11.1.3.1 ALARA Program Considerations

The SHINE facility is designed with administrative programs and procedures 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 are applied during the design of SHINE facility. The design is reviewed for ALARA considerations and updated and modified as experience is gained. ALARA reviews include the plant design and layout, considering shielding, ventilation, and monitoring instrument designs as they relate to traffic control, security, access control, and health physics.

Similarly, the location of equipment and routing of pipe containing radioactive fluids is reviewed as part of the design effort. This confirms that equipment and piping expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure of personnel.

Nuclear industry radiation protection practices are incorporated in the design of the SHINE facility. Operating experience is incorporated through NRC inspection and enforcement bulletins, information notices, and other documents. This allows integration of experience and ALARA considerations from plant operators and plant designers and promotes incorporation of recent operating and service experience and lessons learned.

These activities are consistent with the recommendations of Regulatory Guide 8.8.

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.

11.1.3.2 ALARA Program Design Considerations

Design considerations for maintaining personnel exposures ALARA are presented in the following paragraphs. The basic management philosophy guiding the SHINE facility design effort so that radiation exposures are ALARA can be expressed as:

- Design structures, systems and components such that radioactive material, to the greatest extent practical, is remote handled and isolated from on-site personnel by shielded compartments and hot cells.

- Design structures, systems and components for reliability and maintainability, thereby reducing the maintenance requirements on radioactive components.

- Design structures, systems and components to reduce the radiation fields and control streaming, thereby reducing radiation exposure during operation, maintenance, and inspection activities.

- Design structures, systems and components to reduce access, repair and removal times, thereby reducing the time spent in radiation fields during operation, maintenance, and inspection.

- Design 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 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.

- Target solution preparation, recovery, and storage.

- Mo separation, purification, and packaging.

- Radioactive waste handling and disposal.

- 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 target solution and light water pool chemistry conditions, such that corrosion and resulting activation product source terms are minimized.

- Features to allow draining, flushing, and decontaminating equipment and piping.

- Design of equipment to minimize the creation and buildup of radioactive material and to ease flushing of crud traps.

- 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 hatches 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 (for example, the use of seamless piping 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 high quality valves and valve packing, and the direction of leakage via drip pans and piping to sumps and floor drains.

- Provisions for isolating equipment from radioactive process fluids.

- Clean-up systems to limit contamination of the light water pool and cooling systems.

11.1.3.2.3 Facility Layout Design Considerations for ALARA Exposures

Facility 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 to allow maintenance of one component while the other component is in operation.

- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms as necessary.

- Highly radioactive passive components with minimal maintenance requirements are located in completely enclosed shielded cells and are provided with access via a shielded hatch 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.

- Provide adequate lighting and support services (electrical power, compressed air, demineralized water, ventilation, and communications) in radiation areas requiring access.

11.1.4 RADIATION MONITORING AND SURVEYING

11.1.4.1 Radiation Monitoring

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 controlled and restricted areas of the facility. 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. The following design requirements are applied:

- Sample for airborne tritium within each irradiation unit (IU) cell.

- Sample for tritium in the TPS glovebox room.

- Sample for tritium in the facility stack effluent.

This is accomplished using fixed continuous instruments for area monitoring, ventilation duct sampling, and effluent sampling at the facility stack as described below.

d. Continuous Noble Gases, Aerosols, Iodine, and Tritium Effluent Monitoring

The following design requirements apply to the stack release monitor (SRM) on the facility effluent stack.

- Continuous monitoring of radioactive stack releases.

- Generate real time data for control room display and recording.

The SRM provides continuous on line sampling of releases of gaseous effluents from the facility to demonstrate that releases are within the regulatory limits.

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

There are no radioactive liquid effluent discharges from the facility; therefore, there are no liquid effluent monitors.

Continuous monitoring of closed loop process cooling water systems to detect coolant leakage between primary and secondary circuits due to failure in heat exchanger and other system boundaries is provided.

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.

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. Various types of personnel monitoring equipment are used at the facility access points (e.g., "friskers", hand/foot monitors, and portal monitors). Monitoring stations also include methods for small article monitoring for loose contamination to allow for the release of, or assist in the decontamination of, articles that are to be removed from the restricted areas.

- Portal Monitor

  A radiation monitor built like a door frame, inside which a person stands while the monitor scans for radioactive contamination.

- Frisker

  Typically consists of a hand-held probe connected to a count rate meter. Hand held friskers are used to ensure effective control of the spread of contamination.

- Hand and Foot Monitors

  These typically consist of multiple detectors arranged to monitor only hands and feet. Hand and foot monitors are used in applications where "pass-throughs" are frequent and where hand and foot monitoring is the major requirement.

- Small Article Monitoring

  To facilitate the free release of materials, such as deliverable product, packaged radioactive waste, empty shipping containers, etc. swab stations and counters are provided.

Calibration of radiological monitoring equipment is performed in accordance with written procedures and documented prior to the initial use of each monitor. Periodic checks are performed in accordance with written procedures. Calibrations are performed at least annually.

Portal monitors, hand and foot monitors and friskers have the required sensitivity to detect contamination on personnel to ensure that radioactive materials do not spread to the areas outside the restricted areas. Instruments are calibrated in accordance with the National Institute of Standards and Technology or equivalent.

h. Criticality Monitoring

Criticality monitoring in the RPF is provided by the criticality accident and alarm system (CAAS). This system is described in Section 7b.6. In the IF criticality monitoring is provided by the TSV reactivity protection system (TRPS). This system is described in Section 7a2.4.

Radiation monitoring systems, their functions, and their interfaces with the engineered safety features in the facility are described in Section 7a2.7.

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 programs are consistent with the guidance provided in the following references:

- Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring.
- Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data.
- Regulatory Guide 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.
- Regulatory Guide 8.24, Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication.
- Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.

- ANSI N323-1978, Radiation Protection Instrumentation Test and Calibration (ANSI, 1978).

Procedures will include program objectives, sampling protocol, and data analysis methods. Equipment selection is based on the type of radiation being monitored. Procedures are prepared for each of the instruments used and specify the frequency and method of calibration. Maintenance and calibration are in accordance with applicable standards and manufacturers' recommendations.

The survey program procedures also specify the frequency of measurements and record keeping and reporting requirements.

11.1.4.3 Technical Specifications

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

11.1.5 RADIATION EXPOSURE CONTROL AND DOSIMETRY

11.1.5.1 Controlled Area

The NRC in 10 CFR 20 defines a controlled area as an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason. The area of the SHINE facility within the facility physical structure beyond the main reception area but outside any restricted area is part of the controlled area. Due to the presence of administrative and physical barriers, members of the public do not have direct access to this controlled area of the facility and must be processed by security and authorized to enter the facility. Training for access to a controlled area is provided commensurate with the radiological hazard.

Facility visitors include delivery people, tour guests, and service personnel who are temporary, transient occupants of the controlled area. Area monitoring demonstrates compliance with public exposure limits for such visitors. Individuals who are contractor or SHINE employees and who work only in the controlled area are subject to the exposure limits for members of the public.

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 zones 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 of restricted areas 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 SHINE site is an unrestricted area. This area can be accessed by members of the public or by facility personnel. The unrestricted area is governed by the limits in 10 CFR 20.1301. The total effective dose equivalent to individual members of the public from the licensed operation may not exceed 1 mSv (100 mrem) in a year (exclusive of background radiation). The dose in any unrestricted area from external sources may not exceed 0.02 mSv (2 mrem) in any one hour.

b. Restricted area

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

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

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

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

- An area in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.05 mSv (5 mrem) in 1 hour (hr) at 30 cm (11.8 in.) from the radiation source or from any surface that the radiation penetrates is designated a "radiation area" as defined in 10 CFR 20.1003.

- An "airborne radioactivity area" means a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations (1) In excess of the derived air concentrations (DACs) specified in Appendix B, to 10 CFR 20.1001 - 20.2401, or (2) To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the annual limit on intake or 12 DAC-hours. Note that entry into this area does not automatically require the wearing of a respirator.

- A "high radiation area" is an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in 1 hr at 30 cm (11.8 in.) from the radiation source or from any surface that the radiation penetrates. No areas of this type are accessible to individuals during routine operation of the facility. Such areas are radiologically shielded and isolated from access to individuals by the use of engineered physical barriers. These include structural shield blocks, and/or locked shield doors.

- A "contaminated area" is an area which SHINE defines as an area where removable contamination levels are above 0.33 Becquerel per 100 square centimeters (Bq/100 cm^2) (20 disintegrations per minute per 100 square centimeters [dpm/100 cm^2]) of alpha activity or 16.7 Bq/100 cm^2 (1,000 dpm/100 cm^2 beta/gamma activity).

- Areas of “caution” exist within the restricted area. For instance, the NRC limits the soluble uranium intake of an individual to 10 milligrams in a week in consideration of chemical toxicity.

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.

Because there are high radiation areas in the facility, access to those areas is physically prevented due to radiation level. Access control is by a combination of administrative methods and active as well as passive engineered safeguards.

Personnel who have not been trained in radiation protection procedures are not allowed access to the restricted area without escort by other trained personnel.

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

Special radiological zones within the restricted areas (e.g., airborne radioactivity area) are clearly identified by physical means such as placarding or boundary marking is done in accordance with 10 CFR 20.1902. The radiation and contamination levels from the most recent survey are clearly noted on each posting.

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 protective clothing. 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 protective clothing. Areas requiring protective clothing are posted at each of their entry points. Radiation protection management and associated technical staff are responsible for determining the need for protective clothing in each work area.

11.1.5.5 Personnel Monitoring for External Exposures

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

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

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

Anytime an administrative limit is exceeded, the radiation protection manager 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.

11.1.5.6 Determination of Internal Exposures

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

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

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

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

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

3. Separately assess the contribution of fractional intakes of Class D, W, or Y compounds of a given radionuclide (see 10 CFR 20, Appendix B) to the committed effective dose equivalent.

d. If SHINE chooses to assess intakes of Class Y material using the measurements given in 10 CFR 20.1204(a)(2) or (3), SHINE may delay the recording and reporting of the assessments for periods up to 7 months, unless otherwise required by 10 CFR 20.2202 or 20.2203, in order to permit SHINE to make additional measurements basic to the assessments.

e. If the identity and concentration of each radionuclide in a mixture are known, the fraction of the DAC applicable to the mixture for use in calculating DAC hours must be either of the following:

1. The sum of the ratios of the concentration to the appropriate DAC value (e.g., D, W, Y) from 10 CFR 20, Appendix B for each radionuclide in the mixture.
2. The ratio of the total concentration for radionuclides in the mixture to the most restrictive DAC value for any radionuclide in the mixture.

f. If the identity of each radionuclide in a mixture is known, but the concentration of one or more of the radionuclides in the mixture is not known, the DAC for the mixture must be the most restrictive DAC of any radionuclide in the mixture.

g. When a mixture of radionuclides in air exists, SHINE may disregard certain radionuclides in the mixture if the following are true:

1. SHINE uses the total activity of the mixture in demonstrating compliance with the dose limits in 10 CFR 20.1201 and in complying with the monitoring requirements in 10 CFR 20.1502(b).
2. The concentration of any radionuclide disregarded is less than 10 percent of its DAC.
3. The sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent.

h. ALI assumptions

1. In order to calculate the committed effective dose equivalent, SHINE may assume that the inhalation of one ALI, or an exposure of 2,000 DAC-hours, results in a committed effective dose equivalent of 5 rems (0.05 Sv) for radionuclides that have their ALIs or DACs based on the committed effective dose equivalent.
2. When the ALI (and the associated DAC) is determined by the nonstochastic organ dose limit of 50 rems (0.5 Sv), the intake of radionuclides that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) (the stochastic ALI) is listed in parentheses in Table 1 of 10 CFR 20, Appendix B. In this case, SHINE may, as a simplifying assumption, use the stochastic ALIs to determine committed effective dose equivalent. However, if SHINE uses the stochastic ALIs, SHINE must also demonstrate that the limit in 10 CFR 20.1201(a)(1)(ii) is met.

11.1.5.7 Evaluation of Doses

Dose evaluations may be performed at more frequent intervals and should be performed when reasonable suspicion exists regarding an abnormal exposure. The internal and external exposure values are summed in accordance with 10 CFR 20.

Procedures for the evaluation and summation of doses are based on the guidance contained in Regulatory Guides 8.7 and 8.34.

11.1.5.8 Locker and Change Rooms

Locker rooms for men and women are provided for personnel to change into appropriate work clothing and store personal belongings. The following facilities are provided for in the locker room area:

- Restrooms - restrooms for men and women are provided. These rooms are not for personnel decontamination.
- First aid station - a first aid station is provided to treat injured personnel.
- Personnel decontamination area - a personnel decontamination area is provided to handle cases of accidental radioactive contamination. A hand washing sink and a shower are provided for contamination removal.
- Information area - an information area is provided to notify personnel of information important to radiation protection.

11.1.5.9 Storage Areas

Storage areas are provided for the following items:

- Protective (i.e., anti-contamination) clothing.
- Respiratory protection equipment.
- Radiation protection supplies.

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.

General equipment and facility layout design considerations to prevent the spread of contamination to the facility and 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, for example the neutron driver assembly and the subcritical assembly are located in sealed shielded compartments (access is provided via shielded hatches as required).

Where operator intervention is required during processing activities, for example Mo 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 with sumps and hard piped drains (with leak detection) to facilitate confinement, isolation, and collection of potential liquid spills to minimize the spread of contamination to the facility and the environment. Additionally, these shielded compartments and shielded hot cells are provided with closed ventilation systems which are operated at negative pressures with respect to the surrounding environment.

11.1.6.2 Monitoring and Controlled Entry and Egress to Restricted Area

Potentially radioactive components, piping, and materials are located within the hardened structure of the restricted area. Access to and egress from this area is strictly controlled via administrative procedures (i.e., radiation work permits) and passive confinement structure design.

Personnel access and egress is controlled by radiation protection department 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.

Material removed from the restricted area (for example, production material, tools, disposed equipment, various process, and maintenance consumables) are appropriately packaged in pre-approved containers, inventoried, and monitored prior to release. Criteria for classifying contaminated materials and equipment will be provided in the FSAR.

The restricted area is 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.

Health physics personnel routinely perform radiation and contamination assessments of accessible areas in the restricted area. Special surveys are performed, prior to entry, if access is required to normally unoccupied areas.

11.1.6.3 Piping

The use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA.

To the extent possible, radioactive piping is located inside shielded compartments or hot cells. For transfers between hot cells the piping is located in shielded pipe chases which provide for liquid and airborne confinement and detection of leakage. Cover blocks are provided to allow for visual inspection of piping.

11.1.6.4 Light Water Pool

The light water pools which provide shielding and cooling to the subcritical assemblies is designed to eliminate unidentified leakage to the facility and the environment.

11.1.6.5 Process Tanks

Process tanks are designed, fabricated and tested in accordance with national codes and standards. The tanks are seismically supported and are located in seismically designed concrete vaults that are designed to eliminate unidentified leakage to the facility and the environment.

11.1.7 ENVIRONMENTAL MONITORING

11.1.7.1 Environmental Monitoring Program

The requirement to have a radiological environmental monitoring program is documented in 10 CFR 20.1302. The radiological environmental monitoring program is used to verify the effectiveness of plant 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. Methods for establishing and conducting environmental monitoring are provided in Regulatory Guide 4.1. Regulatory Guide 4.1 refers to NUREG-1301 for detailed guidance for conducting effluent and environmental monitoring.

SHINE is committed to implementing a radiological environmental monitoring program for the SHINE facility. Regulatory Guide 4.1 and NUREG-1301 are written for nuclear power plants, but, due to the similarities between airborne releases of radioactivity from nuclear power plants and those released from the SHINE facility, guidance provided in Regulatory Guide 4.1 and NUREG-1301 was considered when developing the radiological environmental monitoring plan (REMP) for the SHINE facility. In addition to the guidance provided in Regulatory Guide 4.1 and NUREG-1301, 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

Effluent releases from the SHINE facility are limited to the airborne pathway. Airborne effluents include noble gases, iodine and other halogens, particulates, tritium, and Ar-41. The DQO process indicates the following radiation exposure pathways represent plausible public exposure scenarios.

- Direct radiation exposure pathway monitored using thermoluminescent dosimeters (TLDs).
- Airborne exposure pathway monitored using continuous air samples.
- Ingestion exposure pathway.

The SHINE facility is a zero process liquid effluent release plant due to extensive reuse of process liquids. Although there are liquid discharges from the SHINE facility (e.g., sanitary waste and storm water runoff), there are no liquid effluent discharges from the RCA, and thus there is no plausible exposure route to members of the public through the liquid effluent pathway and no liquid effluent monitoring is conducted. However, groundwater monitoring is conducted as part of the Community Environmental Monitoring Program (CEMP). Refer to Subsection 11.1.7.3 for a discussion of the CEMP and groundwater monitoring.

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 TLDs. The TLDs quantify direct radiation from radiation sources contained within the SHINE facility building, from sources contained within the waste staging and shipping facility, from radioactivity in the airborne effluent, and from deposition of airborne radioactivity onto the ground.

A description of TLD locations and the rationale for TLD locations are provided in Table 11.1-8. TLD locations are shown on Figure 11.1-3. Table 3.12-1 of NUREG-1301 recommends 40 TLD locations, i.e., an inner ring and an outer ring of TLDs with one TLD in each ring at each of the 16 meteorological sectors and the balance of TLDs to be located at special interest areas. At least one TLD 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, a minimum number of TLD locations (i.e., nine) are specified. These are located in order to provide annual direct dose information at on-site locations which are expected to have significant occupancy and at property locations in the north, south, east, and west directions (to ensure all directions are monitored, including the direction of the theoretical maximum exposed individual [MEI] and the direction of the nearest occupied structure). In addition, at least one location includes a paired TLD so that data quality can be determined.

Background radiation is subtracted from the TLD results. The background radiation values are those established during the baseline environmental survey which obtained baseline TLD readings at each TLD location. While a control (background) TLD is deployed as part of the REMP, the control TLD is not used for background subtraction. This is due to the high variance (relative to the dose constraint) in background radiation caused by difference in terrestrial radiation, cosmic radiation, and seasonal variations. As such, the baseline survey ensures these sources of variation are accounted for by collecting data at the specific location that is used to monitor direct radiation under the REMP during the operational period.

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 should they occur. These types of releases could result in exposure via inhalation, direct radiation (cloud immersion) and ingestion. Airborne effluent streams from the SHINE facility that have the potential to include radioactive iodine are treated (e.g., using silver-impregnated zeolite and/or carbon filters) to remove the iodine. Some particulate activity (other than iodine) and tritium could also be released in airborne effluents; however, most of the off-site exposure due to airborne effluent releases is associated with noble gas and radioactive iodine releases.

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 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 possibly airborne particulate radioactivity releases, the radiological environmental monitoring program includes airborne sampling.

11.1.7.2.2.1 Air Sampling Locations

The guidance provided in Table 3.12-1 of NUREG-1301 is 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. The CAMs that are used to obtain continuous air samples include a radioiodine canister for weekly iodine-131 (I-131) analysis and a particulate sampler which is analyzed for gross beta radioactivity and for quarterly isotopic analysis.

Four CAM locations are near the facility property line in the north, south, east, and west direction sectors 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. There is also a CAM located a sufficient distance from the SHINE facility to provide background information for airborne activity. Table 3.12-1 of NUREG-1301 suggests an additional CAM location in the vicinity of a community having the highest calculated annual average ground-level D/Q. This CAM requirement is combined with the CAM located at the site boundary in the north direction (refer to Table 11.1-8). A description of air sample locations and the rationale for air sample locations are provided in Table 11.1-8. CAM locations are shown on Figure 11.1-3.

11.1.7.2.3 Ingestion Pathway (Biota Monitoring)

NUREG-1301 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. Since the SHINE source term is more modest than that of a nuclear power plant and particulate and iodine radionuclides are not normally expected to be present in significant quantities within airborne effluent releases from the SHINE facility, biota monitoring is not routinely included in the REMP. Monitoring of the milk pathway will be performed as part of the CEMP, as described in Subsection 11.1.7.3

In the event that the results of environmental airborne samples, effluent monitor sample results, or milk sampling results indicate iodine or particulates in quantities large enough to result in a calculated dose greater than that predicted for normal releases (e.g., from GENII models used to show compliance with the 10 CFR 20.1101(d) dose constraint), then a more comprehensive sampling campaign is undertaken. The sampling campaign is planned under the DQO process thus ensuring the appropriate types and numbers of samples are collected to best represent potential public doses based on the radionuclides of concern in the environmental airborne, effluent monitor, or milk samples.

11.1.7.3 Community Environmental Monitoring Program

In addition to the monitoring that is performed to meet regulatory requirements, SHINE has a CEMP. The CEMP initially includes groundwater monitoring and monitoring of the milk pathway. Additional initiatives may be undertaken in the future.

Milk is one of the most important foods contributing to the radiation dose to people if milk animals are pastured in an area near a facility that releases radioactive material. Dairy production takes place approximately one-half mile (mi.) (0.8 kilometers [km]) to the east of the SHINE facility and goat production takes place at approximately 0.69 mi. (1.1 km) northeast of the facility. A description of the milk sampling program will be provided with the FSAR.

11.1.7.3.1 Groundwater Monitoring

There is no liquid effluent release pathway from the RCA associated with the SHINE facility and thus, 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 radiological environmental monitoring plan. 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 radiological environmental monitoring plan.

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 facility building. Although there are no defined liquid effluent release pathways from the RCA 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, 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 11.1-8.

11.1.7.3.2 Other Potential Special Sampling Initiatives

After SHINE operations are underway, additional sampling initiatives may be undertaken. For example, detectors may be placed within areas of community interest to allow for real-time gamma monitoring that can be observed via the internet. Other initiatives may include collection of high-volume air samples using portable air samplers in areas of community interest.

11.1.7.4 Preoperational Baseline Monitoring

As previously indicated, effluent releases from the SHINE facility are limited to releases via the airborne pathway. Environmental monitoring of the SHINE facility includes the use of TLDs for monitoring direct radiation and CAMs for detecting iodine and particulate activities in airborne effluents. A preoperational baseline survey is performed to obtain TLD readings at the nine TLD locations and to obtain air sample radioactive iodine and particulate surveys at the five air

sample locations described in Table 11.1-8. The preoperational baseline TLD readings and the preoperational baseline air sample survey results represent background radiation values that are used with operational surveys to establish the radiological impact of the SHINE facility on the environment.

Since groundwater is sampled via test wells to the west and south of the SHINE facility building as part of the CEMP and since the test wells that are north and east of the SHINE facility building could be sampled in the future, groundwater sampling of the four test wells is included in the preoperational baseline survey. Since milk sampling will be performed as part of the CEMP, milk sampling will be included in the preoperational baseline survey. Additional biota sampling (soil, broad leafed plants, and meat) will only be conducted if there are significant quantities of iodine or particulates in other sample results. Since there is a possibility that complete biota sampling could be performed at some future date, biota sampling is included in the preoperational baseline survey.

11.1.7.5 Environmental Monitoring Program Procedures

Environmental surveys conducted in support of the REMP are performed in accordance with written plans documented in facility procedures. Changes to the REMP or to environmental survey plans are reviewed for adequacy and approved prior to implementation in accordance with facility procedures.

11.1.7.6 REMP Reports

A radioactivity effluent discharge summary report, i.e., a “Radioactive Effluent Release Report” and a radiological environmental surveillance program report, i.e., a “Radiological Environmental Operating Report” is provided to the NRC on an annual basis representing a one year monitoring period. The one year monitoring period is in accordance with Section C, Staff Regulatory Guidance, of Regulatory Guide 4.20.

As required by the 1979 NRC Branch Technical Position (included as Appendix A to NUREG-1301), a laboratory inter-comparison program is established to crosscheck sample analysis results. The results of the inter-comparison crosscheck sample analysis are included in the annual Radiological Environmental Operating Report.

Although biota monitoring is not planned, an annual land use census is conducted during the growing season. The location of the nearest milk animal is determined during the annual land use census (as required by the 1979 NRC Branch Technical Position which is included as Appendix A to NUREG-1301). The results of the annual land use census are included in the annual Radiological Environmental Operating Report and are available if it is determined that a biota monitoring sampling campaign is to be undertaken.

Table 11.1-1 Parameters Applicable to TSV Source Term

Parameter	Nominal Values	Limiting Values	Bounding Values
19.75 weight percent (wt.%) enrichment	19.75 wt.%	19.75 wt.%	19.75 wt.%
Uranium-234 (U-234), uranium-236 (U-236) concentrations	Nominal	Max	Max
Power	[redacted]	[redacted]	[redacted]
Irradiation Time	5.5 days	5.5 days	5.5 days
TSV Dump Tank Decay Time	[redacted]	[redacted]	[redacted]
Mo Extraction %	[redacted]	[redacted]	[redacted]
[redacted]	90%	~ 90%	~ 80%
[redacted]	99%	> 97%	~ 90%
Supercell Extraction Time	[redacted]	[redacted]	[redacted]]
Chem Remix + 1/M Time	[redacted]	[redacted]	[redacted]
Number of Cycles	[redacted]	[redacted]	[redacted]

Table 11.1-2 Limiting versus Bounding Radionuclide Inventories in Target Solution

	Actinide Activity (Ci)			Fission Product Activity (Ci)		
Case	At Shutdown	[redacted]	Post Extraction	At Shutdown	[redacted]	Post Extraction
Limiting Values[a]	[redacted]					
Bounding Values[a]	[redacted]					
% Difference	0.16%	0.25%	0.43%	0.15%	0.68%	4.47%

a) Extraction time = [redacted] minutes (limiting case), [redacted] minutes (bounding case)

Table 11.1-3 Irradiated Target Solution Activity for Select Radionuclides [redacted] Following Shutdown
(Sheet 1 of 2)

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

Table 11.1-3 Irradiated Target Solution Activity for Select Radionuclides [redacted] Following Shutdown (Sheet 2 of 2)

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

Table 11.1-4 Airborne Radioactive Sources

System	Component	Location	Major Sources	Estimated Maximum Activity, Ci	Exterior Dose Rate mRem/hr
TPS	Tritium Purification System	TPS Glovebox	H-3	(c)	<0.25 (e)
Neutron driver assembly system (NDAS)	Driver Vacuum Hardware	IU Cell	H-3	(c)	<0.25 (e)
TOGS	Off-Gas Piping, Zeolite Beds	TOGS Shielded Cell	I, Kr, Xe[a]	(d)	<0.25 (e)
NGRS	Noble Gas Decay Tanks	Noble Gas Storage Cell	Kr, Xe[b]	6.00E+04	2.64E-01 30 cm from wall
RVZ1	IU Cell Atmosphere	IU Cell	Ar-41	(Facility Production rate: 15.2 Ci/yr)	N/A
PVVS	PVVS Piping	Pipe Trenches	I, Kr, Xe	To be provided in the FSAR	To be provided in the FSAR

a) The tracked iodine and noble gas isotopes are: I-131, I-132, I-133, I-134, I-135, I-136, Kr-83m, Kr-85, Kr-85m, Kr-87, Kr-88, Kr-89, Kr-90, Kr-91, Kr-92, Kr-93, Xe-131m, Xe-133, Xe-133m, Xe-135, Xe-135m, Xe-137, Xe-138, Xe-139, Xe-140, Xe-141, and Xe-142.

b) The tracked noble gas isotopes are: Kr-83m, Kr-85, Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133, Xe-133m, Xe-135, Xe-135m, and Xe-138.

c) The neutron driver and tritium purification system are designed such that the total dose from gaseous releases, including tritium, is kept below the limit in 10 CFR 20.1101(d). Values to be provided in the FSAR.

d) Values to be provided in the FSAR.

e) For normally-occupied areas.

Table 11.1-5 Liquid Radioactive Sources

System[(b)]	Component[(b)]	Location[(b)]	Major Sources	Estimated Maximum Activity, Ci	Exterior Dose Rate mrem/hr
Subcritical Assembly System (SCAS)	Neutron Driver and TSV (operating)	IU cell	U-235 Fission (Neutrons and Photons)	[redacted]	0.6
SCAS	TSV, TSV Dump Tank (shutdown)	IU cell	(see Table 11.1-3)	[redacted]	0.06
Primary Closed Loop Cooling System (PCLS)	Primary Coolant Pump and Piping	Primary coolant pump cell, pipe trench	N-16	(a)	<0.25 (c)
Molybdenum Extraction and Purification System (MEPS)	Mo Extraction Column	Supercell	(see Table 11.1-3)	[redacted]	<0.25 (c)
MEPS	Mo-99 Product	Supercell	Mo-99, Tc-99	[redacted]	<0.25 (c)
Uranyl Nitrate Conversion System (UNCS)	Uranyl Nitrate Conversion Tank	UREX hot cell	(see Table 11.1-3)	[redacted]	<0.25 (c)
UNCS	Recycle Target Solution Tank	Tank Vault	(see Table 11.1-3)	[redacted]	<0.25 (c)
UNCS	Raffinate Hold Tank	UREX hot cell	(see Table 11.1-3)	[redacted]	<0.25 (c)
UNCS	Recycle Uranyl Nitrate Hold Tank	Tank Vault	U-234, U-237, U-239	[redacted]	<0.25 (c)
Radioactive Liquid Waste Storage System (RLWS)	Liquid Waste Storage Tank	Tank vault	(see Table 11.1-3)	(a)	<0.25 (c)
Radioactive Liquid Waste Evaporation and Immobilization System (RLWE)	Liquid Waste Collection Tank	Waste evaporation hot cell	(see Table 11.1-3)	(a)	<0.25 (c)

a) Values to be provided in the FSAR.
b) Physical and chemical properties of process solutions, special nuclear material inventories, and descriptions of the systems can be found in Chapter 4.
c) For normally-occupied areas.

Table 11.1-6 Solid Radioactive Sources

System[(b)]	Component[(b)]	Location	Major Sources
NDAS, TOGS	Neutron Driver, TOGS Components	IU cell	Activation Products, I
SCAS	Neutron Multiplier, SASS	IU cell	[redacted]
MEPS	Spent Extraction Columns	Supercell	[redacted](a)
Target Solution Preparation System (TSPS)	Fresh Uranium Metal and Uranium Oxide	Uranyl sulfate preparation area	U-234, U-235, U-238
RLWE	Solidified Waste Drums	Liquid waste solidification cell	(see Table 11.1-3)
Solid Radwaste	Spent Filters, Spent Extraction Columns	Solid waste hot cell	[redacted](a)
SCAS	Subcritical Multiplication Source	IU cell	Alpha-neutron Startup Source

a) The significant isotopes include [redacted]

b) Descriptions of the systems and their physical characteristics can be found in Chapter 4.

Table 11.1-7 Administrative Radiation Exposure Limits

Type of Dose	10 CFR Part 20 Limit (rem/year)	SHINE Annual Administrative Limit (rem/year)
Adult Radiological Worker		
The more limiting of:		
Total effective dose equivalent to whole body, or	5	0.5
Sum of deep-dose equivalent and committed dose equivalent to any organ or tissue other than lens of eye	50	5
Eye dose equivalent to lens of eye	15	1.5
Shallow-dose equivalent to skin of the whole body or any extremity	50	5
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 11.1-8 Environmental Monitoring Locations
(Sheet 1 of 2)

Monitoring Type	Location	Rationale
Groundwater Sampling Locations		
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.
TLD Locations (At least one TLD location includes a paired TLD for data quality determination)		
TLD #1	Off-site Location	Distance is sufficiently large such that there is no significant dose rate associated with SHINE activities or associated with airborne effluents.
TLD #2	Southeast corner of administration building	Administration building is an on-site area with regular occupancy outside the SHINE facility. The southeast corner of the building is closest to the SHINE facility.
TLD #3	North side of the support facility building	The support facility building is an on-site area with regular occupancy outside the SHINE facility. The north side of the support facility building is closest to the SHINE facility.
TLD #4	Operating area boundary fence directly east of the waste staging and shipping building	TLD is positioned to detect direct radiation from the waste staging and shipping building.
TLD #5	Security station	The security station is normally occupied.
TLD #6	Property line to the east of the SHINE facility vent stack	This location is in the direction of dairy production and the horse pasture. Also the prevailing wind is from the west as indicated by the annual wind rose so this is the location of the MEI.

Table 11.1-8 Environmental Monitoring Locations
(Sheet 2 of 2)

Monitoring Type	Location	Rationale
TLD #7	Property line to the west of the SHINE facility	This location ensures all directions are monitored.
TLD #8	Property line to the north of the SHINE facility vent stack	This location is in the direction of Janesville.
TLD #9	Property line to the south of the SHINE facility vent stack	This location is in the direction of the nearest occupied structure.
Air Sampler Locations		
Air Sampler (CAM #1)	Off-site location	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 (CAM #2)	Close to property line, directly north of the SHINE facility vent stack	This direction has high 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 (CAM #3)	Close to property line, East of the SHINE facility vent stack	This direction has high D/Q and is in the direction of dairy production and the horse pasture.
Air Sampler (CAM #4)	Close to property line, west of the SHINE facility	This location ensures all directions are monitored.
Air Sampler (CAM #5)	Close to property line, South of the SHINE facility vent stack	This location is in the direction of the nearest occupied structure.

Table 11.1-9 TSV Noble Gas and Iodine Production Rates and Annual Releases at the Site Boundary After 960 Hours of NGRS Holdup

Radionuclide	Half Life	Production Rate (Ci/sec)	Annual Release (Ci)
Kr-83m	1.86 hours	[redacted]	
Kr-85	10.76 years	[redacted]	1.2E+02
Kr-85m	4.48 hours	[redacted]	
Kr-87	1.27 hours	[redacted]	
Kr-88	2.84 hours	[redacted]	
Kr-89	3.15 min	[redacted]	
I-131	8.02 days	[redacted]	9.5E-01
I-132	2.28 hours	[redacted]	
I-133	20.80 hours	[redacted]	
I-134	52.60 min	[redacted]	
I-135	6.57 hours	[redacted]	
Xe-131m	11.90 days	[redacted]	5.5E+02
Xe-133	5.24 days	[redacted]	1.1E+04
Xe-133m	2.19 days	[redacted]	
Xe-135	9.10 hours	[redacted]	
Xe-135m	15.30 min	[redacted]	
Xe-137	3.82 min	[redacted]	
Xe-138	14.10 min	[redacted]	

11.2 RADIOACTIVE WASTE MANAGEMENT

SHINE produces medical isotopes by the fission of low-enriched uranium (LEU) 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 the identification, classification, control, processing (as required), and packaging for transport and disposal, 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. 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 waste program is coordinated with the radiation protection program and under the Plant Manager. The goal of the waste management program is to minimize waste generation, minimize exposure of personnel, and to protect the general public and environment. An official charter describing the authority, duties, and responsibilities of personnel in the waste management organization will be developed for the FSAR.

11.2.1.1 Plant Management Responsibilities

- Accepts overall responsibility for safety of waste management operations.
- Assigns responsibility and delegates commensurate authority to implement established waste policy.
- Formulates radioactive waste management policy.
- Provides a waste management staff appropriate to the scope of operations and experienced in waste management operations.
- Ensures that the waste management self-assessment program is implemented.

11.2.1.2 Radiation Protection Management Responsibilities

- Perform the required radiation monitoring surveys to support waste transfers and shipments.
- Support the preparation of waste shipping documentation.
- Participate in audits of the waste management program.

11.2.1.3 Waste Management/Operations Management Responsibilities

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

- Conducts self-assessments of waste management practices and compliance with procedures in accordance with the waste management self-assessment program.

11.2.1.4 Training Management Responsibilities

- Develops waste management training and qualification program.
- Provides training to personnel and visitors commensurate with the radiological waste hazard to which they may be exposed.
- Provides annual requalification of previously-trained waste management personnel. Includes training on procedure changes and changes in required skills.
- Evaluates the waste management and qualification training program at least annually. Reviews program content to ensure it remains current and adequate to ensure worker safety.

11.2.1.5 Supervisors

- Accepts responsibility for the safety of operations under their control.
- Ensures personnel under their supervision are trained and qualified, as applicable, commensurate with responsibilities performed.
- Provides input to the development of written procedures applicable to waste management operations.

11.2.1.6 Operating Procedures

- Are developed and periodically reviewed in accordance with the SHINE procedure program.
- Facilitate the efficient and safe conduct of the operation. Procedures are organized and presented for convenient use by operators.
- Include those controls and limits significant to the waste management of the operation.

11.2.1.7 Record Keeping and Document Controls

SHINE maintains records of the radioactive waste management program, including:

a. The provisions of the program.
b. Audits and other reviews of program content and implementation.
c. SHINE retains the records required in accordance with SHINE's record keeping and document control programs.

11.2.1.8 Waste Management Audits

Facility radioactive waste management audits are conducted, at a minimum, on an annual basis for the purpose of reviewing the functional and safety elements of the waste management program. The audits also evaluate programmatic efforts to minimize production of radioactive wastes. The audit activity is led by waste management and the results are sent to Executive Management. Any deficiencies of the audit are addressed by the corrective action process.

11.2.1.9 Technical Specifications

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

11.2.2 RADIOACTIVE WASTE CONTROLS

The wastes generated by the SHINE facility are not spent nuclear fuel, high-level waste, or byproduct material. Therefore, the radioactive wastes generated by the SHINE facility are all classified as low level waste (LLW). The LLW generated by the SHINE facility may be sub-classified as Class A, Class B, Class C, or Greater-than-Class C (GTCC).

- For the purposes of transportation, packaged wastes may be categorized as low specific activity (LSA), or as requiring Type A packaging, or as 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 waste acceptance criteria for the intended disposal site.
- ALARA goals and implementation are detailed in Subsection 11.1.3.
- Radioactive waste management operating procedures are discussed in Subsection 11.2.1.6. These procedures, which ensure proper identification, characterization, and separate treatment of radioactive wastes will be described in the FSAR.

11.2.2.1 Radioactive Waste Minimization

The SHINE facility will have a program for pollution prevention and waste minimization. The radioactive waste management program is a subset of this program and will include specific considerations for the minimization of radioactive waste. Key features of the pollution prevention and waste minimization program include:

a. Incorporation of radioactive waste minimization design features.
b. Employee training and education on general environmental activities and hazards regarding the facility, operations, and the pollution prevention program, as well as waste minimization requirements, goals, and accomplishments.
c. Responsibilities for pollution prevention and waste minimization.
d. Recognition of employees for efforts to improve environmental conditions.
e. Requirements for employees to consider pollution prevention and waste minimization in day-to-day activities and engineering.

11.2.2.2 Waste Stream Sources

Waste management operations occur in the production facility and waste staging and shipping buildings. Specific waste processes are identified in the general arrangement drawing (Figure 1.3-2). Equipment and associated features for volume reduction, containment, and/or packaging, storage, and disposal of solid, liquid, and gaseous radioactive waste are provided in Subsections 9b.7.3, 9b.7.4, and 9b.7.5.

A number of waste streams have been identified. Table 11.2-1 summarizes the identified waste streams, characteristics, generation rates, and shipment categories.

11.2.2.2.1 Target Solution Preparation and Adjustment

Waste generated by this process includes used cans in which new uranium metal is received, PPE, and spent filters. Wastes from this process are Class A waste and are consolidated and shipped under an optimized shipping plan.

11.2.2.2.2 Irradiation Unit

ORIGEN and Monte Carlo N-Particle (MCNP) runs were made to determine the specific level of activity after one year of service and also to determine external dose rates. Based on that analysis, the irradiation unit is expected to be Class A waste and appropriate for LSA shipping.

11.2.2.2.3 TSV Off-Gas System

The TOGS (Class A and GTCC waste) is designed to collect and treat the gas produced by radiolysis during irradiation of the target solution. The irradiation process produces hydrogen and oxygen, as well as small mass quantities of krypton (Kr), Xe, and iodine. Wastes generated by this process include spent iodine removal beds and scrap equipment.

11.2.2.2.4 Mo-99 Extraction

Waste (Class A) generated by this process includes the spent adsorption columns, spent wash solution, [concentration columns], and rotary evaporator condensate. These liquid waste streams are consolidated with other liquid waste streams in radioactive liquid waste tanks and treated for disposal.

11.2.2.2.5 Mo-99 Purification

The glassware used in this process is replaced 200 to 400 times per year. It is not anticipated this waste stream (Class A) contains significant quantities of long-lived radionuclides.

11.2.2.2.6 Target Solution Clean-up

Waste generated by the target solution clean-up process includes the following.

[redacted]

This waste stream (Class B) is kept separate from the other liquid waste streams in order to meet the overall objective of generating the smallest volume of waste greater than Class A. [redacted] this waste stream is projected to be Class B waste. Calculations assuming either [redacted] also indicated [redacted] leading to this waste stream being greater than Class A waste.

UREX Raffinate

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

Thermal Denitration Evaporator Condensate

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

Spent Solvent Replacement

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

Spent Resin Column

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

11.2.2.2.7 Process Vessel Vent System

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

11.2.2.2.8 Decontamination Waste

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

11.2.2.2.9 Coolant Clean-up Systems

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

11.2.2.2.10 Radioactive Liquid Waste Processing

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

11.2.2.2.11 Noble Gas Removal System

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

11.2.2.3 Technical Specifications

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

11.2.3 RELEASE OF RADIOACTIVE WASTE

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

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

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

11.2.3.1 Solid Wastes

The subsections below discuss processing needs and methodology for the eventual release of major solid wastes generated by the SHINE facility. The tables referred to in this section provide functional requirements by major waste stream. Many requirements are driven by the receiving facility's waste acceptance requirements. The tables are based on requirements from Energy*Solutions*' waste acceptance criteria (WAC), the facility currently projected to receive the waste streams discussed.

11.2.3.1.1 Irradiation Units

There are two sub-streams associated with the irradiation units: components meeting the "standard debris" Energy*Solutions* WAC definition and components meeting the "oversize" definition. Requirements differ for each of these waste streams in that the first category (standard debris) includes materials that are less than 10 inches (25.4 centimeters) in at least one dimension and no longer than 12 feet (3.66 meters) in any dimension. Debris that does not meet this size criterion is categorized as oversize debris. These waste streams and their disposition methodologies are presented in Table 11.2-2.

11.2.3.1.2 Process Extraction Columns

Extraction columns are held in the hot cell for two weeks, then moved to vaults for further decay to meet shipping requirements. Requirements for this waste stream are presented in Table 11.2-3.

11.2.3.1.3 Process Glassware

Glassware is discarded. Glassware is crushed in the solid waste processing cell. Requirements for this waste stream are presented in Table 11.2-4.

11.2.3.1.4 Iodine Removal Beds

The iodine removal beds are a component of the TOGS. The toxicity characteristic leaching procedure (TCLP) may or may not result in the classification of this waste as RCRA waste; however, testing of untreated silver mordenite at Hanford indicated the material exceeds TCLP limits prior to solidification (WTP, 2002).The waste is also radioactive and as such may be a mixed low level waste (MLLW). H-3, I, Xe, and Kr enter these beds. Only iodine is trapped. The waste classification for this material is a function of both the efficiency of the iodine removal beds and the change out frequency of the beds. The current design goal is for the beds to last the lifetime of the facility. If that goal is met, the beds may collect enough I-129 to be classified as GTCC waste.

11.2.3.1.5 Recombiner Beds, Demister and Component Replacement

This waste stream is associated with the TOGS. This waste stream is produced very infrequently, if at all, as the system components are being designed for the lifetime of the facility.

11.2.3.2 Liquid Waste Streams

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

11.2.3.2.1 Consolidated Liquids

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

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

11.2.3.2.2 [redacted]

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

11.2.3.2.3 [Ion Exchange Resins]

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

11.2.3.3 Gaseous Waste Streams

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

Table 11.2-1 Waste Stream Summary

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

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

Table 11.2-2 Waste Methodology for Irradiation Unit

Requirement	Basis
Disassemble irradiation unit (separate accelerator from H-3 chamber and drift tube).	Operational requirement.
Wrap H-3 chamber and drift tube portion in protective plastic sleeve.	Standard procedure for contamination control.
Determine if free liquid is present and absorb liquids, if present.	Required to meet WAC maximum free liquids requirement of 1%. This is particularly applicable to drift tubes and H-3 chamber waste.
Provide storage, waste segregation, consolidation and packaging capacity.	Items meeting the “standard debris” definition are shipped in a roll-off. It is anticipated this container will be routinely stored in the waste staging and shipping building. This container could accommodate numerous years of drift tubes and H-3 chambers. Dependent upon final design of the unit, various components may be easily separated and amenable for disposal as standard debris. One roll-off is continuously stored in the waste staging and shipping building. Oversized items (non-standard debris) are shipped in a cargo container. One container is continuously on-site.
Provide capacity to load oversized debris into cargo container.	WAC requirement for oversized debris.
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.
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 Extraction Columns

Requirement	Basis
Hold extraction column in hot cell for two weeks.	Extraction column is highly radioactive when removed from active service. Hold time is needed to facilitate transfer to longer term hold-for-decay storage.
Provide safe storage outside of hot cell. • Shielding • Potential need for cooling • Potential noble gas emissions	Required to allow protected storage until columns meet LSA shipping requirements. Approximate hold time per column is 550 days to meet LSA shipping limits. Approximately hold time for Type A shipment is 400 days. Xe is generated via decay of tellurium to I and I to Xe.
Provide management controls.	Since five columns are in each hot cell post service, it is necessary to ensure the column that has been held for approximately two weeks is the one transferred.
Determine if free liquid is present and absorb liquids, if present.	Required to meet WAC maximum free liquids requirement of 1%.
Fill void space (if required) in accordance with the WAC.	Required to meet WAC requirement to minimize void space.

Table 11.2-4 Waste Methodology for Process Glassware

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

Table 11.2-6 Waste Methodology for [redacted]]

Requirement	Basis
Waste stream segregated from evaporator concentrate.	[redacted]]
Sample influent waste stream prior to solidification.	Sampling of influent waste stream is performed to gather initial characterization data.
Provide sufficient solidification hot cell capacity.	Treatment cell capacity needed for this waste stream is approximately 270 gallons per year or approximately 11 gallons per week.
Waste stream solidified.	Using Portland cement (or equivalent) with the potential need for small additions to treat sulfate. Approximate ratio of waste to cement is 0.5 to 0.7. Required to meet WAC maximum free liquids requirement for solidified waste forms (0.5% by volume).
Limit void space	WAC requirement to minimize void space.
Establish dedicated area in the waste staging and shipping building for final decay time and shipment consolidation.	Hold time can be accomplished through pre-treatment tankage or post treatment within the waste staging and shipping building.
Maintain records relative to drums in the storage area.	Drums may not be removed from the storage area before decaying to Class A and DOT limits.

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

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

11.3 RESPIRATORY PROTECTION PROGRAM

The facility uses process and engineering controls to control the concentration of radioactive material in air. The design of the heating, ventilation, and air conditioning system is presented in Sections 9a2.1 and 9b.1. However, there may be instances when it is not practical to apply process or other engineering controls. When it is not practical to control the concentrations of radioactive material in the air to values below those that define an airborne radioactivity area, other means are implemented to maintain the total effective dose equivalent (TEDE) ALARA. In these cases, the ALARA goal is met by an increase in monitoring and the limitation of intakes by one or more of the following means:

- Control of access.
- Limitation of exposure times.
- Use of respiratory protection equipment.
- Other controls, as available and appropriate.

If an ALARA analysis is performed to determine whether or not respirators should be used, safety factors other than radiological factors may be considered. The impact of respirator use on workers' industrial health and safety is factored into decisions to use respirators. If the decision is made to permit the use of respiratory protection equipment to limit the intake of radioactive material, only National Institute of Occupational Safety and Health (NIOSH) certified equipment is used. The respiratory protection program meets the requirements of 10 CFR 20, Subpart H, Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas.

The respiratory protection program includes the following elements:

- Air sampling to identify the potential hazard, select proper equipment and estimate doses.
- Surveys and, when necessary, bioassays to evaluate actual intakes.
- Performance testing of respirators for operability (user seal check for face sealing devices and functional check for others) immediately prior to each use.
- Written procedures for:

    - Monitoring, including air sampling and bioassays.
    - Supervision and training of respirator users.
    - Fit testing.
    - Respirator selection.
    - Breathing air quality.
    - Inventory and control.
    - Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment.
    - Record keeping.
    - Limitations on periods of respirator use and relief from respirator use.

- Determination by a physician that the individual user is medically fit to use respiratory protection equipment:

    - Before the initial fitting of a face sealing respirator.
    - Before the first field use of non-face sealing respirators.

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

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

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

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

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

11.4 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT TECHNICAL SPECIFICATIONS

Potential variables, conditions, or other items that are probable subjects of a technical specification associated with the Radiation Protection Program and waste management are provided in Chapter 14.

11.5 REFERENCES

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